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

November 14, 2005

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 05000259/2005008

Dear Mr. Singer:

On October 15, 2005, 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 November 7, 2005, with Mr. Jon Rupert 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 05000259,260,296/2005004, issued October 27, 2005, is the most recent Unit 2 and 3 Integrated Quarterly Report. Although that report did not contain any site inspections in these cornerstones, they will continue to be documented in ROP integrated quarterly reports such as that one.

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. A significant portion of your engineering activities, Unit 1 Recovery Special Program implementation, and modification activities were reviewed during this inspection period and found to be effective with no significant problems identified. However, based on the results of this inspection, two Severity Level IV violations of NRC requirements were identified resulting from failure to install all parts required for four splices on multi-conductor cables and inadequate measures to assure that the design change documents for installation of straps for thermal overloads on 480-volt breakers were correctly translated into work instructions. However, the NRC is treating these findings as non-cited violations (NCVs) consistent with Section VI.A of the NRC Enforcement Policy. If you contest the NCVs in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at Browns Ferry Nuclear Plant.

Overall, we primarily found only minor discrepancies, generally 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 thermal overloads and cable splices. In addition, due to the small sample of completed activities in these areas inspected and the number of new splices and thermal overloads that were incorrectly installed, additional inspections will be required to determine if these Special Programs were being implemented satisfactorily.

Based on current and previous inspections of Unit 1 Recovery activities associated with five 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 Intergranular Stress Corrosion Cracking, Drywell Steel Platforms and Upper Drywell Platforms, Conduit Supports, Cable Tray Supports, and Configuration Management/Design Baseline. 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/

Stephen J. Cahill, Chief
Reactor Projects Branch 6
Division of Reactor Projects

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

Enclosure: Inspection Report 05000259/2005008
w/Attachment: Supplemental Information

cc w/encl: (See page 3)

cc w/encl:
Ashok S. Bhatnagar
Senior Vice President
Nuclear Operations
Tennessee Valley Authority
Electronic Mail Distribution

Larry S. Bryant, General Manager
Nuclear Engineering
Tennessee Valley Authority
Electronic Mail Distribution

Brian O' Grady
Site Vice President
Browns Ferry Nuclear Plant
Tennessee Valley Authority
Electronic Mail Distribution

Robert J. Beecken, Vice President
Nuclear Support
Tennessee Valley Authority
Electronic Mail Distribution

General Counsel
Tennessee Valley Authority
Electronic Mail Distribution

John C. Fornicola, Manager
Nuclear Assurance and Licensing
Tennessee Valley Authority
Electronic Mail Distribution

Bruce M. Aukland, Plant Manager
Browns Ferry Nuclear Plant
Tennessee Valley Authority
Electronic Mail Distribution

Glenn W. Morris, Manager
Corporate Nuclear Licensing
and Industry Affairs
Tennessee Valley Authority
Electronic Mail Distribution

William D. Crouch, Manager
Licensing and Industry Affairs
Browns Ferry Nuclear Plant
Tennessee Valley Authority
Electronic Mail Distribution

State Health Officer
Alabama Dept. of Public Health
RSA Tower - Administration
Suite 1552
P. O. Box 303017
Montgomery, AL 36130-3017

Chairman
Limestone County Commission
310 West Washington Street
Athens, AL 35611

Jon R. Rupert, Vice President
Browns Ferry Unit 1 Restart
Browns Ferry Nuclear Plant
Tennessee Valley Authority
P. O. Box 2000
Decatur, AL 35609

Robert G. Jones, Restart Manager
Browns Ferry Unit 1 Restart
Browns Ferry Nuclear Plant
Tennessee Valley Authority
P. O. Box 2000
Decatur, AL 35609

Distribution w/encl: (See page4)

Distribution w/encl:
M. Chernoff, NRR
E. Brown, NRR
L. Slack, RII EICS
RIDSRIDSNRRDIPMLIPB
PUBLIC

☐ SISP REVIEW COMPLETE: Initials: ________ ☐ SISP REVIEW PENDING*: Initials: ________ *Non-Public until the review is complete
☐ PUBLICLY AVAILABLE ☐ NON-PUBLICLY AVAILABLE ☐ SENSITIVE ☐ NON-SENSITIVE
ADAMS: ☐ Yes ACCESSION NUMBER:________________________

OFFICE	DRP/RII	DRP/RII	DRS/RII	DRS/RII	DRS/RII	DRS/RII	DRS/RII
SIGNATURE	WCB	EXC	SJV	LSM	NXM	JLL per email	JDF per email
NAME	WBearden	EChristnot	SVias	LMellen	NMerriweather	JLenahan	JFuller
DATE	11/14/2005	11/14/2005	11/09/2005	11/09/2005	11/09/2005	11/09/2005	11/09/2005
E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO
PUBLIC DOCUMENT	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO

OFFICE	DRS/RII	DRS/RII	DRS/RII	DRS/RII	DRS/RII		
SIGNATURE	GTH	LSM for	CJF per email				
NAME	GHopper	RBernhard	CFong				
DATE	11/09/2005	11/09/2005	11/09/2005				
E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO
PUBLIC DOCUMENT	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO

OFFICIAL RECORD COPY DOCUMENT NAME: E:\Filenet\ML053180487.wpd

U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No:	50-259
License No:	DPR-33
Report No:	05000259/2005008
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Browns Ferry Nuclear Plant, Unit 1
Location:	Corner of Shaw and Nuclear Plant Roads Athens, AL 35611
Dates:	July 17 - October 15, 2005
Inspectors:	W. Bearden, Senior Resident Inspector, Unit 1 E. Christnot, Resident Inspector R. Bernhard, Senior Reactor Analyst (Section E1.12) S. Vias, Senior Reactor Inspector (Sections E1.13, E1.14, E8.3) L. Mellen, Senior Reactor Inspector (Section E1.12, E8.2) G. Hopper, Senior Reactor Inspector (Section E1.12) J. Lenahan, Senior Reactor Inspector (Section E1.10, E1.11, E1.15, E8.4) N. Merriweather, Senior Reactor Inspector (Sections E1.6, E1.7, E1.8, E1.9) J. Fuller, Reactor Inspector (Sections E1.15, E1.16, E8.1) C. Fong, Reactor Inspector (Section E8.2)
Approved by:	Stephen J. Cahill, Chief Reactor Project Branch 6 Division of Reactor Projects

Enclosure

EXECUTIVE SUMMARY

Browns Ferry Nuclear Plant, Unit 1
NRC Inspection Report 05000259/2005008

This integrated inspection included aspects of licensee engineering and modification activities associated with the Unit 1 recovery project. This report covered a 3-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 configuration management/design baseline; fuses; cable splices; environmental qualification of electrical equipment; thermal overloads; small bore piping and instrument tubing; drywell steel platforms and upper drywell platforms; control rod drive insert and withdrawal piping; seismic class II over I, spatial system interactions and water spray; cable tray supports; conduit supports; intergranular stress corrosion cracking; 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 five 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 six permanent plant design changes were observed and found to be performed in accordance with the documented requirements. (Section E1.1)

- Activities associated with five temporary alterations which affected emergency lighting, drywell outage chill water, Reactor Water Cleanup System, and Residual Heat Removal Service Water System did not cause any significant impacts on the operability of equipment required to support operations of Units 2 and 3. (Section E1.2)

- System Return to Service activities continued to be performed in accordance with procedural requirements. Licensee walkdowns performed for system turnover were aggressive and comprehensive as evident by the identification of slope problems and other related deficiencies. The ongoing walkdowns identified deficiencies which would need to be resolved prior to performance of system restart testing. Any system deficiencies were identified and appropriately addressed by the licensee's corrective action program. (Section E1.3)

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

- Electrical cable installation activities were performed in accordance with documented requirements. Control of cable pull tension was adequate and cable installation bend radius and cable training radius were properly monitored. (Section E1.5)

- Based on limited reviews, the inspectors concluded that corrective actions to resolve the problems with mis-application of current-limiting fuses are acceptable. The program is equivalent in scope to those previously applied to the restart of the other units at Browns Ferry. However, additional inspection will be required to verify that the Fuse Special Program is being implemented adequately. (Section E1.6)

- The licensee identified a failure by engineering to identify specific Raychem parts needed to make four multi-conductor cable splices which was determined to be a non-cited violation. This resulted in splice installation problems; a modification installed four splices on multi-conductor cables without a required Raychem Breakout Boot being installed over the spliced cables. The lack of familiarity by engineering with the requirements for breakout boots for splices of multi-conductor cables was the apparent cause of the violation. Licensee corrective actions associated with this issue were reviewed and were deemed to be adequate. (Section E1.7)

- Raychem cable splices examined during this inspection were found to be installed in accordance with plant procedures and the manufacturer's instructions. The splices were in good material condition with no visible signs of degradation. However, due to the small sample of splices inspected (4 out of 522) and the number of new splices that were found incorrectly installed, additional inspections will be required to determine if the cable splice Special Program is being implemented satisfactorily. (Section E1.7)

- The inspectors concluded that the licensee's program for the environmental qualification of electrical equipment on Unit 1 is consistent with that used for restart of Units 2 and 3. However, additional inspection will be necessary by the NRC to verify that this Special Program is being implemented adequately. (Section E1.8)

- A non-cited violation was identified for failure to assure that the design changes for strapping out the thermal overloads in 480-Volt Reactor motor-operated valve (MOV) Boards were correctly translated into work instructions. This resulted in the as-built configuration for thermal overloads in the plant deviating from design drawings. Due to the need to inspect additional thermal overloads and because a number of previously installed thermal overloads were incorrectly installed, additional inspections will be required to determine if this Special Program is being implemented satisfactorily. (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. However, additional samples will need to be inspected prior to the NRC being able to reach a conclusion on closure of this Special Program. (Section E1.10)

- The inspectors determined that a modification for control rod drive (CRD) hydraulic drive piping supports was implemented in accordance with design requirements. However, additional samples of supports will be inspected prior to closure of the CRD piping Special Program. (Section E1.11)

- The inspectors concluded that the licensee had developed an acceptable program to maintain configuration management and control of essential calculations and re-establish the design basis for Unit 1. Based on this inspection, no further inspections are anticipated for the Configuration Management/Design Baseline Special Program. (Section E1.12)

- The licensee's self-assessments associated with the Configuration Management/Design Baseline Special Program were found to be aggressive and effective. (Section E1.12)

- Seismic Class II over Class I/Spatial System Interactions and Water Spray Special Program activities were performed in accordance with documented requirements. Completed or planned actions to address this Special Program for Unit 1 are consistent with those previously performed for Units 2 and 3. No issues related to this Special Program that would negatively impact restart of Unit 1 were identified. Based on this inspection, no further inspections are anticipated in this area. (Section E1.13)

- Cable Tray and Conduit Supports activities continued to be performed in accordance with documented requirements. Completed or planned actions to address these Special Programs for Unit 1 are consistent with those previously performed for Units 2 and 3. No issues related to cable tray and conduit supports that would negatively impact restart of Unit 1 were identified. Based on this and previously documented NRC inspections, no further inspections are anticipated for these Special Programs. (Section E1.14)

- The inspectors determined that corrective actions to resolve deficiencies identified in design and construction of the drywell structural steel for Unit 1 are adequate and consistent with those previously performed for Units 2 and 3. No further inspections of the Drywell Steel Platforms and Upper Drywell Platforms Special Program are anticipated. (Section E1.15)

- The inspectors determined that TVA had developed a thorough program to mitigate the long-term effects of Intergranular Stress Corrosion Cracking (IGSCC) on Unit 1. Completed or planned actions to address these issues for Unit 1 are consistent with those previously performed for Units 2 and 3. Ongoing licensee inspection activities of the Reactor Vessel Internals have not identified any issues that would negatively impact the restart of Unit 1. Therefore, no further inspections are anticipated for this Special Program. (Section E1.16)

- The inspectors concluded that the licensee's Inservice/Preservice Inspection (ISI/PSI) Program was meeting applicable regulatory requirements and licensing commitments, and that the licensee has an appropriate threshold for identification and resolution of ISI/PSI issues. (Section E1.17)

- Observation of ongoing electrical testing and licensee inspection activities performed at the licensee's offsite Power Service Shop indicated that work at that location continued to satisfy regulatory requirements. Work included major equipment overhauls, such as large pump motors. Recovery activities at that location involved a high level of professionalism. No violations or deviations were identified. (Section E1.18)

- The licensee's program for oversight of Unit 1 recovery activities performed at the Power Service Shop was well planned and Nuclear Assurance assessors were knowledgeable of the applicable work document requirements, programs, and work processes for work performed at that location. (Section E7.1)

Inspection Results - Maintenance

- The Maintenance organization continued to provide appropriate and comprehensive repairs to Unit 1 components which do not require design changes to support the Unit 1 Restart. (Section M1.1)

- Ongoing actions to flush the RWCU System were adequate to support removal of remaining purge dam material and to ensure that the system satisfied documented cleanliness requirements. (Section M1.2)

- The licensee's program for foreign material exclusion while working in the Reactor Pressure Vessel, reactor cavity, or fuel storage pool satisfied regulatory requirements and licensee commitments. The inspectors concluded that the licensee's plan to perform extensive RPV inspections and retrieval of foreign material debris prior to completion of in-vessel activities was an appropriate safety decision. (Section M1.2)

Table of Contents

II. Engineering 1

E1 Conduct of Engineering 1
E1.1 Permanent Plant Modifications 1
E1.2 Temporary Plant Modifications 5
E1.3 System Return to Service Activities 8
E1.4 System Restart Testing Program Activities 10
E1.5 Special Program Activities - Cable Installation and Cable Separation . 12
E1.6 Special Program Activities - Fuse Program 13
E1.7 Special Program Activities - Cable Splices 14
E1.8 Special Program Activities - Environmental Qualification of Electrical Equipment 17
E1.9 Special Program Activities - Thermal Overloads 18
E1.10 Special Program Activities - Small Bore Piping and Instrument Tubing 20
E1.11 Special Program Activities - Control Rod Drive Insert and Withdrawal Piping 22
E1.12 Special Program Activities - Configuration Management/Design Baseline 22
E1.13 Special Program Activities - Seismic II/I Spatial System Interactions and Water Spray 24
E1.14 Special Program Activities - Cable Tray Supports and Conduit Supports 25
E1.15 Special Program Activities - Drywell Steel Platforms and Upper Drywell Platforms 27
E1.16 Special Program Activities - Intergranular Stress Corrosion Cracking . 28
E1.17 Inservice/Preservice Inspection 30
E1.18 Power Service Shop Activities 32

E7 Quality Assurance in Engineering Activities 33
E7.1 Licensee Quality Assurance Oversight of Recovery Activities (Identification and Resolution of Problems) 33

E8 Miscellaneous Engineering Issues 34
E8.1 (Closed) GL 94-03, Intergranular Stress Corrosion Cracking of Core Shrouds in BWRs 34
E8.2 (Closed) GL 88-14, Instrument Air Supply System Problems Affecting Safety-Related Equipment 34
E8.3 (Closed) GL 96-06, Assurance of Equipment Operability and Containment Integrity During Design Basis Accident Conditions 35
E8.4 (Closed) Unresolved Item 259/87-26-03 RHR Pump Suction and Nozzle Load Allowables Are Not Exceeded 37
E8.5 (Closed) TMI Action Item II.K.3.27, Common Reference Level for BWRs 37

III. Maintenance 38
M1 Conduct of Maintenance 38
M1.1 Maintenance Program 38
M1.2 System Cleanliness and Flushing Activities 39

V. Management Meetings 41
X1 Exit Meeting Summary 41

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, re-installation of plant equipment and structures continued. Recovery activities include ongoing replacement of small bore piping and instrument tubing in the drywell; re-installation 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. Limited system return to service activities continued during this reporting period. The licensee continued mechanical stress improvement of new piping welds and completed the Boiling Water Reactor Vessel and Internals Project examinations of reactor vessel internals during the reporting period.

