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

February 11, 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 05000259/2004009

Dear Mr. Singer:

On January 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 February 3, 2005 with Mr. John Rupert and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations, with the commitments in your Unit 1 Recovery Program, and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel. Based on the results of this inspection, no findings or violations of significance were identified.

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 05000259/2004009
w/Attachment: Supplemental Information

cc w/encl: (See page 2)

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

Larry S. Bryant
Engineering and Technical Services
Tennessee Valley Authority
Electronic Mail Distribution

Michael D. Skaggs
Site Vice President
Browns Ferry Nuclear Plant
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

Kurt L. Krueger, Plant Manager
Browns Ferry Nuclear Plant
Tennessee Valley Authority
Electronic Mail Distribution

Fredrick C. Mashburn
Sr. Program Manager
Nuclear Licensing
Tennessee Valley Authority
Electronic Mail Distribution

Timothy E. Abney, 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 page 4)

Distribution w/encl:
E. Brown, NRR
L. Slack, RII EICS
RIDSRIDSNRRDIPMLIPB
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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No:	50-259
License No:	DPR-33
Report No:	05000259/2004009
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Browns Ferry Nuclear Plant, Unit 1
Location:	Corner of Shaw and Nuclear Plant Roads Athens, AL 35611
Dates:	October 10, 2004 - January 15, 2005
Inspectors:	W. Bearden, Senior Resident Inspector, Unit 1 E. Christnot, Resident Inspector P. Fillion, Reactor Inspector (Sections E1.3, E1.4) B. Crowley, Senior Reactor Inspector (Sections E1.5, E1.6) J. Lenahan, Senior Reactor Inspector (Section E1.7) C. Smith, Senior Reactor Inspector (Section E1.8) G Wiseman, Senior Reactor Inspector (Sections F1.1, F1.2, F1.3, F7.1)
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 05000259/2004-009

This integrated inspection included aspects of licensee engineering and modification activities associated with the Unit 1 recovery project. 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. The report covered a 3-month period of resident inspector inspection. In addition, NRC staff inspectors from the regional office conducted inspections of Fire Protection Program Unit 1 activities associated with the Unit 1 recovery project and inspections of the licensee's Recovery Special Programs in the areas of cable installation/separation; cable ampacity; flexible conduit; and containment coatings.

Inspection Results - Engineering

- Inspectors reviewed five Unit 1 modification design packages and concluded that the design changes were appropriately developed, reviewed, and approved for implementation per procedural requirements. The packages adequately addressed changes needed for Unit 1 operation per current requirements. (Section E1.1)

- Modification installation activities associated with five permanent plant design changes were observed and found to be performed in accordance with the documented requirements. (Section E1.2)

- The licensee's Special Programs to resolve previous problems with Cable Installation and Cable Separation continued to support Unit 1 restart. Inspection of the methodology for the Cable Installation and Cable Separation Special Programs, together with performance based inspection, led to the inspector's conclusion that implementation of these programs is proceeding in accordance with licensee commitments. (Section E1.3)