II. Engineering

E1 Conduct of Engineering

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

a. Inspection Scope

The inspectors reviewed planned Design Change Notice (DCN) packages associated with the installation of a new digital feedwater control system, modifications to the residual heat removal service water (RHRSW) system, residual heat removal (RHR) system, 480-VAC distribution, and 4160-VAC distribution. 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 Control Room Design Review (CRDR) activities in Panel 1-9-25-32, modification activities to the primary containment, RHRSW system, Control Rod Drive (CRD) System, and 480-VAC distribution. 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 51076

DCN 51076, Reactor Feedwater Instrumentation and Control - Control Bay, System 3, is intended to implement the modifications recommended for the reactor feedwater system in the Control Bay. This DCN also impacted the Reactor Recirculation, System 68. The DCN consisted of removal of the current Division I and Division II feedwater and recirculation control systems; installation of new digital control systems; installation of new conduits and cable; and abandonment of old cables and conduits.

DCN 51090

DCN 51090, 480-VAC Reactor Motor-Operated Valve (RMOV) Boards and 480-VAC Shutdown Boards - Control Bay, Systems 57- 4, is intended to implement the electrical modifications recommended for the 480-VAC distribution system in the Control Bay. The DCN consisted of 77 stages affecting both RMOV boards. Modifications to RMOV Boards 1A and 1B included replacement of circuit breakers, fuses, and thermal overloads (TOLs) in selected compartments. Due to the abandonment of RMOV Boards 1D and 1E, selected buckets, fuses, trip units, and circuit breakers will be moved to RMOV Board 1A and 1B. In addition, this DCN will also change selected relay settings on Shutdown Transformers TS1A and TS1B; replace selected cables in the Control Bay HVAC System; replace and abandon selected cables in Primary Containment System for Unit 2 and Unit 3; add cables and a switch, for an alternate feed, to battery charger SB-C from RMOV Board 2A; replace and abandon selected cables in Standby Gas Treatment; and modify wiring on the transfer switches for Diesel Auxiliary Boards A and B. Systems affected by this DCN included System 64, Primary Containment; System 65, Standby Gas Treatment; System 68, Reactor Recirculation; System 69, Reactor Water Clean Up (RWCU); System 71, RCIC; System 73, HPCI; System 74, RHR; and System 74, Core Spray.

DCN 51217

DCN 51217, 4160 VAC Distribution - Reactor Building, System 57-5 is intended to implement electrical modifications recommended for the shutdown boards in the reactor building. Scheduled modifications for this DCN included installation of new conduits and junction boxes; reworking old conduits; determination and abandoning old cables; installation and termination of new cables; labeling new cables; and labeling old cables as abandoned. Plant equipment affected by this DCN were Transformers TDE and TS1E and RHRSW pumps A1 and A2.

DCN 51177

DCN 51177, RHRSW - Reactor Building, System 23, is intended to implement the modifications recommended for the RHRSW system in the reactor building. Scheduled modifications for this DCN included installation of packing, new valves, and new valve actuators for selected flow control valves; installation of new relief valves and thermal wells in selected piping; installation of new conduits, junction boxes, and electrical cables; modification of Containment Purge, System 64B, duct supports to avoid interference with modified RHRSW piping; and upgrading three dresser couplings located inside the service water tunnel.

DCN 51199

DCN 51199, RHR - Reactor Building, System 74, is intended to implement the modifications recommended for the RHR system in the reactor building. Scheduled modifications included removal and replacement of instrumentation, tubing, manifolds, and drain valves in the Division I and Division II instrumentation panels; verification or adjustment of slope of instrument sensing lines; permanent removal of Division I to Division II crosstie valve 1-FCV-74-46; installation of packing and new valve actuators for selected valves; replacement of selected Division I and Division II instrumentation; replacement of asbestos caskets on the heads of heat exchangers 1A and 1C; and installation of large-bore and small-bore pipe supports as required.

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. Observations of any post-modification testing activities are discussed in Section E1.4.

DCN 51177

The inspectors reviewed and observed permanent plant modification activities associated with DCN 51177, RHRSW - Reactor Building, System 23, which involved the RHRSW pumps and piping to the heat exchangers. The activities were controlled by Work Orders (WOs) 02-015487-001, 02-015487-087, 02-015487-091, and 03-019416-091. Activities observed by the inspectors included installation of splices from the old cables to the new cables inside the reactor building for the A1 pump; termination of the power cables to the A1 pump inside 4160V AC shutdown board 1A; performance of bump testing of A1 and A2 pump motors to check for rotation; and installation of new design flow control valves, 1-FCV-23-34 and 1-FCV-23-40, for the RHRSW discharge of the 1A and 1C RHR heat exchangers.

DCN 51090

The inspectors reviewed and observed permanent plant modification activities associated with DCN 51090, 480-VAC - Control Bay, System 57-4, which involved the 480-VAC RMOV Boards 1A and 1B, including activities in support of transferring electrical loads from the 480-VAC RMOV Boards 1D and 1E to RMOV Boards 1A and 1B in preparation for the abandonment of RMOV Boards 1D and 1E. The activities were controlled by WOs 02-021841-054, 02-021841-035, 02-021841-042, 02-021841-076, 02-021841-080, and 02-021841-091. Activities observed in RMOV Board 1A included installation of new breakers in cubicle 19E, Unit 1 Preferred Inverter, System INUT-252-1A; cubicle 20A recirculation Pump 1B discharge valve 1-FCV-68-79; in cubicle 19A, RHR Loop I inboard injection valve 1-FCV-74-53; cubicle 20C, RHR System I suppression pool cooling and test valve 1-74-FCV-59; and cubicle 20E, RHR System I minimum flow valve 1-FCV-74-07. Activities observed in RMOV Board 1B included installation of breakers in cubicle R11B, RHR test valve 1-FCV-74-73; cubicle 3A, RHR LPCI outboard injection valve 1-FCV-74-66; and cubicle 7C, RHR shutdown cooling valve 1-FCV-74-36.

DCN 51240

The inspectors reviewed and observed permanent plant modification activities associated with DCN 51240, Control Rod Drive - Reactor Building, System 85, which involved the Anticipatory Transient Without Scram (ATWS) System. The activities were controlled by WOs 03-006607-010, 03-006607-026, 03-006607-041, and 03-006607-057. Among the activities were the following: the installation of the two ATWS panels 1-LPNL-925-0416, ATWS Channel A 250-VDC Logic Panel and 1-LPNL-925-0613, ATWS Channel B 250-VDC Logic Panel; installation of panel internal relays, terminal boards, and wiring; installation of panel external switches and indicating lights; and the labeling of the various components.

DCN 51106

The inspectors reviewed and observed permanent plant modification activities associated with DCN 51106, CRDR for Auxiliary Instrument Control Panel 1-9-25-32, which involved the Remote Shutdown System. The activities were controlled by WOs 02-011701-08, 02-011701-09, 02-011701-17, and 02-011701-20. Activities observed included relocation of the System 3, Feed Water, hand switches 1-HS-3-98C and 99C, and transfer switches 1-XS-3-98 and 99; relocation of switches, replacement and modification of switch handles for 1-XS-43-14 and 1-HS-43-14C; replacement of switch handles and escutcheons for Raw Cooling Water, System 24; changing of indicator and scale for 1-PI-23-61 and re-identification as 1-PI-23-61/1, RHR Service Water, System 23; and replacement of switch handles and escutcheons for RBCCW, System 70.

DCN 51216

The inspectors reviewed and observed permanent plant modification activities associated with DCN 51216, 480-V Distribution - Reactor Building, System 57-4, which involved transformer TS1E, Alternate Power to 480-V Shutdown Boards 1A and 1B; transformer TDE, Alternate Power to 480-V Diesel Auxiliary Boards A and B; and 480-V RMOV 1A. The activities were controlled by WOs 03-023424-06, 03-001001-49, and 03-001001-51. Activities observed included installation and termination of alternate power cables to 480-V Shutdown Boards 1A and 1B; installation and termination of alternate power cables to 480-V Diesel Auxiliary Boards A and B; reworking of 4160-V Shutdown Boards 1B, cubicle 14, to install a bus bar and fuse holder; installation and termination of power cables to 4160-V/480-V transformers TDE and TS1E, and the installation of a cable penetration seal and termination of alternate power cables to 480-V RMOV 1A.

DCN 51189

The inspectors reviewed and observed permanent plant modification activities associated with DCN 51189, Primary Containment - Reactor Building, System 64A, which involved the installation of the Hardened Wet Well Vent (HWWV). The activities were controlled by WOs 02-016916-02, 02-016916-04, 02-016916-44, 02-016916-46, and 02-016914-65. Activities observed included selected portions of installation of the 14-inch HWWV piping from the suppression chamber 20-inch torus vacuum relief piping to the common HWWV header; addition of supports for the 14-inch HWWV piping; and installation of the electrical components associated with the flow solenoid valves for flow control valves 1-FCV-64-221 and 1-FCV-64-222.

c. Conclusions

The inspector's review of modification design packages associated with five 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 six 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 with Outage Chill Water, Unit 1 Appendix R Emergency Lighting, RWCU, and RHRSW systems. 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 Appendix R Emergency Lighting, fuses for RWCU isolation valves, temporary chill water for the reactor building, and the RHRSW system. The temporary alterations potentially impacted Secondary Containment and involved the temporary removal of power to emergency lighting, re-installation of fuses for power to allow testing of the RWCU pumps, removal of a temporary pressure boundary for RHRSW to support the installation of new valves and replacement piping, removal of the piping jumper installed in RHRSW supply piping, and installation of temporary chill water to the reactor building 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:

- TACF 0-04-005-247, Appendix R Battery Emergency Lighting, System 247, was initiated to temporarily remove and re-route power to Appendix R battery powered emergency lighting fixtures to support the installation of DCN 51177, RHRSW - Reactor Building, System 23. The TACF will be removed after the installation of affected portions of DCN 51177 are completed. Four Appendix R battery powered emergency lighting fixtures, 1-LGT-247-RB026, 1-LGT-247-RB027, 1-LGT-247-RB029, and 1-LGT-247-RB030, located in the reactor building were impacted by this TACF. The inspectors observed installation activities for this TACF and verified that required testing in accordance with test procedure FP-0-247-INS004, Appendix R Battery Operated Emergency Lighting Quarterly Test, was performed following installation.

- TACF 1-04-013-069, Revision 1, RWCU Inboard and Outboard Primary Containment Isolation System (PCIS) Relays. Revision 0 to this TACF had been previously installed for lifted leads, jumpers, and pulled fuses in Panels 1-9-42 and 1-9-43 to allow for the operation of certain RWCU valves with PCIS de-energized. Fuses pulled were 16A-F17 and 16A-F18 and lifted leads and jumpers were on relay 16A-K26, located in panel 1-9-42, and on relay 16A-K27, located in panel 1-9-43. RWCU isolation valves affected included 1-FCV-69-1, 1-FCV-69-2, and 1-FCV-69-12. The inspectors observed installation of Revision 1 of this TACF which included re-installation of inboard PCIS logic fuse 16A-F17 to provide power to the instrumentation system for the RWCU pumps and allow the pumps to be operated during system testing.

- TACF 0-05-005-044, affected the System 44, Building Heating; System 64, Secondary Containment; System 70, Reactor Building Closed Cooling Water (RBCCW); and the temporary connection to outage chill water. This TACF was initiated to install piping connections from outage chill water in the Unit 1 Drywell to the reactor building heating system. The drywell outage chill water piping enters secondary containment and connects into RBCCW system to provide chill water for the drywell coolers. This TACF installed two flexible piping connections and six valves between the supply and return lines for both the building heating to the Unit 1 reactor building and the drywell chill water systems. The purpose of the six valves was to isolate or restore the reactor building heating system, when necessary, to the Unit 1 reactor building. The inspectors observed installation of selected portions of this temporary alteration. The inspectors noted that secondary containment was maintained at all times during the installation of this TACF.

- TACF 1-04-003-023, had been previously issued to install temporary piping in the Unit 1 RHRSW loop A/C to allow for the flushing of the loop. Flow through the old permanent system piping could potentially affect the cleanliness of the new system piping and the 1A and 1C RHR heat exchangers. This piping jumper provided a closed loop flow path from the separate A/C RHRSW supply headers to the common loop A/C discharge header as a means of flushing the supply headers. The installation of TACF 1-03-002-023 was documented in NRC Inspection Report 259/2004-08. This short term piping jumper supported the implementation of DCN 51177, RHRSW - Reactor building, System 23, modification activities on the A/C RHRSW headers. The flushing activities on the A/C RHRSW headers were completed. Prior to the removal of the piping jumper a blank flange was placed inside the RHRSW piping and this TACF was closed. The inspectors reviewed and observed selected portions of the removal of the TACF and activities associated with returning the system to normal status. The inspectors verified that secondary containment pressure boundary was maintained during the modification removal activities.

- TACF 1-03-002-023 had been previously installed to provide a secondary containment pressure boundary barrier during modification activities on RHRSW valves 1-HCV-23-31 and 1-HCV-23-37. NRC review of installation of this TACF was documented in NRC Inspection Report 259/2004-06. This TACF supported the implementation of DCN 51177, RHRSW - Reactor Building, System 23, modification activities. Modification activities on these valves have been completed and TACF 1-03-002-023 was no longer required and was removed. The TACF was closed. The inspectors observed removal activities associated with selected portions of this temporary alteration and activities associated with returning the system to normal status. The inspectors verified that the secondary containment pressure boundary was maintained during the modification removal activities.

c. Conclusions

The inspectors determined that activities associated with the five temporary alterations which affected emergency lighting, outage chill water, the RWCU System, and the RHRSW System 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)

a. Inspection Scope

The inspectors reviewed and observed 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.

SRTS activities reviewed and observed by the inspectors during this reporting period were focused on System 23, RHRSW, and System 67, Emergency Equipment Cooling Water (EECW). Both systems share the same twelve pumps, which were maintained as operable to support Units 2 and 3. Normal operational lineup is with four pumps aligned for RHRSW, four pumps aligned for EECW, and four pumps in standby.

b. Observations and Findings

SRTS activities continued but were limited to RHRSW and EECW. No other significant SRTS activities occurred during the reporting period. 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-Operation 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. The SRTS activities reviewed and observed by the inspectors included completion of SPAE and SPOC I processes for these two systems. The inspector reviewed and observed portions of the licensee's SRTS activities for the following:

- System 23, RHRSW, which included completion of the SPAE and SPOC Phase I activities. Specific areas reviewed included closure activities for DCN 51101, CRDR, repairs and modifications to CR Panel 1-9-21; DCN 51177, RHRSW - Reactor Building; DCN 51199, RHR - Reactor Building, System 74; DCN 51336, RHRSW - Reactor Building, installation of large bore system piping supports; and DCN 51409, RHRSW - Reactor Building, installation of small bore system piping supports. Ongoing work activities observed and reviewed included WO 02-011699-04, installation of Temperature Recorder 1-TR-74-80 in CR panel 1-9-20; WO 05-7115148-00, verification of the operation of flow control valve 1-FCV-23-34, RHR heat exchanger 1A outlet; WO 05-7115149-00, verification of the operation of flow control valve 1-FCV-23-40, RHR heat exchanger 1C outlet; WO 02-015487-43, termination and testing of cables in 480-V RMOV Board 1A compartment 5D; and WO 02-015487-65, installation, termination and testing cables associated with flow control valve 1-FCV-23-46, RHR heat exchanger 1B outlet.

- System 67, EECW, which included completion of the SPAE and SPOC Phase I activities. Specific areas reviewed included closure activities for DCN 51192, EECW - Reactor Building; DCN 51100, CRDR, repairs and modifications to CR Panel 1-9-20; DCN 51106, CRDR, repairs and modifications to remote Panel 1-25-32; DCN 51340, EECW - Reactor Building, installation of large bore system piping supports; DCN 51411, EECW - Reactor Building, installation of small bore system piping supports; and DCN 51182, Control Air - Reactor Building, System 32. Ongoing work activities observed and reviewed included WO 02-013229-01, replacement of valves on header to core spray room coolers 1A and 1B; WO 02-013229-24, installation of new stainless steel piping on EECW in the reactor building; and WO 02-013229-27, work associated with flow control valve 1-FCV-67-50.

Activities observed included periodic meetings to discuss the SRTS status of both systems, which included the status of the SPOC I checklists, the status of the SPAE process, and status of outstanding work items and identified deficiencies. These activities also included observation of licensee walkdowns of portions of plant systems, review of various identified deficiencies that impacted the RHRSW and EECW systems. The inspectors verified that these deficiencies were documented as Problem Evaluation Reports (PERs) in the corrective action program and designated as required to be addressed as part of the SPOC I process. Specific PERs reviewed are included in the attachment. In addition, the inspectors noted that issues related to slope of instrumentation sensing lines and small bore piping were being identified during the ongoing walkdowns. Four PERs issued to document sensing line problems and inadequate slope were reviewed and are also included in the attachment.

c. Conclusions

SRTS activities continued to be performed in accordance with procedural requirements. The inspectors determined that the licensee's system turnover walkdowns were aggressive and comprehensive as evident by the identification of slope problems and other related deficiencies. The ongoing walkdowns identified a number of deficiencies which would need to be resolved prior to the performance of system restart testing. Any system deficiencies were identified and appropriately documented in the licencee's corrective action program.

E1.4 System Restart Testing Program Activities (37551)

a. Inspection Scope

The inspectors reviewed on-going activities associated with the Restart Test Program (RTP). Areas reviewed included testing CRDR modifications and changes to instrumentation associated with a RWCU flow control valve required by Appendix R. Testing was performed in accordance with Post-Modification Test Instructions (PMTIs).

b. Observations and Findings

b.1 Observation of Testing

The following PMTIs were developed and approved to test portions of the associated DCNs. Specific areas reviewed included observation of ongoing testing in the control room, Unit 1 auxiliary instrument room, and RWCU valve and heat exchanger rooms. Systems affected included System 23, RHRSW; System 24, Raw Cooling Water (RCW); System 25, Raw Service Water (RSW); System 26, High Pressure Fire Protection (HPFP); System 32, Plant Control Air (PCA); and System 69, RWCU. Pre-test briefings were held, assignments were made, and communications were established prior to performance of testing. The inspectors observed and reviewed portions of the following testing:

- 1-PMTI-BF-023-055, tested Stage 28 of DCN 51094. This DCN was part of the CRDR program for the RHRSW system. Stage 28 consisted of modifications to hand switches 1-HS-23-94A/1 located on Control Room (CR) Panel 1-9-3. Testing was intended to demonstrate that the hand switch performed its design function. The inspectors observed portions of ongoing testing which included starting and stopping RHRSW Pump D3 using the hand switch, simulation of a motor trip-out by depressing the local breaker trip button while the motor/pump was operating. The inspectors verified that the associated indicating lights functioned correctly and that acceptance criteria for the test were met. The inspectors determined that this testing successfully fulfilled the testing requirements for work performed under DCN 51094, Stage 28. There were no test exceptions.

- 1-PMTI-BF-51093-STG01, tested Stage 01 of DCN 51093. This DCN was part of the CRDR program for the HPFP system. Stage 01 consisted of moving hand switches 1-HS-26-98A and 1-HS-26-104A from CR Panel 1-9-2 to Panel 1-9-20. Testing was intended to demonstrate that the hand switches performed their design functions. The inspectors observed portions of ongoing testing and verified that acceptance criteria for the test were met. The inspectors determined that this testing successfully fulfilled the testing requirements for work performed under DCN 51093, Stage 01. There were no test exceptions.

- 1-PMTI-BF-51100-STG16, tested Stage 16 of DCN 51100. This DCN was part of the CRDR program. Stage 16 consisted of relocating hand switch 1-HS-24-10A and Amp meter 1-EI- 24-10 for RCW Pump B, and hand switch 1-HS-24-16A and Amp meter 1-EI-24-16 for RCW Pump D on CR Panel 1-9-20. Testing was intended to demonstrate that the components performed their design functions. The inspectors observed portions of ongoing functional testing of the hand switches which consisted of manual operation of the switches. The inspectors verified that the associated ammeters were calibrated by the use of an approved procedure and that acceptance criteria for the test were met. The inspectors determined that the testing successfully fulfilled the testing requirements for work performed under DCN 51100, Stage 16. There were no test exceptions. There was a test problem identified which is discussed in section b.2.

- 1-PMTI-BF-51177-STG03, tested Stage 01 of DCN 51177, RHRSW - Reactor Building, System 23. Stage 01 consisted of installing new cables to the newly installed flow control valve, 1-FCV-23-34, RHRSW Outlet Valve for RHR Heat Exchanger 1A. The PMTI also served as testing for DCN 51094. DCN 51094 was part of the CRDR program for CR Panel 1-9-3. Testing was intended to demonstrate that the flow control valve would open and close from the CR panel hand switch 1-HS-23-34A and from local control station switch 1-HS-23-34B. The inspectors observed functional testing of the valve which was performed by manual operation of the hand switches. Inspectors noted that this testing verified agreement between the local/remote indicating lights and actual valve position. The testing also verified the valve interlocks associated with the RHRSW Pumps A1 and A2. The inspectors determined that acceptance criteria for the test were met and that testing successfully fulfilled the testing requirements for work performed under DCN 51100, Stage 16 and DCN 51094, Stage 08. There were no test exceptions.

- 1-PMTI-BF- 51194-(1-FCV-69-94), Functional Test of Appendix R Required Valve, tested RWCU flow control valve 1-FCV-69-94. The valve was installed to meet Appendix R requirements by DCN 51194, RWCU Piping - Reactor Building. The control air tubing and associated valves were installed by DCN 51235, RWCU, System 69 and Sampling/Water Quality, System 43 - Reactor Building, Stage 01. The valve was required to close when the fusible link melts or when air is manually vented from the valve operator. The inspectors observed portions of ongoing testing which verified that the valve opened when air was aligned to the valve operator and closed when air was vented from the valve operator. In addition, testing verified that the valve closed when the fusible link was removed. The inspectors noted that red and green valve position indicating lights functioned as required. The inspectors determined that acceptance criteria for the test were met and that testing successfully fulfilled the testing requirements for work performed under DCN 51194 and DCN 51235, Stage 01. There were no test exceptions.

b.2 Review of Test Deficiencies

The inspector observed that the above testing resulted in no Test Deficiencies or PERs during the testing process. However, one minor test problem was identified for test 1-PMTI-BF-51100-STG16 in that Steps 6.2.[18] and 6.3.[19] were not performed as written. The steps required that the breakers for RCW Pumps B and D be reset by depressing the manual reset on the individual breakers. Previous steps, Steps 6.2.[15] and 6.3.[15], in the test procedure had reset the breakers when the respective hand switches were placed in the STOP position. The inspectors determined that this did not affect the test results or the acceptance criteria for this test.

c. Conclusions

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

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

a. Inspection Scope

The inspectors continued to observe and/or review the licensee's activities associated with the installation of electrical cables. The installation activities were controlled by modification WOs and licensee procedures. Among the procedures were the following: Modification and Addition Instruction (MAI) 1.3, General Requirements for Modifications; 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. Revision 45; and MAI-3.7, Cable Pull Force Monitoring Breaklink Fabrication, Verification, and Control.

b. Observations and Findings

The licensee continued to perform limited cable installation activities during this report period. These were mostly power distribution cables and load shed cables installed as part of DCN 51216, Electrical 480-V Distribution - Reactor Building, System 57-4. The activities associated with this DCN were completed during this report period. Activities observed or reviewed included:

- WO 03-001001-51, replace alternate feeder cables for the 480-V RMOV Board 1A

- WO 03-001001-53, replace alternate feeder cables for the 480-V RMOV Board 1B

- WO 03-023434-04, replace cables for the 4160/480-V Transformer TS1E, alternate power for 480-V Shutdown Board A and 480-V Shutdown Board B

- WO 03-001001-49, replace cables for the 4160/480-V Transformer TDE, alternate power for 480-V Diesel Auxiliary Board A and 480-V Diesel Auxiliary Board B

The inspectors observed selected portions of the ongoing cable installation activities and verified that installation activities were performed in accordance with the documented requirements. The inspector verified that control of cable pull tension was adequate (break link was used). In addition, the inspectors noted that the Quality Control inspectors and craft supervisors monitored cable installation bend radius and cable training radius by taking periodic measurements during the installation activities.

c. Conclusions

Electrical cable installation activities were performed in accordance with documented requirements. Control of cable pull tension was adequate and cable installation bend radius and cable training radius was properly monitored.

E1.6 Special Program Activities - Fuse Program (37550)

a. Inspection Scope

In Section III.13.6 of the BFN Nuclear Performance Plan, Revision 2, TVA described corrective actions for an electrical problem involving the mis-application of fuses that limit current in overload protection. The corrective action program as it was previously applied to support Units 2 and 3 restart contained the following actions:

- Revise the BFN fuse substitution program control document to reflect the appropriate standards.

- Perform calculations using revised design standards to specify the appropriate fuses for each application and document this activity on the fuse tabulation document.

- Conduct plant walkdowns to determine and document the installed fuses for compliance with the fuse tabulation, with the exception of motor control centers, where allowable substitution has been identified.

- Compare the results of the fuse tabulation with the walkdown for reconciliation.

- Document and resolve by the corrective action process all inadequate fuses.

- Delete and replace fuse ratings on design drawings with a fuse identification before restart. The fuse tabulation would be the single source of fuse requirements for the applicable fuses.

This inspection examined the fuse program activities that were being implemented for restart of Unit 1. The inspectors reviewed completed fuse replacement work order packages, fuse sizing calculations and design change packages, and conducted field walkdown inspections of a selected sample of completed breaker cubicles to verify that the correct fuses had been installed in accordance with design basis documents.

b. Observations and Findings

The inspectors examined the fuses in the 480-VAC Motor Control Center (MCC) breaker cubicles listed below:

- 480-VAC Reactor MOV Board 1A Cubicles 12A, 4D, 5D, R9D2, and 17E

- 480-VAC Reactor MOV Board 1B Cubicles 7A, 8A, 14C-2, 15C, 1E, 4E, 5E, 6E, 7E, and 8E

The inspectors examined the installed fuses in the referenced MCC breaker cubicles to verify that new replacement fuses were being installed in accordance with the Unit 1 Fuse Program. The inspectors performed walkdown inspections and compared the installed fuse nameplate data against work order records, design change notices, fuse sizing calculations, and the Master Equipment List to verify that the installed fuses were the correct size and type as specified by the design documents.

c. Conclusions

Based on limited reviews the inspectors concluded that corrective actions to resolve the problems with mis-application of current-limiting fuses were acceptable. The program is equivalent in scope to those previously applied to the restart of the other units at Browns Ferry. However, additional inspection will be required to verify that the Fuse Special Program is being implemented adequately.

E1.7 Special Program Activities - Cable Splices (37550)

a. Inspection Scope

This Special Program addresses issues associated with electrical cable splices and terminations in Environmental Qualification (EQ) applications. In 1986, the NRC issued Information Notice (IN) 86-53 alerting licensees to a potential safety problem involving improper installation of heat-shrinkable tubing over electrical splices and terminations. In addition to this IN, an employee concern had been brought up at BFN regarding problems with existing site procedures for installing electrical splices. Based on these concerns, TVA had initiated a comprehensive program at BFN to ensure the adequacy of all site class 1E electrical cable splices and terminations in harsh environments.

TVA’s comprehensive splice program as described in the Nuclear Performance Plan, Revision 2, required all splices and terminations subject to 10 CFR 50.49 to be inspected and replaced if the splices did not meet installation standards. This program was implemented as part of the restart effort on Units 2 and 3. The NRC staff reviewed the implementation of this program during the previous restarts of Units 2 and 3 and found it to be acceptable.

Unlike Units 2 and 3, the Unit 1 cable splice Special Program will require fewer walkdowns of existing cable splices since most of the splices are to be replaced prior to restart of Unit 1. The EQ cables and splices that are not scheduled to be replaced will be inspected and incorporated into the Unit 1 EQ Program through Unit 1 Restart DCNs similar to what was done on Units 2 and 3. The licensee indicated that there are approximately 522 Unit 1 EQ splices. Of those, 504 new EQ splices will be installed as part of the Unit 1 restart. The remaining 18 EQ cable splices (currently installed) will be incorporated, if acceptable, by documentation only changes to the EQ Program (i.e., EQ Change supplements).

During this inspection, the inspectors reviewed ongoing activities to implement the cable splice program for Unit 1 restart. This inspection examined completed cable splice WO packages and installation procedures, and included field walkdown inspections of selected Raychem heat shrink splices and terminations located both in the reactor building and drywell to determine if the splices were installed in accordance with manufacturer's instructions and plant installation procedures. The inspectors also reviewed the licensee's evaluation and corrective action for a PER involving four EQ splices that were installed by a modification and had incorrect material over the spliced cables.

b. Observations and Findings

The inspectors selected a sample of four Raychem heat shrink cable splices to review the licensee's implementation of the EQ splice program. The four cable splices consisted of two splices that had been installed during the Unit 2 and 3 restart effort in 1990 that were being retained as part of the Unit 1 restart effort. The other two splices were recently installed in June and August of 2005. The four splices were installed by the following WO and DCN packages:

- WO 03-019416-054, Installed Splice No. 0-$ES-023-0088A, 8/17/05
- WO 90-04666-00, Installed Splice No. 0-$ES-067-327A, 5/2/90
- WO 02-009389-008, Installed Splice No. 1-$PC-069-0306A, 6/6/05
- DCN W12179A WP 1078-90, Installed Splice No. 0-$ES-211-1838A, 8/21/90

The inspectors reviewed the referenced installation work records for the associated splices to determine if the splice configuration as well as the materials used to assemble the splices was selected in accordance with MAI 3.3 and the Raychem instructions for either the specific kits or other heat-shrink parts used. The inspectors conducted field walkdown inspections to verify that the as-built splices were installed consistent with installation procedures and work records, and to verify that the materials were properly applied with no visible signs of degradation.

The inspectors also reviewed PER No. 86817 that was entered into the licensee's corrective action program (CAP) on July 30, 2005. The licensee had conducted a self-assessment of completed splice WO Packages and determined that four EQ cable splices had been installed and accepted with a required breakout boot being omitted from the installed splice configuration. The splices were installed on multi-conductor cables 1RP2154-1A and 1RP2155-1A by WO 03-009542-094 and cables 1RP2160-IIB and 1RP2161-IIB by WO 03-009542-097.

The licensee determined that incorrect heat-shrink data sheets had been prepared and signed by field and design engineers during the work process. As a consequence, on the 25th and 26th of July, 2005, the workers used the incorrect heat-shrink data sheets to install and accept four splices. 10 CFR 50, Appendix B, Criterion V, Instructions, Procedures, and Drawings, in part, states that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. MAI-3.3 Paragraphs 6.2.11, 6.2.15, and 6.2.2.2.n.1, require engineering to specify on the heat shrink data sheet if a breakout boot is required. Contrary to the above, as of July 26, 2005, the licensee failed to specify on the heat-shrink data sheet that a breakout boot was required for the splices.

The licensee has concluded that the splice installation problems were limited to the four splices identified in the PER. The apparent causes for the installation deficiencies were determined to be the result of inadequate training in the area of field splices for field engineers in Maintenance/Modifications and Design Engineering Groups. The licensee's proposed corrective actions included re-training personnel on Raychem splices. The failure to provide adequate instructions in the heat-shrink data sheets for installing the splices is a violation of 10 CFR 50, Appendix B, Criterion V, Instructions, Procedures, and Drawings. This is a Severity Level IV Violation per the criteria in Supplement II, Facility Construction, in the NRC Enforcement Policy. Because this Severity Level IV violation was identified by the licensee and has been entered into the licensee's CAP (PER 86817), this violation is being treated as a Non-Cited Violation (NCV), consistent with Section VI.A of the NRC Enforcement Policy and will be identified as NCV 50-259/2005-008-01, Engineering Did Not Follow Procedures and Document on the Heat-Shrink Data Sheets All the Parts Required to Install Four Splices on Multi-Conductor Cables.

c. <u>Conclusions</u>

The failure by engineering to identify on the heat-shrink data sheet the specific Raychem parts needed to make four multi-conductor cable splices was identified as a licensee-identified non-cited violation. The lack of familiarity by engineering with the requirements in MAI-3.3 for breakout boots for splices of multi-conductor cables was the apparent cause of the violation. Corrective actions associated with this issue were reviewed and were deemed to be adequate.

The four Raychem cable splices examined during this inspection were found to be installed in accordance with plant procedures and the manufacturer's instructions. The splices were in good material condition with no visible signs of degradation. However, due to the small sample of splices inspected and the fact that a number of new splices were incorrectly installed due to inadequate training, additional inspections will be required to determine if the cable splice program was being implemented satisfactorily.

E1.8 Special Program Activities - Environmental Qualification of Electrical Equipment (37550)

a. Inspection Scope

The inspectors reviewed the status of the EQ program that is being implemented to support the Unit 1 restart.

b. Observations and Findings

The NRC had previously reviewed and accepted the EQ program that was implemented to support the restart of Units 2 and 3. The evaluation of the program is discussed in Section 3.2 of NUREG-1232, Volume 3, dated April 1989. In that evaluation, the staff concluded that the Browns Ferry equipment qualification program of electrical equipment located in harsh environments complies with the requirements of 10 CFR 50.49. The inspectors compared the Unit 1 EQ program to the Unit 2 and 3 programs to determine if they were equivalent. The Unit 1 EQ Program uses the same processes and procedures that are used for the Unit 2 and 3 EQ Programs. For example, the existing Unit 2 and 3 Equipment Qualification Data Packages (EQDPs) are being revised to address the Unit 1 EQ equipment including the Qualification Maintenance Data Sheets and the Field Verification Data Sheets. New EQDPs will also be issued for those Unit 1 EQ equipment items that are not currently included in the Unit 2 and 3 EQ Program.

The licensee plans to replace most of the Unit 1 EQ equipment including cables and splices prior to the Unit 1 restart. The Unit 1 EQ equipment that is not scheduled to be replaced is being added to the BFN EQ Program through EQ Change Supplement (EQCS) documents which are included in the Unit 1 Restart DCN packages. The number of existing cables and components that have been identified to be added to the BFN EQ Program for Unit 1 Restart are as follows:

- 352 cables
- 18 splices
- 23 saved components other than cables and splices

The licensee stated that for the 23 saved components, other than cables and splices, that are not being replaced, a review of the maintenance history will be performed to document that no unqualified piece parts or sub-components have been installed during maintenance activities.

The list of EQ equipment required to meet 10 CFR 50.49 will be contained in the Master Equipment List data base. The inspectors informed the licensee that additional inspections including plant walk downs will be required as work progresses to verify proper implementation of the EQ Program.

c. Conclusions

The inspectors concluded that the licensee's program for the environmental qualification of electrical equipment on Unit 1 is consistent with that used for restart of Units 2/3. However, additional inspection will be necessary to verify that the program was implemented adequately.

E1.9 Special Program Activities - Thermal Overloads (37550)

a. Inspection Scope

In Section III.13.4 of the BFN Nuclear Performance Plan, Revision 2, TVA described a design control problem with the application of thermal overloads (TOLs) in 480-VAC and 250-VDC motor control centers. The corrective action program as it was applied to support Units 2 and 3 restart contained the following actions:

- Inspect the 480-VAC and 250-VC safety-related motor control centers to determine and document the installed TOL ratings.
- Develop and issue a sizing criteria for TOLs.
- Evaluate the walkdown results against the sizing criteria.
- Replace or reset improperly sized TOL elements, as appropriate.
- Properly sized or replaced TOLs will be documented on a TVA design drawing to assure that current and future installations of thermal overloads are correct.
- For those Unit 2 harsh environment safe shutdown TOLs with qualification deficiencies, TVA will issue a design to disable the TOLs by disconnecting the control circuit interlocks until qualified TOLs are obtained.

This inspection focused on the corrective actions that were being implemented by TVA to resolve the thermal overload concern for Unit 1 Restart. The inspection was conducted by reviewing work order records, design basis documents, and conducting walkdown inspections of as-built thermal overload installations.

b. Observations and Findings

The inspectors reviewed the following TOL replacement WOs:

- 03-021841-003, Completed 5/25/05
- 03-021841-057, Completed 7/13/05
- 04-720414-046, Completed 7/13/05
- 03-021841-020, Completed 4/26/05
- 03-021841-074, Completed 3/18/05
- 03-021841-021, Completed 2/03/05
- 03-021841-094, Completed 4/29/05

The referenced WOs replaced or strapped out (i.e., jumpered) the TOL heater elements in a total of 15 breaker cubicles on the 1A and 1B 480-VAC Reactor MOV Boards. The inspectors selected the referenced WO packages to assess the adequacy of the licensee's TOL replacement program. The inspectors conducted walkdown inspections of each breaker cubicle to verify that the correct TOLs had been installed. The inspectors found configuration control problems in two out of the 15 breaker cubicles inspected. The inspectors found that the installed TOLs in Cubicles 14C-2 and 15C on 480-Volt Reactor MOV Board 1B did not match the as-constructed drawing configuration. Wiring Diagram 1-45N1750-5 specified that TOLs for valves 1-FCV-23-46 and 1-FCV-23-52 were to be strapped out per DCAs 51090-134 and 51090-137, respectively. Contrary to this requirement, during the walkdown inspections of the cubicles, it was discovered that the TOLs had model C695A heater elements installed instead of being strapped out.

The original design change to replace the TOL heaters in cubicles 14C-2 and 15C was issued in DCN 51090, Stage 24, on December 29, 2003. Post Issuance Change (PIC) No. 62058 to DCN 51090 was subsequently issued on October 27, 2004, which added design guidance for strapping out TOLs in cubicles 14C-2 and 15C. WO 03-021841-57 was implemented on July 29, 2005, which installed model C695A heater elements in cubicles 14C-2 and 15C contrary to the requirements of PIC 62058 for strapping out the TOLs. The licensee later closed DCN 51090, Stage 24, on August 3, 2005, on the basis that the WO and PIC were completed.

10 CFR 50, Appendix B, Criterion III, Design Control, requires that measures be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. Contrary to the above, on August 3, 2005, measures were not adequate to assure that the design changes (PIC 62058) for strapping out the TOLs in 480-V Reactor MOV Board Cubicles 14C-2 and 15C were correctly translated into work instructions. As a result of these

inadequate measures, DCN 51090, Stage 24, and PIC 62058 were closed and the as-constructed drawings were issued, without the TOLs being strapped out in the plant. A Severity Level IV Violation was identified against the criteria in Supplement II, Facility Construction, of the NRC Enforcement Policy. This violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy, NCV 50-259/2005-08-02, Measures Were Not Adequate to Assure that the TOLs in 480-V MOV Board 1B Cubicles 14C-2 and 15C Were Strapped Out. This issue was documented by the licensee in PER 89577. The licensee indicated that a special team will be formed to investigate the root cause and extent of condition for this design control problem.

c. Conclusions

A Severity Level IV NCV was identified for failure to comply with 10 CFR 50, Appendix B, Criterion III, in that, measures were not adequate to assure that the design change output documents for strapping out the TOLs in 480-V Reactor MOV Board 1B Cubicles 14C-2 and 15C were correctly translated into work instructions. This resulted in the as-built TOL configuration of the plant deviating from design drawings. Based on the above finding, additional inspection will be necessary to verify that the TOL Special Program was adequately implemented.

E1.10 Special Program Activities - Small-Bore Piping and Instrument Tubing (37551)

a. Inspection Scope

The small-bore piping (less than 2-1/2 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 a 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 Safety Evaluation Reports dated October 24, 1989, and January 23, 1991.

The inspectors reviewed walkdown procedures and design criteria, reviewed design calculations and DCNs, walked down selected small bore piping and instrument tubing systems, and examined completed modifications.

b. Observations and Findings

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 Feedwater System, Core Spray System, High Pressure Coolant Injection (HPCI) System, and Reactor Core Isolation Cooling (RCIC) system to verify that the system walkdowns were completed in accordance with the licensee's walkdown procedures and deficiencies were identified and documented. The deficiencies identified by the licensee included excessive span lengths between supports, missing hardware on supports, overloaded supports, and inadequately constructed supports. Another type of deficiency identified by licensee engineers during the walkdowns were supports located too close to anchor points. The inspectors reviewed calculations which evaluated the deficiencies and the design output documents such as DCNs which specified the required field work to correct the deficiencies. The inspectors walked down portions of the feedwater and core spray systems 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 that the correct type of hardware was installed as specified in the DCN and that the supports (those installed too close to anchor points) were removed.

c. Conclusions

During the walkdown inspection, the inspectors verified that 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 concluded 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. However, additional samples will be inspected prior to the NRC reaching a conclusion on closure of this Special Program.

No findings of significance were identified.

E1.11 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 during the Unit 2 recovery, the licensee identified an issue regarding attachment of control rod drive hydraulic (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. TVA implemented modifications to the Unit 2 CRDH support frames prior to Unit 2 restart. A walkdown inspection of the Unit 3 CRDH piping and CRDH 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 and replace them with new structural steel frames.

b. Observations and Findings

During the current inspection, the inspector examined one of the new Unit 1 CRDH support frames, number 105. The new frame was inspected against the design drawings for configuration, member size, weld size, type and length connection details, method of attachment of the CRDH piping to the new support frame, and other construction requirements stipulated by the licensee's procedures.

c. Conclusions

No discrepancies were identified during the walkdown inspection. The inspectors concluded that the modification to the CRDH support frame was implemented in accordance with design requirements. However, additional samples will need to be inspected prior to closure of this Special Program. No findings of significance were identified.

E1.12 Special Program Activities - Configuration Management/Design Baseline (37551)

a. Inspection Scope

The inspectors reviewed the design basis of two safety-related systems and completed design calculations and evaluated the plant configuration to verify that it satisfied the design basis. This review included lessons learned from Units 2 and 3 as described in a TVA letter to NRC, dated June 13, 1991, Design Baseline Verification Program. This also included a review of the commitment described in a NRC letter to TVA, dated November 21, 1991, Assessment of Browns Ferry Nuclear Plant, Units 1 and 3 Design

Baseline Verification Program. The HPCI and RCIC systems were selected for this review.

a. Observations and Findings

TVA's program to re-establish the design basis for Unit 1 and evaluate plant configuration to ensure that it satisfied the design basis was performed in a manner essentially the same as that previously used during the Unit 3 recovery. In addition, the licensee's program included a review of essential engineering calculations for Unit 1. The inspectors observed the as-installed configuration of the HPCI and RCIC systems. The inspectors also reviewed electrical, mechanical, and instrumentation and controls calculations associated with the HPCI and RCIC systems. A list of the calculations reviewed is included in the attachment.

Prior to the arrival of the NRC inspection team, TVA had performed a focused self-assessment for these systems. As a result of this self-assessment, TVA identified various minor deficiencies which resulted in the revision of 13 calculations and seven PERs. These revisions had minimal impact on the conclusions reached by the calculations and were generally associated with a lack of attention to detail. The PERs issued by TVA will evaluate the extent of condition associated with these revisions to the calculations. TVA's self-assessment covered 10 of the 11 items which the NRC identified during the in-office review of the HPCI and RCIC calculations. The single remaining item identified by the NRC that was not identified during the self-assessment was associated with the HPCI and RCIC pump injection phase total developed head (TDH). The NRC inspectors found that MDQ0-999-2004-0040, HPCI and RCIC System Test Requirements, did not have an analytical or technical basis for the TDH acceptance criteria. The inspectors and licensee engineers re-calculated the TDH and the calculation was revised. This oversight by the licensee had minimal impact on the calculation's result and the licensee issued a PER to determine extent of condition. The inspection also resulted in one additional calculation revision and four additional PERs.

c. Conclusions

Based on observations, document reviews, and discussions with engineering personnel, the inspectors determined that TVA had developed an acceptable program to maintain configuration management and control of essential calculations for Unit 1. In addition, TVA demonstrated that the self-assessment program was aggressive and effective.

No violations or deviations were identified during this review of the licensee's configuration management and control of essential calculations for Unit 1. Based on this inspection, the inspectors concluded that implementation of the Configuration Management/Design Baseline Special Program was adequate and no additional inspections in this area are anticipated.

E1.13 Special Program Activities - Seismic Class II Over Class I/Spatial System Interactions and Water Spray (37551)

a. Inspection Scope

The inspectors reviewed the Browns Ferry Unit 1 Seismic Class II over Class I/Spatial System Interactions and Water Spray activities as detailed below to ensure that these activities were in compliance with regulatory requirements and licensee commitments.

b. Observations and Findings

This issue involved the effects of Spatial System Interactions and water spray from potential failure of non-safety related piping on safety-related equipment. The inspectors verified that the Unit 1 Special Program implemented to evaluate this issue was similar to the one implemented for Units 2 and 3. The inspectors reviewed the report issued by Facility Risk Consultants, Inc., "Seismic-Induced II/I Spray Evaluations at Browns Ferry Unit 1, March 2004," to determine the scope and adequacy of the required actions for this area of concern. The inspectors found that the key engineering attributes of the seismic II/I evaluation program consisted of in-plant screening walkdown evaluations and identification of potential outliers; further evaluations and resolution of potential outliers; engineering design of plant modifications to resolve outliers; and WOs to address general maintenance and housekeeping items. The inspectors verified that the in-plant screening walkdown evaluations of seismic II/I spray hazards were performed on an area-by-area basis in accordance with Walkdown Instruction WI-BFN-0-CEB-06, Engineering Walkdown Instruction for Evaluation of Seismic-Induced Spray Hazards, which was also reviewed to verify that it could accomplish the actions required. A total of 27 designated plant areas were included. Screening evaluations focused on certain key attributes of the non-seismic Class I (Class II) piping and fluid pressure boundary systems that may potentially pose as spray hazards to surrounding seismic Class I systems and components in the event of an earthquake.

Walkdown results, including a total of 179 potential identified outliers, are documented in the Walkdown Data Packages (WDPs) for the respective plant areas. Potential outliers identified during the in-plant screening walkdowns were further evaluated to the acceptance criteria of TVA Design Criteria BFN-50-C-7306, Qualification Criteria for Seismic Class II Piping, Pipe Supports, and Components. The inspectors verified that the evaluations and bounding analyses of these potential outliers were of hand calculations using basic engineering mechanics techniques for simple configurations, and rigorous piping analyses (TPIPE computer program) for more complex piping configurations. A total of 19 outliers were found to have not met the acceptance criteria. The inspectors reviewed the plant modifications as they were designed and DCN 51669, U1 Recovery - Reactor Building: Seismic II/I Water Spray, which was issued to implement the changes so that all of these concerns were resolved. The inspectors performed walkdowns of a sufficient portion of the 19 outliers in various plant areas to verify that the scope of the issues identified in each outlier was accurate and would resolve the issues. Furthermore, 13 maintenance and/or housekeeping items were also identified for corrective action and maintenance work order requests were issued to address these items. The inspectors verified that these maintenance items were either completed or being tracked properly for future completion.

c. Conclusions

Based on field walkdowns and observations, document reviews, and discussions by the inspectors with engineering personnel, the inspectors determined that TVA has developed an aggressive program for Seismic Class II over Class I/Spatial System Interactions and Water Spray Program on Unit 1. Licensee actions to address issues for the Seismic Class II over Class I/Spatial System Interactions and Water Spray Special Program 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 committed to and performed for Units 2 and 3. No issues related to the Seismic Class II over Class I/Spatial System Interactions and Water Spray Special Program that would negatively impact the restart of Unit 1 were identified as the result of the above review. Based on this inspection, 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 Seismic Class II over Class I/Spatial System Interactions and Water Spray Special Program for Unit 1.

E1.14 Special Program Activities - Cable Tray Supports and Conduit Supports (37551)

a. Inspection Scope

The inspectors reviewed the Browns Ferry Unit 1 Cable Tray and Conduit Supports activities as detailed below to ensure that these activities were in compliance with regulatory requirements and licensee commitments. See NRC Inspection Report 50-259/2004-006 for previous inspections in this area.

b. Observations and Findings

b.1 New Cable Tray and Conduit Supports

New Unit 1 supports are those installed since the restart of Units 2 & 3. These supports are designed in accordance with the licensee's seismic design criteria and installed under the licensee's quality assurance program requirements.

The inspectors performed walkdown inspections and examined new cable trays and conduit supports. New supports were inspected against the design drawings for configuration, member size, weld size, type and length, connection details, and other construction requirements. Additional acceptance criteria utilized by the inspectors during the walkdown inspections included WI-BFN-0-GEN-01 Walkdown Instructions, Revision 1; BFN-50-C-7104, Design of Structural Supports, Revision 12; and VE-2-2001 NEMA Standards Publication, 2001.

The inspectors reviewed the following calculations for new cable tray and conduit supports for completeness, accuracy and adherence to design criteria and procedural requirements: CDQ1 362 2004 0212, CDQ1 361 2004 0280, and CDQ1 361 2005 0141. The modifications were implemented with the following DCNs: 51227 and 51223. No deficiencies were identified.

b.2 Evaluation of Existing Cable Tray and Conduit Supports

Seismic verification of existing cable tray and conduit supports is being accomplished using the Generic Implementing Procedure (GIP) for Seismic Verification of Nuclear plant Equipment. The GIP was issued by the Seismic Qualification Utility Group (SQUG) in response to NRC Unresolved Safety Issue A-46 (USI A-46), Seismic Adequacy of Mechanical and Electrical Equipment in Operating Plants. The licensee committed to complete the A-46 walkdown for cable tray and conduit supports in Unit 1 prior to restart of Unit 1. The inspectors reviewed WI-BFN-0-CEB-04, Seismic Verification Walkdown Instruction for USI A-46 and Seismic IPEEE, Revision 0. This procedure specifies the instructions for implementation of the GIP requirements for personnel qualifications, precautions, methodology, acceptance criteria, and documentation requirements.

The licensee has completed the A-46 walkdown inspections for existing cable tray and conduit supports in all Unit 1 Category I structures. During the A-46 walkdowns, the licensee evaluated cable tray fill, spans, and supports, including anchorage and conduit spans, supports and anchorage using the criteria in the GIP. Cable trays, conduits, and supports which did not meet the GIP acceptance criteria were designated as outliers.

The inspectors reviewed the results of the licensee's A-46 walkdowns. Problems (outliers) were documented on Outlier Seismic Verification Sheets. The outliers were addressed either through a plant work request, or by a design evaluation documented in a calculation using the GIP acceptance and the licensee's design criteria. For this program area, 14 outliers were identified during the TVA walkdowns and, as of this inspection, six outliers have been resolved with a total of three modification packages being developed and implemented for the resolution of these completed issues.

Items addressed by work requests included missing or damaged hardware covered by existing plant maintenance procedures. Outliers which involved questionable design and/or construction practices, e.g., conduit over-spans, apparent inadequate anchorages, potential seismic interactions, supports which do not meet current design practices, etc. were evaluated by the licensee in calculation number CDQ1 000 2003 2203, USI A-46 Seismic Verification of Cable Tray and Conduit Raceway Systems, and DCN 51521, U1 Recovery Reactor Building Structural Modification Required by A-46 Evaluation (Structural components required by the A-46 qualifications of cable tray and conduit supports.) The inspectors reviewed the calculation for completeness, accuracy, and adherence to design criteria and procedural requirements. No deficiencies were identified.

The inspectors performed walkdowns of various modifications implemented to resolve A-46 outliers, and verified that the modifications were implemented in accordance with the DCN. Following various walkdowns, any deficiencies noted by the inspectors, were subsequently addressed by the licensee and addressed in the licensee corrective action program as PERs to document and 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 supports to perform their intended function.

c. Conclusions

Based on observations, document reviews, and discussions with engineering personnel, the inspectors determined that TVA has developed an aggressive program for cable tray and conduit supports on Unit 1. Licensee actions to address issues for cable tray and conduit supports 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 committed to and performed for Units 2 and 3. No issues related to the Cable Tray Supports and Conduit Supports Special Programs 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 these Special Programs.

No violations or deviations were identified during this review of the licensee's Cable Tray Supports and Conduit Supports Special Programs for Unit 1.

E1.15 Special Program Activities - Drywell Steel Platforms and Upper Drywell Platforms (37551)

a. Inspection Scope

During investigations performed in the 1980s by the licensee and the NRC related to restart of Unit 2, numerous deficiencies were identified in the design and construction of safety-related structural steel platforms. These included cracking of clip angles which connect structural members, 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 the addition of more loads to the platforms).

The licensee's commitments for resolution of issues associated with the drywell 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 the 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 drywell structural steel platforms were submitted to the 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.

b. Observation and Findings

The Unit 1 drywell structural steel platforms have been re-designed to meet current design criteria. Design changes were issued to modify the structural steel platforms to correct the deficiencies. The majority of the original structural steel members for the platforms on elevations 563 and 584 were removed and replaced. The connections on the platforms on elevations 604, 616, and 628 were modified by replacing bolts, adding clip angles and stiffeners, and reinforcing existing welds. The modified structural platforms are intended to meet current design criteria and have a design margin for the addition of future loads, if necessary.

The modifications for the drywell structural steel platform were examined by Region II inspectors during inspections documented in NRC inspection reports 50-259/2003-009, 50-259/2003-011, 50-259/2004-006, and 50-259/2004-007. The inspections included review of design calculations, design drawings, work control instructions, quality control inspection procedures, procurement and receipt inspection of new structural steel members for modification of the structural steel platforms, examination of completed modifications, and review of quality assurance records. During the walkdown inspections, the inspectors verified that 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 verification of correct type bolts in existing connections. The inspectors also reviewed quality assurance oversight of the drywell structural steel design and construction program.

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 structural steel platforms complied with commitments to the NRC. Based on this inspection and previously documented NRC inspections, the inspectors concluded that no further inspections are anticipated for this Special Program. No findings of significance were identified.

E1.16 Intergranular Stress Corrosion Cracking (IGSCC) Special Program (37551)

a. Inspection Scope

As discussed in Section 3.6 of NUREG 1232, Safety Evaluation Report on the Browns Ferry Nuclear Performance Plan (BFNPP), and in Section III.7.0 of the BFNPP, intergranular stress corrosion cracking (IGSCC) was identified in a number of stainless steel piping systems and reactor vessel (RV) safe ends during nondestructive examination (NDE) of these systems in response to NRC Generic Letter 88-01.

As stated in NUREG-1232, Volume 3, the NRC staff concluded that the IGSCC program defined in Section III.7.0 of the BFNPP was acceptable. Specific commitments that Browns Ferry has made with respect to IGSCC mitigation actions for all three units were summarized in Table III-7, IGSCC Mitigation Actions, of the BFNPP. These commitments include the implementation of a Hydrogen Water Chemistry (HWC) control program, removal of head spray piping, replacement of piping that is susceptible to IGSCC, stress improvement to reduce residual weld stresses, and application of leak detection to potentially inaccessible welds if necessary. In the BFNPP, the licensee committed to use piping materials that are more resistant to IGSCC in accordance with guidance stated in NUREG 0313, Revision 2, for replacement of the Reactor Recirculation System piping. Inspectors have observed and/or reviewed these activities as stated in this inspection report and the following inspection reports: 50-259/2003-009, 50-259/2003-010, 50-259/2003-011, 50-259/2004-006, 50-259/2004-007, 50-259/2004-009, and 50-259/2005-006.

As detailed in TVA Browns Ferry Unit 1 Regulatory Framework Letters December 13, 2002 and February 28, 2003, and Letter of Response to Request for Supplemental Information on the Regulatory Framework for the Restart of Unit 1, dated June 11, 2003, the applicable Codes for the recirculation piping replacements are: 1) ASME Section XI, 1995 Edition, 1996 Addenda, and 2) ASME Section III, Class 1, 1995 Edition, 1996 Addenda.

b. Observations and Findings

The inspectors observed Mechanical Stress Improvement Process (MSIP) activities for a 24-inch RHR System pipe weld, 1-012-026. The inspectors reviewed the applicable procedures and qualification records for equipment and personnel involved, such as: pressure gauge calibration, MSIP performance and verification records, MSIP calculation sheets, stud and clamp certifications, and the MSIP weld traveler. In addition, the inspectors reviewed MSIP documentation packages for the following welds:

- RWR 1-002-042, Reactor Water Recirculation System, ASME Class 1
- RWR 1-001-050, Reactor Water Recirculation System, ASME Class 1
- CS 1-002-022, Core Spray System, ASME Class 2

The inspectors observed in-vessel remote ultrasonic (UT) examinations of Unit 1 core shroud welds H6 and H4. The inspectors held discussions with UT Level III analysts regarding acceptance criteria of indications identified in core shroud welds. Inspectors also reviewed the work orders associated with the in-vessel examinations of the Reactor Pressure Vessel (RPV), and the replacement of the 48 RPV shroud bolts in Unit 1 with newly designed bolts. In addition, the inspectors reviewed DCN 51193, Unit 1 Reactor Building Mechanical System 68. This DCN addresses RPV components, RPV head leakage problems, and IGSCC in RPV access hole covers and shroud head bolts.

As stated in previous inspection reports, inspectors reviewed and observed licensee activities related to the pipe replacement of reactor recirculation inlet and outlet safe ends, reactor recirculation piping, core spray and RHR piping inside the containment, reactor water cleanup inside and outside of containment (includes penetration piping), and jet pump instrumentation nozzle safe ends and seal assemblies. These components were replaced with materials that are less susceptible to IGSCC. The replacement of piping susceptible to IGSCC is more extensive in Unit 1 than the piping replacement efforts for Units 2 or 3.

As stated in NRC Inspection Report 50-259/2003-011, the licensee committed to implement a HWC control program to control the chemical environment to which the replacement pipe is exposed. Browns Ferry Unit 1 intends to adopt a HWC control program similar to that employed by Browns Ferry Units 2 and 3. The inspectors verified that the licensee was following appropriate HWC guidance given in NUREG 0313, Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping, Rev. 2, June 1986.

c. Conclusions

Based on observations, document reviews, and discussions with engineering personnel, the inspectors determined that TVA has developed a thorough program to mitigate the long term effects of IGSCC on Unit 1. Licensee actions to address issues in Generic Letter (GL) 88-01, NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping, and GL 94-03, IGSCC of Core Shrouds in BWRs, have been performed or are being performed by the licensee. The NRC had previously issued a Safety Evaluation Report (SER) on the licensee's response to GL 88-01 for Unit 3 on December 3, 1993. The licensee's response to GL 94-03 was determined to be satisfactory and the NRC issued a SER on January 13, 1995. Completed or planned actions to address these issues for Unit 1 are consistent with those previously performed for Units 2 and 3. In addition, the inspectors' review of TVA's actions for long-term mitigation of IGSCC of Unit 1 reactor vessel internals determined that they were satisfactory. No issues related to IGSCC that would negatively impact the restart of Unit 1 were identified as the result of the referenced review. Based on this and previously documented NRC inspections, no further inspections are anticipated for this Special Program.

No violations or deviations were identified during this review of the licensee's Intergranular Stress Corrosion Cracking Special Program for Unit 1.

E1.17 Inservice Inspection/Preservice Inspection (ISI/PSI) (55050, 71111.08)

a. Inspection Scope

The inspectors reviewed the Browns Ferry Unit 1 ISI/PSI activities as detailed below to ensure that these activities were in compliance with regulatory requirements and licensee commitments. See NRC Inspection Reports 05000259/2004007, 05000259/2004009, and 05000259/2005006 for previous inspections in this area.

As detailed in the licensee's ISI program, the first ten-year ISI interval, which began August 1, 1974, for Browns Ferry Unit 1 is currently in its third period and will end one year following the restart of the unit. The applicable Codes for the ISI/PSI programs are ASME Section XI, 1995 Edition, 1996 Addenda, and for sample selection, ASME Section XI, 1974 Edition, with Addenda through Summer 1975.

b. Observation/Review of ISI/PSI Activities

For PSI activities, inspectors observed UT examinations of the following welds:

- RHR 1-012-026, UT PSI, Class 1
- RWR 1-002-042, Reactor Water Recirculation System, UT PSI, Class 1
- RWR 1-001-042, Reactor Water Recirculation System, UT PSI, Class 1

The inspectors reviewed the PSI or ISI examination reports for the welds listed below:

- RWR 1-001-044, Reactor Water Recirculation System, UT PSI, Class 1
- RWR 1-002-040, Reactor Water Recirculation System, UT PSI, Class 1
- RWR 1-001-002, Reactor Water Recirculation System, UT PSI, Class 1
- RWR 1-001-024, Reactor Water Recirculation System, UT PSI, Class 1
- CS 1-002-019, Core Spray System, UT PSI, Class 2
- CS 1-002-022, Core Spray System, UT PSI
- GFW 1-14, MT PSI, Feed Water System, Class 1
- KFW 1-28, MT PSI, Feed Water System, Class 1
- KMS 1-13, MT PSI, Main Steam System, Class 1
- SHPCI 1-3, MT ISI, High Pressure Coolant Injection System, Class 2
- DSCS 1-14, PT ISI, Core Spray System, Class 2

For the referenced examinations, inspectors reviewed the examination data sheets, equipment calibration records, examination procedures, and examination personnel certifications. The PSI/ISI inspection activities and records were compared to the applicable requirements.

At the time of the inspection, ISI and PSI examination activities were continuing with the licensee approximately 90% complete for the Class 1 and Class 2 piping welds and integral attachments required to complete the first 10-year interval. The inspectors held discussions with licensee personnel regarding status and results of examinations and disposition of recordable indications.

c. Conclusions

The inspectors determined that the licensee's (ISI/PSI) activities observed/reviewed met applicable code requirements and licensing commitments. No violations or deviations were identified.

E1.18 Power Service Shop Activities (IPs 71111.17 and 37550)

a. Inspection Scope

The inspectors visited the licensee's electrical repair facilities at the Power Service Shop (PSS) in Muscle Shoals, Alabama, and observed acceptance testing in progress on the 1A RHR motor.

b. Observations and Findings

The inspectors observed ongoing testing activities performed on the 1A RHR motor. The licensee has been performing electrical testing and inspection of large Unit 1 motors which have been refurbished at that location. The inspector verified adequacy of housekeeping, PSS work control processes, PSS test procedures, and that work documents incorporated technical and quality requirements included in the contract. Testing was performed in accordance with PSS Specification J1RA-Gen.3.50, Electrical Motor Testing. The inspectors noted that the assigned technicians, electrical shop foreman, and PSS engineer were knowledgeable of technical and quality assurance requirements. The inspectors reviewed PSS Job Order QQ243 which documented the refurbishment of the RHR motor. The inspectors noted that the test package included specific hold points for the TVA source surveillance contractor. The source surveillance contractor and one member of the Browns Ferry Unit 1 QA organization were present during the testing at the PSS. The inspectors reviewed test results for competed winding resistance checks, insulation resistance checks, surge comparison tests, DC step voltage tests, and high voltage (leakage) tests. In addition, the inspectors observed satisfactory motor vibration and bearing temperature rise testing. This testing was performed for no-load conditions on a test stand at the PSS. The inspectors noted that the motor was operated for two hours, which satisfied the J1RA-Gen.3.50 test requirement to operate the motor for one hour and until stable bearing temperature was reached (1 degree Celsius or less change in 10 minutes). The inspectors also noted that the motor vibration readings remained well below the upper limit of 0.157 inches per second for the duration of the testing. The inspectors concluded that the testing satisfied acceptance criteria and other requirements specified in PSS Specification J1RA-Gen.3.50.

c. Conclusions

Unit 1 Recovery activities performed at the PSS machine shop continued to satisfy regulatory requirements. Ongoing work involved a high level of professionalism. Work documents included applicable technical and quality requirements. No violations or deviations were identified.

E7 Quality Assurance in Engineering Activities (71152)

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

a. Inspection Scope

The inspectors observed licensee source surveillance and Nuclear Assurance (NA) oversight activities associated with testing of the 1A RHR motor at the PSS. 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

Prior to a trip to the PSS to observe ongoing electrical testing of the 1A RHR motor, the inspectors reviewed the licensee's inspection plan for the scheduled NA observation visit at the PSS. The inspector verified that the licensee's inspection plan included specific guidance for review of PSS test procedures and that work documents incorporated technical and quality requirements included in the contract with PSS. The inspectors noted that both source surveillance personnel and a member of the Unit 1 NA assessment group was present during the testing. Source surveillance inspections are provided for TVA by various contracts with outside independent organizations. The assessors were knowledgeable of the applicable work document requirements and PSS programs and work processes. The inspectors noted that work documents included specific hold points for TVA source surveillance personnel.

Although no deficiencies were identified during observation of the ongoing testing, the inspectors reviewed selected PSS Corrective Action Reports (CARs) to verify the adequacy of the PSS program for identification and resolution of problems. The inspectors reviewed a report which included all PSS CARs generated for Unit 1 recovery activities during the period starting 12 months prior to the date of the visit to the PSS. The inspectors noted that for any PSS CARs that involved potentially significant issues associated with Unit 1 components, a cross-referenced Unit 1 PER number was included in the PSS database.

c. Conclusions

The licensee's program for oversight of Unit 1 recovery activities performed at PSS was well planned and NA assessors were knowledgeable of the applicable work document requirements and PSS programs and work processes. No violations or deviations were identified.

E8 Miscellaneous Engineering Issues (92701)

E8.1 (Closed) GL 94-03, Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors

IGSCC of Boiling Water Reactor (BWR) internal components has been identified as a technical issue of concern by both the NRC staff and the industry. In order to verify compliance with the structural integrity requirements of 10 CFR 50.55a and to assure that the risk associated with core shroud cracking remains low, the NRC concluded that it is appropriate for BWR licensees to implement inspection and/or repairs, as appropriate, at their BWR facilities. Licensees were asked to inspect their core shrouds and perform an evaluation and/or repair based on the results of the inspection. In addition, they were asked to perform a safety analysis for the operating units supporting continued operation until inspections are performed.

The inspectors reviewed the Safety Evaluation of TVA's response to GL 94-03 for Units 2 and 3 to verify that the licensee is taking similar actions for Unit 1. The inspectors observed core shroud inspections and reviewed the preliminary results of examinations. Through the inspectors' review of TVA's actions for long term mitigation of IGSCC of Unit 1 reactor vessel internals, namely the core shroud, the inspectors determined that the licensee's actions were aggressive and satisfactory. The inspectors verified that the licensee is conducting inspections of the core shroud using the latest technology. The inspectors verified that the licensee has a program in place to follow guidance stated in the applicable BWR Vessel and Internals Project (BWRVIP) documents, and to work closely with the BWR Owners group with respect to addressing IGSCC of BWR internals. In addition, the inspectors reviewed related materials and plant-specific safety analysis documented in the licensee's response to GL 94-03. The inspectors concluded that the licensee's engineering staff can conduct thorough reviews of crack indications, and will take additional measures for any identified adverse crack indications. Therefore, because this item is effectively being tracked in the licensee's corrective action program, is being corrected similarly to Unit 2 and 3, 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.

E8.2 (Closed) GL 88-14, Instrument Air Supply System Problems Affecting Safety-Related Equipment

NUREG 1275, Volume 2, Operating Experience Feedback Report - Air Systems Problems, indicated that the performance of air-operated safety-related components may not be in accordance with their intended safety function because of inadequacies in the design, installation, and maintenance of the instrument air system. GL 88-14 requested licensees to review NUREG 1275, Volume 2, and perform a design and operations verification of the instrument air system. In addition, licensees were required to include in their response a discussion of their program for maintaining proper instrument air quality. Review of this item prior to the restart of Unit 3 was documented in NRC Inspection Reports 50-259,260,296/95-38 and 50-259,260,296/95-56 based on review of the licensee's program for Unit 3.

The inspectors reviewed licensee actions required by GL 88-14 to ensure adequacy of licensee actions and verified that TVA had completed or planned all aspects of GL 88-14. The inspectors also verified that TVA had completed commitments described in TVA letter to the NRC dated February 23, 1989, Browns Ferry Nuclear Plant, Sequoyah Nuclear Plant, and Watts Bar Nuclear Plant response to GL 88-14, Instrument Air Supply System Problems Affecting Safety-Related Equipment, TVA letter to the NRC dated July 30, 1993, Supplemental Response to GL 88-14, Instrument Air Supply System Problems Affecting Safety-Related Equipment, and NRC letter to the TVA dated May 9, 1989, GL 88-14, Instrument Air Supply System Problems Affecting Safety-Related Equipment (TAC Nos. 71631/71632/71633).

The inspectors verified the implementation of commitments made by TVA in their response to GL 88-14. After conducting reviews of testing and associated acceptance criteria, maintenance practices, emergency procedures, design documentation, and licensee corrective actions, the inspectors determined that TVA had developed an acceptable program to comply with commitments made in response to GL 88-14. Based on these reviews, the inspectors concluded that existing licensee commitments serve to ensure that instrument air systems at TVA Browns Ferry will perform their safety-related functions and that air-operated safety-related components will perform as expected in accordance with all design basis events, including a loss of instrument air. 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.

E8.3 (Closed) GL 96-06, Assurance of Equipment Operability and Containment Integrity During Design Basis Accident Conditions

Generic Letter (GL) 96-06, Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions, requests that all addressees submit certain information relative to safety-significant issues that could affect containment integrity and equipment operability during accident conditions. In the NRC's completion of licensing action letter dated February 15, 2000, GL 96-06 was considered closed for BFN Units 1, 2 and 3; however, for Unit 1, the GL stated that Unit 1 will be revisited in the event of a restart. NRC's closeout letter of GL 96-06 and safety evaluation for Unit 1, dated February 7, 2005, states two licensee commitment actions prior to Unit 1 restart: 1) modify Unit 1 drywell floor and equipment drain sump discharge lines to provide a designed method of over-pressure protection, and 2) revise plant procedures to ensure that water is partially drained from the portions of the demineralized water system inside the drywell and the system left open to the drywell during power operation. In addition, the inspectors reviewed the licensee's response to GL 96-06.

The first commitment consists of installing new check valves (CKV-77-600, -603, -625, and -628) with a 1/16" diameter hole drilled in the disc to prevent over-pressure. These valves are located downstream of the sump pumps. The inspectors reviewed DCN 51154, Revision A, and verified that the valves had been installed and oriented in the same fashion as the original valves per Drawing 67 M 1-47E482-3, R001, in the drywell, El 550'.

The second commitment consists of procedural changes to ensure that the demineralized water system cannot be over-pressurized. The inspectors verified that the procedures had been revised to require that the system is sufficiently drained following use and is open to containment during operation by opening a demineralized water service connection isolation valve at a low elevation in the drywell as was done in Units 2 and 3 through prior commitments, including Licensee Event Report (LER) 50-259/97-01. The inspectors reviewed 0-OI-2C, Attachment 1A, Demineralized Water System Valve Lineup Checklist - Unit 1 (dated April 28, 2005) and 1-GOI-200-2, Drywell Closeout, Revision 0000. The Drywell Closeout procedure for Unit 1 is consistent with Units 2 and 3; however, the Demineralized Water System Valve Lineup Checklist - Unit 1 is not consistent. Unit 2 and Unit 3 Demineralized Water System Valve Lineup Checklist (0-OI-2C, Attachments 1B and 1C) identify valve SHV-002-1199, demineralized water service connection which is located in the Drywell basement elevation 550', as OPEN. Currently, the licensee incorporates DCNs into its procedures as an ongoing process; however, the changes made to Unit 2 and Unit 3 procedures stemming from the GL had not been incorporated into the Demineralized Water System Valve Lineup Checklist for Unit 1. In addition, Units 2 and 3 checklists describe the draining of this valve for over-pressurization protection following a loss of coolant accident (LOCA) in a footnote considering GL 96-06. Demineralized Water System Valve Lineup Checklist 0-OI-2C, Att. 1A, for Unit 1 does not address this valve (which is represented in Drawing 67 M 1-47E856-2, R001, Flow Diagram Demineralized Water) or the reference to GL 96-06. The inspectors questioned these discrepancies and the licensee responded by stating that a thorough review will be performed to ensure that these items are addressed prior to closing the DCN as part of their normal work control process.

The inspectors also reviewed the licensee's use and application of computer codes for performing water-hammer and two-phase flow issues in response to GL 96-06. TVA has determined that the liquid in the containment cooler coils will not boil during a design-basis steam line break or loss of coolant accident (LOCA) based on the analysis performed using the Generation of Thermal-Hydraulic Information for Containments (GOTHIC) computer program. The inspectors held discussions with TVA personnel about the qualification and evaluation of the program codes used. The program is supported through an EPRI User's Group and controlled programmatically by procedure SPP-2.6, Computer Software Control, Revision 10. GOTHIC is category B Quality Assurance (QA) level software which can be used in calculations that support safety-related design basis of the plant.

The inspectors determined that no further NRC action was required in this area; 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, 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.

E8.4 (Closed) Unresolved Item (URI) 259/87-26-03, RHR Pump Suction and Nozzle Load Allowables Are Exceeded

This item concerned allowable stresses in the RHR nozzles. This issue was originally identified by the licensee in Deficiency Number 87-13-6 of Engineering Assurance Audit 87-13. The licensee revised Calculation Number CDQ1-073-2003-0248 (System N1-173-5R) and generated new Calculation Number CD-Q3074-910631 (System N1-374-5R) and CD-Q-3074-910400 (System N1-374-7R) to evaluate the RHR pump suction anchor and nozzle loads. The revised and new calculations qualified the applied loads based on revised Design Criteria BFN-50-C-7103, General Design Criteria for Structural Analysis and Qualification of Mechanical and Electrical Systems (Piping and Instrument Tubing). The applied loads include I.E. Bulletin 79-14 requirements. The licensee also completed walkdown inspections to obtain as-built information and data which were used in the updated calculations.

The inspectors reviewed the following calculations which qualified the nozzle loads:

- Calculation Number CDQ-1303-2003-0672, Revision 2, dated 11/15/04, Qualification of Anchor Frame for RHR Pump 1A Suction
- Calculation Number CDQ-1303-2003-0662, Revision 1, dated 11/15/04, Qualification of Anchor Frame for RHR Pump 1B Suction
- Calculation Number CDQ-1303-2003-0663, Revision 2, dated 11/8/04, Qualification of Anchor Frame for RHR Pump 1C Suction
- Calculation Number CDQ-1303-2003-0664, Revision 1, dated 11/9/04, Qualification of Anchor Frame for RHR Pump 1D Suction

Based on this review, the inspectors determined that the calculations complied with the licensee's design criteria and were acceptable. The piping stresses are within code allowable values and no modifications were required. Therefore, no violation of NRC requirements occurred. This issue is closed.

E8.5 (Closed) TMI Action Item II.K.3.27, Common Reference Level for BWRs

The inspectors reviewed TMI Action Item II.K.3.27, Common Reference Level for BWRs. TVA letter dated December 3, 1982, provided the licensee's response to this item and stated that the licensee intended to resolve this item in conjunction with the CRDR program as required by TMI Action Item I.D.1. Closure of this item prior to restart of Unit 2 was documented in NRC Inspection Report 50-259,260,296/91-10, based on review of the implemented modifications. In addition, closure of this item prior to restart of Unit 3 was documented in NRC Inspection Report 50-259,260,296/95-16. The inspectors reviewed DCNs 51076 and 51094 which provided the details associated with planned upgrades to the Unit 1 Reactor Pressure Vessel (RPV) level instruments. The inspectors noted that RPV level instruments (1-LI-3-52, 1-LI-3-62A, 1-LR-3-62, 1-LIS-3-52, and 1-LIS-3-62A) would be re-scaled to match the existing instrument zero of 528 inches above vessel zero in accordance with Human Engineering Deficiency (HED) 283, in the licensee's program. 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.

III. Maintenance

M1 Conduct of Maintenance

M1.1 Maintenance Program

b. Inspection Scope

The inspectors continued to observe and/or review the licensee's activities involved with the maintenance program. Maintenance activities were observed to verify that work was controlled by approved plant procedures and WOs.

b. Observations and Findings

System return to service (SRTS) activities for System 23, RHRSW, and System 67, Emergency Equipment Cooling Water (EECW), represented a significant portion of ongoing maintenance activities during this report period. System 67 activities included WO 02-013229-00 for the actuator on valve 1-FCV-67-50, EECW north header flow control valve; WO 04-712868-00, 1-SHV-67-576 south header manual shutoff valve; WO 04-712979-00, 1-CHV-67- 577 south header check valve; WO 04-712980-00, 1-CHV-67- 642 north header check valve; and WO 04-715132-00, 1-FCV-67-50 north header flow control valve. The valves were located in the north to south cross-connect header which provided a cooling water supply to the RBCCW heat exchangers 1A, 1B, and the plant spare heat exchanger 1C. An additional cooling water supply to the Unit 1 RBCCW heat exchangers is from the RCW, System 24. System 23 activities included WOs 03-019416-44, replace small bore manual shutoff valves; 02-000969-02, perform work on 1C RHR heat exchanger; WO 05-715149-00, set travel limits on flow control valve 1-FCV-23-40, RHRSW discharge from 1C RHR heat exchanger; WO

05-719536-00, remove blind flange from RHRSW supply piping to the 1C RHR heat exchanger; WO 05-715150-00, set travel limits on flow control valve 1-FCV-23-34, RHRSW discharge from 1A RHR heat exchanger; WO 05-715151-00, check for proper rotation and stroke flow control valve 1-FCV-23-46, RHRSW discharge from 1B RHR heat exchanger; and WO 05-715147-00, check for proper rotation and stroke flow control valve 1-FCV-23-52, RHRSW discharge from 1D RHR heat exchanger.

Other significant maintenance work activities observed and reviewed included installation of the 1A and 1C RHR pump motors; refurbishment of the Hydraulic Control Units (HCU) for the Control Rod Drive; installation of the Condenser Circulating Water (CCW) Pumps 1A , 1B, and 1C; and preparations for installation of the 1A, 1B, and 1C Condensate Booster Pumps.

The inspectors reviewed the applicable WO packages and observed selected portions of the ongoing maintenance activities for the above maintenance activities. 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

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.

M1.2 System Cleanliness and Flushing Activities (37551)

a. Inspection Scope

The inspectors continued to monitor the effectiveness of the licensee’s Cleanliness Verification Program (CVP) to evaluate the adequacy of this program to support the Unit 1 Restart. Specifically, the inspectors reviewed licensee’s actions associated with satisfying 1-TI-474, Cleanliness Verification Program, Class B cleanliness and system flushing requirements for the RWCU System. A significant portion of RWCU piping and other components had been replaced and the system had recently completed SPOC I activities. In addition, the inspectors reviewed licensee actions to prevent introduction of foreign material into the RPV, reactor cavity, and fuel storage pool during the ongoing RPV internals exams. The inspectors reviewed the licensee’s foreign material exclusion (FME) program to determine if existing measures to identify and retrieve new or historical items were adequate. The inspectors also reviewed corrective action documents issued by the licensee to address FME issues identified during these activities.

b. Observations and Findings

b.1 System Cleanliness Program Activities

The inspectors reviewed selected SRTS activities associated with the RWCU System to determine the adequacy of system flushing activities. The inspectors noted that Procedure 1-TI-474 requires that the RWCU system meet Class B requirements, which require a high level of internal cleanliness and includes systems which have a direct fluid contact with the reactor core and fuel. The inspectors reviewed the licensee's flushing plan and WO 05-717682-000, which controlled system flushing activities following RPV fill. The inspectors held discussions with engineering personnel to determine the extent of flushing for this system. In addition, the inspectors reviewed PERs 51772 and 86609, which had been issued to address problems associated with purge dam material present in systems after dams were left in place through reactor vessel fill. The inspectors noted that most purge dams were removed from systems prior to RPV fill. However, some dams needed to be left in place for radiological reasons and were then flushed out following RPV fill. Systems which had some dams left in included RWCU, Recirc, and Core Spray. The remaining purge dams were removed by specific step text included in WO 05-717682-000 which flushed out the purge dams during initial operation of the RWCU system with the 1B RWCU Pump and installed suction strainer. The 40-micron suction strainer was clogged as anticipated and the strainer was removed and cleaned. After several iterations the purge dam material was removed. The inspectors determined that the licensee's actions to flush the RWCU System were adequate to support removal of remaining purge dam material and to ensure that the system satisfied documented cleanliness requirements.

b.2 Foreign Material Exclusion In Reactor Vessel and Fuel Storage Pool

The inspectors reviewed the licensee's program for FME during ongoing inspection and modification activities in the RPV, reactor cavity, and fuel storage pool. Specifically, the inspectors reviewed the licensee's FME program and observed ongoing activities to verify that adequate measures were taken to prevent loss of foreign objects in these areas during the ongoing examination of RPV internals. RPV internals activities continued throughout the inspection period with the RPV head removed and the reactor cavity gates removed. A significant amount of work occurred in or over the reactor vessel, reactor cavity and refuel storage pool. Ongoing activities included visual and UT exams of RPV internals; removal and replacement of incore instruments, removal and replacement of control rod blades; RVP vertical weld UT exams. The inspectors met with the Unit 1 FME Coordinator, System Engineering Manager, and General Electric (GE) personnel involved with the RPV internals exams. In addition, the inspectors observed ongoing activities on the refueling floor. The inspectors verified that the licensee's program included adequate measures to prevent dropping items into the RPV, reactor cavity, and refueling storage pool. In addition, the inspectors verified that the licensee performed dedicated inspections for FME and that reasonable effort was conducted to retrieve any new or historical items identified during the inspections. The inspectors noted that the licensee maintained a log of known items and that an engineering evaluation would be required for any known foreign item that would be left in the RPV. The inspector reviewed selected PERs which identified specific examples of foreign items or FME issues identified during ongoing activities. A list of the PERs reviewed is included in the attachment. The inspectors concluded that the licensee's program for FME, while working in the RPV, reactor cavity, or fuel storage pool, satisfied regulatory requirements and licensee commitments. GE personnel were in the process of conducting final FME inspections on RPV internals and vacuuming of internals was in progress at the end of the inspection period. At the time of this inspection there were no recently dropped foreign items which had been identified in the RPV which had not been retrieved. However, various historical items which were previously lost are likely to remain in the RPV. The licensee informed the inspectors that they plan to perform a series of visual inspections and extensive cleaning of the RPV during the next inspection period. This effort will include removal of several control rod guide tubes to gain access to the lower RPV head region.

c. Conclusions

Based on review of ongoing activities and corrective action documents the inspectors determined that the licensee's actions to flush the RWCU System were adequate to support removal of remaining purge dam material and to ensure that the system satisfied documented cleanliness requirements and that the licensee's program for FME while working in the RPV, reactor cavity, or refuel storage pool satisfied regulatory requirements and licensee commitments. The inspectors concluded that the licensee's plan to perform extensive RPV inspections and retrieval of FME prior to completion of in-vessel activities was an aggressive safety conscience decision.

V. Management Meetings

X1 Exit Meeting Summary

On November 7, 2005, the resident inspectors presented the inspection results to Mr. Jon Rupert and other members of his staff, who acknowledged the findings. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

M. Bali, Design Engineering, Unit 1
R. Baron, Nuclear Assurance Manager, Unit 1
M. Bennett, QC Manager, Unit 1
T. Bowlin, Field Engineering, Unit 1
A. Brock, Field Engineering, 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. Dean, EQ Engineer, Unit 1
J. Dizon, Facility Risk Consultants
S. Eder, Facility Risk Consultants
B. Hargrove, Radcon Manager, Unit 1
K. Hess, SWEC Project Director
E. Hollins, Maintenance and Modifications Manager, Unit 1
R. Jackson, Bechtel
B. Ditzler, TVA Welding Engineering Supervisor, Unit 1
G. Jones, Design Field Support, Unit 1
R. Jones, Plant Recovery Manager, Unit 1
S. Kane, Licensing Engineer
D. Kehoe, Nuclear Assurance, Unit 1
J. Lewis, ISI Program Engineer, Unit 1
G. Lupardus, Civil Design Engineer, Unit 1
J. McCarthy, Licensing Supervisor, Unit 1
J. Ownby, Project Support Manager, Unit 1
J. Rupert, Vice President, Unit 1 Restart
J. Schlessel, Maintenance Manager, Unit 1
J. Symonds, Modifications Manager, Unit 1
E. Thomas, Bechtel
D. Tinley, NDE Level III & Unit 1 ISI Project Manager
J. Valente, Engineering Manager, Unit 1

INSPECTION PROCEDURES USED

IP 37550	Onsite Engineering
IP 37551	Engineering
IP 71111.17	Permanent Plant Modifications
IP 71111.23	Temporary Plant Modifications
IP 92701	Follow-up
IP 50090	Pipe Support and Restraint Systems
IP 55050	Nuclear Welding General Inspection Procedure
IP 57080	Nondestructive Examination Procedure Ultrasonic Examination Procedure Review / Work Observation / Record Review

IP 57090	Radiographic Examination Procedure Review/Work Observation/Record Review
IP 71153	Event Followup
IP 73055	Preservice Inspection
IP 73501	Inservice Inspection - Review of Program
IP 37550	Onsite Engineering
IP 37551	Engineering
IP 71111.17	Permanent Plant Modifications
IP 71111.23	Temporary Plant Modifications
IP 71152	Identification and Resolution of Problems
IP 92701	Follow-up
IP 71111.08	Inservice Inspection Activities

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

50-259/05-08-01	NCV	Engineering did not Follow Procedures and Document on the Heat Shrink Data Sheets all the Parts Required to Install Four Splices on Multi-Conductor cables. (Section E1.7)
50-259/05-08-02	NCV	Measures were not Adequate to Assure that the TOLs in 480V MOV Board 1B Cubicles, 14C-2 and 15C, were Strapped Out. (Section E1.9)

Closed

94-03	GL	Intergranular Stress Corrosion Cracking of Core Shrouds in BWRs (Section E8.1)
88-14	GL	Instrument Air Supply System Problems Affecting Safety Related Equipment (Section E8.2)
96-06	GL	Assurance of Equipment Operability and Containment Integrity During Design Basis Accident Conditions (Section E8.3)
87-26-03	URI	RHR Pump Suction and Nozzle Load Allowables are not exceeded (Section E8.4)
II.K.3.27	TMI	Common Reference Level for BWRs (Section E8.5)

Discussed

None

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

DCNs

DCN 51076, Reactor Feedwater Instrumentation and Control - Control Bay, System 3
DCN 51090, 480 VAC Reactor Motor Operated Valve (RMOV) Boards and 480 VAC Shutdown Boards - Control Bay, Systems 57- 4
DCN 51177, RHRSW - Reactor Building, System 23 DCN 51199, RHR - Reactor Building, System 74
DCN 51106, CRDR for Auxiliary Instrument Control Panel 1-9-25-32
DCN 51216, 480V Distribution - Reactor Building, System 57-4 DCN 51189, Primary Containment - Reactor Building, System 64A
DCN 51217, 4160 VAC Distribution - Reactor Building, System 57-5
DCN 51240, Control Rod Drive - Reactor Building, System 85

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
FP-0-247-INS004, Appendix R Battery Operated Emergency Lighting Quarterly Test, Revision 20

Other Documents

TACF 1-03-002-023, RHRSW Secondary Containment Boundary
TACF 1-04-003-023, RHRSW Temporary Flush Piping
TACF 0-05-4005-044, System 44, Building Heating
TACF 1-04-013-069, Revision 1, RWCU Primary Containment Isolation Valves
TACF 0-04-005-247, Appendix R Emergency Lighting

Section E1.3: System Return to Service Activities

Procedures, Guidance Documents, and Manuals

Technical Instruction 1-TI-437, System Return to Service (SRTS) Turnover Process for Unit 1 Restart, Revision 0
0-TI-404, Unit One Separation and Recovery, Revision 4
1-TI-474, Cleanliness Verification Program, Revision 0
0-TI-373, Plant Lay-up and Equipment Preservation, Revision 4
MSI-1-000-PRO001, Cleanliness of Unit 1 Fluid Systems, Revision 1

Problem Evaluation Reports (PERs)

64466, structural splice was identified on a steel frame, section 6WF15.5, in the reactor building that was questionable, some new welds could not be completed because specific weld locations were inaccessible.
87549, wrong drain valves, Parker Hannifin needle valves, were installed instead of the designed Dragon globe valves, as required by DCA 51177-23.
87913, instrumentation sense lines for flow transmitters 1-FT-23-48 and 54 damaged and out of their small bore piping supports.
88064, electrical cables were installed in the reactor building with a vertical drop of greater then 25 feet with out required vertical cable supports
86339, Post Issuance Change (PIC) 64711 AA-01, issued against DCN 51192, which required installation of check and manual shutoff valves did not have enough design information to install the valves.
87099, during a fit-up inspection wrong material was being used for a 4 inch by 1 inch sockolet. When PIC 61520 AA-09, issued against DCN 51192, changed the material specification the material change was not incorporated into the work document.
87309, PIC 61211 AA-04, issued against DCN 51192, allowed the installation of a flexible plastic form through the reactor building wall because conduit sleeves could not be installed. The sleeves could not be installed due to interference with rebar. The flexible form was made of combustible material and the PIC did not provide enough detail to trim the form back to the inside the concrete wall.
88708, during cable pulls deficiencies were observed including signatures missing for verification of prerequisites prior to pulling, verification of attributes during pulling, and post pull verifications; data sheets were missing break link strength entries, and entries for partial pulls did not clearly state the From - To locations; and not listing the number of break links used for a specific pull.
90015, jet pump instrumentation sensing lines did not meet the 1/4 inch per foot slope requirement;
89872, sensing lines for the flow instrumentation on the RHRSW discharge of 1C RHR heat exchanger did not meet slope requirements;
87913, sensing lines for the flow instrumentation on the RHRSW discharge of 1D RHR heat exchanger did not meet slope requirements
89872, sensing lines for flow transmitter 1-FT-23-36 on the RHRSW discharge of 1A RHR heat exchanger did not meet slope requirements.

Section E1.4: Restart Test Program

Procedures and Standards

Technical Instruction 1-TI-469, Baseline Test Requirements, Revision 1
Operating Instruction, 1-OI-69, Reactor Water Cleanup System, Revision 27
Surveillance Instruction 1-SI-3.3.3, ASME Section XI System Pressure Test of Fuel Pool Cooling System, Revision 0
Post Modification Test Instruction (PMTI) 1-PMTI-BF-51090-S57-64+S79, Functional Testing of 480-VAC Reactor MOV Bards and 480-VAC Shutdown Boards - Control Bay, System 57-4, Revision 0

PMTIs

1-PMTI-BF- 51090-S57-64+S79 (Stages 57 Thru 64 and Stage 79), Revision 0
1-PMTI-51100-STG06, Stage 6 and Stage 7, Revision 0
1-PMTI-51203-STG02, Revision 0
1-PMTI-023-052, Stage 25, Revision 0
1-PMTI-002-011, Stage 5, Revision 0

Problem Evaluation Reports (PERs)

79750, test deficiencies during performance of 1-PMTI-BF-51090-S57-64+S79
83861, label deficiencies identified during leak rate test on 1-SHV-75-54A and 54B
84205, failure mode identified in fuel pool cooling logic controls

Other Documents

BTRD 1-BFN-BTRD-079.002, System 79, Fuel Handling and Storage System, Revision 2

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

Procedures and Standards

MAI 1.3, General Requirements for Modifications, Revision 21
MAI-3.2, Cable Pulling for Insulated Cables Rated Up to 15,000 Volts, Revision 41
MAI-3.3, Cable Terminating and Splicing for Cables Rated Up to 15,000 Volts, Revision 45
MAI-3.7, Cable Pull Force Monitoring Breaklink Fabrication, Verification, and Control, Revision 6

DCNs

DCN 51216, Electrical 480V Distribution - Reactor Building, System 57-4.

Work Orders

03-001001-051, replace alternate feeder cables for the 480V RMOV Board 1A
03-001001-053, replace alternate feeder cables for the 480V RMOV Board 1B
03-023434-004, replace cables for the 4160/480V Transformer TS1E, alternate power for 480V Shutdown Board A and 480V Shutdown Board B
03-001001-049, replace cables for the 4160/480V Transformer TDE, alternate power for 480V Diesel Auxiliary Board A and 480V Diesel Auxiliary Board B

Section E1.6: Special Program Activities - Fuse Program

Procedure(s)

TVAN Standard Department Procedure OPDP-7, Fuse Control, Revision 3

Work Order Packages

WO 03-021841-093, Completed 1/18/05
WO 03-021841-056, Completed 3/23/05
WO 03-021841-099, Completed 1/18/05
WO 03-021841-002, Completed 3/14/05
WO 03-021841-009, Completed 1/19/05
WO 03-021841-035, Completed 1/19/05
WO 03-021841-036, Completed 12/23/04

Calculations

ED-Q0067-920666, 480V Reactor MOV Boards 1A/1B Control Circuit Fuse Sizing, Revision 007
ED-Q0268-880134, Fuse Program - 480V Reactor MOV Boards 1A/B, Revision 013

Design Change Notices

51090-Stage-002, Master Equipment List (MEL) Pages 10, 14, 18, 22
51090-Stage-07, MEL Page 4
51090-Stage-007, MEL Pages 4, 6, 12
51090-Stage-013, MEL Pages 8, 12
51090-Stage-014, MEL Pages 8, 12
51090-Stage-024, MEL Pages 24, 28, 32, 36, 40, 44
51090-Stage-034, MEL Pages 8, 12
51090-Stage-036, MEL Pages 38, 42, 46, 50, 54, 58, 62, 66, 70, 74, 78, 82, 86, 90

Section E1.8: Special Program Activities - Environmental Qualification of Electrical Equipment

EQ Change Supplements

BFN0EQ-CABL-010-51177, Revision 1
BFN0EQ-CABL-022-51090, Revision 0
BFN0EQ-CABL-036-51177, Revision 2
BFN0EQ-CABL-014-51217, Revision 1
BFN0EQ-CABL-029-51217, Revision 2
BFN0EQ-CABL-024-51222, Revision 0
BFN0EQ-SPLC-005-51177, Revision 0
BFN0EQ-SPLC-004-51046, Revision 0
BFN0EQ-CABL-041-51177, Revision 0
BFN0EQ-SPLC-001-51177, Revision 4
BFN0EQ-IPS-001-51192, Revision 0
BFN0EQ-SOL-003-51192, Revision 0
BFN0EQ-SPLC-001-51177, Revision 8
BFN0EQ-SPLC-002-51217, Revision 1

Calculations

ND-Q0067-870015, Master Components Electrical List, Revision 10
ND-Q0999-2002-0019, Units 0, 2, and 3 Equipment for Unit 1 Safe Shutdown, Revision 1

Other Documents

BFN Circuit Failure Analysis Work Package DCN 51192, Revision A

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
TVA General Engineering Specification G-32, Bolt Anchors set in Hardened Concrete, Revision 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)
Addendum 2 to Process Specification G-29-S01, 3.C.5.5, Visual Examination of Welds, Revision 0
MAI-4.2A, TVA-BFNP Piping/Tubing Supports, Revision 33, dated 3/29/05
MMDP-10, Controlling Welding, Brazing, and Soldering Processes, Revision 4, dated 1/15/03
Walkdown Instruction WI-BFN-0-GEN-01, General requirements for BFN Unit 1 Walkdowns, Revision 4, dated 4/19/04
Walkdown Instruction WI-BFN-0-CEB-03, Engineering Attribute Walkdown Instructions for Seismic Class I Small Bore Piping, Tubing and Associated Supports, Revision 0, dated 8/26/02

Drawings

0-47B435-1 through -21, Mechanical General Notes, Pipe Supports
1-47B458-922 through -926, and -928, Mechanical Core Spray Cooling System, Pipe Support, Revision 4
0-47B436-67, Mechanical Typical Pipe Supports, Class I and II Structures, Revision 1
1-47B455-2157 and -2158, Mechanical HPCI System, Pipe Support, Revision 0
Drawings for Mechanical RCIC System, Pipe Support
Drawings for Mechanical Reactor Feedwater System, Pipe Supports
Drawings for Mechanical Reactor Core Spray Cooling System, Pipe Supports
Drawings for Mechanical SLC System, Pipe Supports

Calculations

CDQ1-075-2002-0822, Revision 9, Small Bore Piping and Supports Program System Calculation for Unit 1 Seismic Class 1 CS System (75) Piping
CDQ1-071-2002-0824, Revision 1, Small Bore Piping and Supports Program System Calculation for Unit 1 Seismic Class 1 RCIC System (71) Piping
CDQ1-003-2002-1123, Revision 2, Pipe Stress Analysis of Stress Problem N-1-103-51R, -53R, -54R, and -55R

CDQ1-075-2002-0867, Revision 2, Pipe Stress Analysis of Stress Problem N-1-175-53R, and -54R
CDQ1-074-892710, Revision 8, Pipe Stress Analysis of Stress Problem N-1-174-9R
CDQ1-003-2002-0873, Revision 3, Small Bore Piping and Supports Program System Calculation for Unit 1 Seismic Class 1 FW System (03) Piping
CDQ1-073-2003-0248, Revision 10, Summary of Piping Analysis, N1-173-5R
CDQ1-303-2003-0672, Revision 2, dated 11/15/04, Qualification of Anchor Frame for RHR Pump 1A Suction
CDQ1-303-2003-0662, Revision 1, dated 11/15/04, Qualification of Anchor Frame for RHR Pump 1B Suction
CDQ1-303-2003-0663, Revision 2, dated 11/8/04, Qualification of Anchor Frame for RHR Pump 1C Suction
CDQ1-303-2003-0664, Revision 1, dated 11/9/04, Qualification of Anchor Frame for RHR Pump 1D Suction

Miscellaneous Documents

DCN 51255, Feed Water System - Small Bore Piping
DCN 51260, Core Spray System - Small Bore Piping
DCN 51348, Core Spray System - Small Bore Piping
DCN 51416, Core Spray System - Small Bore Piping
TVA Browns Ferry Nuclear Plant, Qualification of Seismic Class I Safe Shutdown Instrument Tubing Task S051, Final Report, dated 8/29/89
TVA Browns Ferry Nuclear Plant, Small Bore Piping Assessment Program, Final Report, dated 6/22/90
Enrico Fermi Atomic Power Plant, Unit No. 2, Evaluation of Containment Coatings, Revision 4, June, 1985
TVA letter to NRC, Browns Ferry Nuclear Plant - Containment Coatings, Docket No. 50-260, dated October 4, 1989
Section 3.7, Containment Coatings, of NUREG-1232, Volume 3, Supplement 2, Safety Evaluation Report on TVA Browns Ferry Unit 2 Restart, dated January, 1991
General Design Criteria Document BFN-50-C-7100, Design of Civil Structures, Revision 13, dated 12/20/00
TVA Nuclear Engineering Civil Design Standard DS-C1.7.1, General Anchorage to Concrete, Revision 9, dated 8/25/99
TVA letter dated February 27, 1991, Subject: Browns Ferry Nuclear Plant - Action plan to Disposition Concerns Related to Small Bore Piping Program
TVA letter dated December 12, 1991, Subject: Browns Ferry Nuclear Plant - Small Bore Piping Program, Tubing, and Conduit Supports for Units 1 and 3 - Additional Information
General Design Criteria Document BFN-50-C-7103, Structural Analysis and Qualification of Mechanical and Electrical Systems (Piping and Instrument Tubing), Revision 5, dated 9/9/91
General Design Criteria Document BFN-50-C-7107, Design of Class I Seismic Pipe and Tubing Supports, Revision 7, dated 4/6/94

Section E1.11: Special Program Activities - Control Rod Drive Insert and Withdrawal Piping

Drawings

Drawings for Mechanical Control Rod Drive System, Pipe Supports

Section E1.12: Special Program Activities - Configuration Management/Design Baseline

Procedures

EEB-TI-23 R6 final App H, Setpoint Calculations
SSP-6.7 Instrumentation Setpoint Calibration and Scaling Program, Revision1
0-TI-175 Control Air Sampling, Revision 10
MIS-1-001-TST001, Testing of Air Supply System (Steam Tunnel Side) for the Main Steam Isolation Valves, Revision 1

Calculations

ED-02999-920170, 1-P-71-4,-12, 1-P-73-4,-21 Setpoint & Scaling Calculation for Pressure Instruments, Revision 4
ED00071920444, Setpoint and Scaling Calculation for 1,3 PDT 71-1A&1B,I,3 PDIS 71 1A&1B, Revision 3
ED-Q0011-930022, RCIC High Steam Flow Isolation Time Delay, Revision 2
EDN0073930027, Setpoint and Scaling Calculation for 1-T-73-54, 2-T-73-54 & 3-T-73-54, Revision 4
EDQ0073930078, Setpoint and Scaling Calculation for 1-P-73-29, Revision 7
ED00073930079, Setpoint and Scaling Calculation for 1-63-73-29, Revision 2
E000073930141, Setpoint and Scaling Calculation for HPCI Pump Flow, Revision 6
BFEP-M2-P0766, Evaluation of HPCI Turbine Control Oil Modification, Revision 0
EDQ1-248-2002-0042, Appendix R Analysis for Unit/shutdown Rd Batt, Revision 8
EDQ1-999-2002-0019, Cable Ampacity Calculation for Dry Well Power Cables, Revision 5
EDQ1-999-2002-0061, 260-VDC Bus and Cable Protection and Breaker/fuse Coordination, Revision 11
EDQ1-999-2002-0071 480-V Motor Control Centers, Cable And Bus Protection I Breaker Coordination, Revision 9
EDQ1-999-2002-0075, Thermal Overload Heater Calculation - Motor Operated Valves, Revision 8
EDQ1-999-2002-0076, Thermal Overload Heater Calculation - Continuous Duty Motors, Revision 4
MDQ0069880274, Raised Design Temperature - Temp Selection - Study, Revision 4
MDQ0071910235, Reactor Core Isolation Cooling System Design Pressure/Temperature, Revision 8
MDQ0073870190, HPCI Piping Pressure Drop and NPSHA, Revision 10
MDQ0073890041, Analytical/Operational Limits for the HPCI Turbine Control, Revision 4
MDQ0073920184, Automatic Transfer to Suppression Pool for LS-73-56A, 6GB, 57A, and 575 Analytical Limits for HPCI Suction, Revision 7
MDQ0999920193, Analytical Limits for HPCI and RCIC System Isolation Temperature, Revision 5
MDQ007320020113, Critical Flow Calculation, Revision 1
MDQ099920040040, HPCI & RCIC System Test Requirements, Revision 2
MDQ107120020012, Unit 1 Reactor Core Isolation Cooling System Modes of Operation, Revision 2
MDQ107120020095, MOV 1-FCV-071-0008, Operator Requirements and Capabilities, Revision 0
MDQ107320020013, Unit 1High Pressure Coolant Injection Modes of Operation, Revision 2
MDQ107320020102, MOV 1-FCV-073-0035; Operator Requirements and Capabilities, Revision 1

MDQ199920020033, Building High Energy Line Break Mass and Energy Release for Environmental Analysis, Revision 0
MDQ0999200050010, Mini-calculation for Safe Shutdown Analysis, Revision 0
MDQ0069880229, Material Adequacy Raised Design Temperature Study, Revision 4
NDQ0073980028, High Pressure Coolant Injection Time Delay Relays, Revision 2
NDQ0999980003, Analytical Limits for RPS/ECCS/LOCA Analyses, Actions, and Permissives, Revision 9

DCNs

51182, Drywell Control Air Systems

Section E1.13: Special Program Activities - Seismic II/I Spatial System Interactions and Water Spray

Procedures and Standards

WI-BFN-0-CEB-06, Engineering Walkdown Instruction for Evaluation of Seismic-Induced Spray Hazards
Design Criteria BFN-50-C-7306, Qualification Criteria for Seismic Class II Piping, Pipe Supports, and Components

DCNs and Work Documents

51669, U1 Recovery Reactor Building: Seismic II/I Water Spray, a total of 18 outliers were identified that required modifications

Other Documents

Facility Risk Consultants, Inc., Seismic-Induced II/I Spray Evaluations at Browns Ferry Unit 1, March 2004

Section E1.14: Special Program Activities - Cable Tray Supports and Conduit Supports

Procedures and Standards:

WI-BFN-0-GEN-01 Walkdown Instructions, Revision 1
BFN-50-C-7104, Design of Structural Supports, Revision 12
WI-BFN-0-CEB-04, Seismic Verification Walkdown Instruction for USI A-46 and Seismic IPEEE, Revision 0
VE-2-2001 NEMA Standards Publication, 2001.
Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment, March 1993 - Seismic Qualification Utility Group (SQUG)

DCNs:

51521, U1 Recovery Reactor Building Structural Modification Required by A-46 evaluation.
51227, BFN Unit 1 Recovery - Electrical Lead DCN - System 362 - Rx Bldg Cable Tray
51223, BFN Unit 1 Recovery - Electrical modifications of existing cables, cable raceways, components and equipment for Core Spray System (75)

Drawings:

1-48B800-3639, -3638, -3642, -3641
1-45B830-124
1-45E830-71, -72, -73, -74, -75, -76
1-48B805-212, -213, -214, -215, -216, -217, -218, -219, -220, -221, -222
1-45N830-17, -1, -87
1-48N880

Calculations:

CDQ1-000-2003-2203, USI A-46 Seismic Verification of Cable Tray and Conduit Raceway Systems for BFN Unit 1
CD-Q0000-931227, Qualification of Cable Tray and Conduit Systems by A-46 Program
CDQ1-362-2004-0212, Bounding Calculation for unit 1 Reactor Building Cable Tray Support Design per DCN 51227
CDQ1-361-2004-0280, Bounding Calculations for Unit 1 Reactor Building Conduit Support Design per DCN 51223
CDQ1-361-2005-0141, Bounding Calculations for Unit 1 Reactor Building Support Design per PIC 63240 of DCN 51223

Walkdown Packages:

BFN1 CEB RCWY DW, Cable Tray and conduit Review for USI A-46 and Seismic IPEEE - Documentation for BFN Unit 1 Drywell
BFN1 CEB RCWY 519, Cable Tray and conduit Review for USI A-46 and Seismic IPEEE - Documentation for BFN Unit 1 Reactor Building El. 519
BFN1 CEB RCWY 639, Cable Tray and conduit Review for USI A-46 and Seismic IPEEE - Documentation for BFN Unit 1 Reactor Building El. 639
BFN1 CEB RCWY 621, Cable Tray and conduit Review for USI A-46 and Seismic IPEEE - Documentation for BFN Unit 1 Reactor Building El. 621
BFN1 CEB RCWY 593, Cable Tray and conduit Review for USI A-46 and Seismic IPEEE - Documentation for BFN Unit 1 Reactor Building El. 593
BFN1 CEB RCWY 565, Cable Tray and conduit Review for USI A-46 and Seismic IPEEE - Documentation for BFN Unit 1 Reactor Building El. 565

Section E1.16: Special Program Activities - Intergranular Stress Corrosion Cracking (IGSCC)

DCNs and Work Documents

DCN 51193, Unit 1 Reactor Building Mechanical System 68
WO 02-016506-006, Replace the 48 RPV shroud bolts in Unit 1 with newly designed bolts
WO 02-016304-000, Perform Reactor Vessel Inspections

Procedures

1-TI-504, Mechanical Stress Improvement Process, Revision 1
GE-UT-503, Procedure for Automated Ultrasonic Examination of the Shroud Assemble Welds, Revision 13
GE-VT-204, Procedure for In-vessel Visual Inspections of BWR 4 RPV Internals

Technical Instruction TI-365, Reactor Pressure Vessel Internals Inspection (RPVII) Units 1, 2, and 3, Revision 16

Section E1.17: Inservice/Preservice Inspection

Procedures and Standards

N-UT-76, Generic Procedure for Ultrasonic Examination of Ferritic Pipe Welds, Revision 4
N-UT-64, Generic Procedure for the Ultrasonic Examination of Austenitic Pipe Welds, Revision 7
N-UT-82, Generic Procedure for the Ultrasonic Examination of Dissimilar Metal Pipe Welds, Revision 1
N-MT-6, Magnetic Particle Examination For ASME and ANSI Code Components and Welds, Revision 26
N-PT-9, Liquid Penetrant Examination of ASME and ANSI Code Components and Welds, Revision 27

E1.18: Power Service Shop Activities

Procedures and Standards

0-TI-494, Repair and Reconditioning Specification for AC Squirrel Cage Motors with Voltage Ratings of 2.3 to 6.9 KV, Revision 1
PSS Specification J1RA-GEN.3.50, Electrical Motor Testing

Work Orders

PSS Job Order QQ243, Perform refurbishment of Unit 1 RHR motor S/N LEJ1126002

Section E8.3: Generic Letter 96-06, Assurance of Equipment Operability and Containment Integrity During Design Basis Accident Conditions

Procedures and Standards

1-GOI-200-2, Drywell Closeout, Revision 0000
2-GOI-200-2, Drywell Closeout, Revision 27
3-GOI-200-2, Drywell Closeout, Revision 16
0-OI-2C, Demineralized Water System, Revision 0051
0-OI-2C, Attachment 1, Demineralized Water System Valve Lineup Checklist-Unit 0, April 29, 2002
0-OI-2C, Attachment 1A, Demineralized Water System Valve Lineup Checklist-Unit 1, April 28, 2005
0-OI-2C, Attachment 1B, Demineralized Water System Valve Lineup Checklist-Unit 2, December 5, 2002
0-OI-2C, Attachment 1C, Demineralized Water System Valve Lineup Checklist-Unit 3, June 30, 2003
SPP-2.6, Computer Software Control, Revision 10

Problem Evaluation Reports (PERs)

BFPER970209, Engineering Evaluation for NRC GL 96-06, 02/03/97

DCNs

51154, U1 Restart Drywell Mechanical Lead System 077, Revision A

Other Documents

Drawing 67 M 1-47E482-3, R001, Mechanical radwaste sump pump piping discharge & miscellaneous piping
Drawing 67 M 1-47E856-2, R001, Flow Diagram Demineralized Water
Licensee Event Report (LER) 50-259/97001, A potential overpressurization condition of a containment penetration pipe due to thermal expansion of entrapped water was identified, 03/05/97

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

Miscellaneous Documents

Nuclear Assurance Assessment Plan for Power Service Shop, August 2005

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

04-723643-00, Perform LLRT of selected valves in the Reactor Feedwater System using Procedure TI-106, General Leak Rate Test Procedure

Problem Evaluation Reports (PERs)

85399, extent of condition of the problems identified with the type EC trip devices

Section M1.2: System Cleanliness

Procedures and Standards

Technical Instruction 1-TI-437, System Return to Service (SRTS) Turnover Process for Unit 1 Restart, Revision 0
0-TI-404, Unit One Separation and Recovery, Revision 4
1-TI-474, Cleanliness Verification Program, Revision 0
0-TI-373, Plant Lay-up and Equipment Preservation, Revision 4
MSI-1-000-PRO001, Cleanliness of Unit 1 Fluid Systems, Revision 1
SPP-6.5, Foreign Material Control, Revision 10

Work Orders

04-713476-000, Perform flush of RWCU system
05-717682-000, Perform flush of RWCU system

Problem Evaluation Reports (PERs)

51772, Purge dam material remaining in systems and potentially impacting system functionally
81336, Triangular fragment found missing from corner of steam dryer support lug
84100, Piece of tool used for in vessel instrument removal dropped into reactor cavity
84116, EDM tool broke and piece fell on RPV annulus floor
84455, Piece of tool used for in vessel instrument removal dropped into reactor cavity
84529, GE identified historical items in RPV annulus
84567, DVD dropped in dryer separator pit
84863, Roll pin fro UT tool came loose and fell on RPV annulus floor
86448, Pry bar fell from fuel prep machine into fuel pool
86609, Purge dams left in RWCU system
86782, Potential adverse trend with 11 instances of items dropped in the FME zone on refuel floor
87078, Brush and nut came loose and fell during brushing control rod guide tube
87079, Metal shavings observed coming out of Core Spray T-box thermal sleeve
87156, Various historical items found in RPV during ongoing exam of internals
89054, Tip broke off of end of IRM drytube and fell to refuel pool floor
89451, Dummy beam bolt keeper fell from UT fixture onto Jet Pump 13
89549, Historical metallic FME item discovered in peripheral fuel assembly location 53-50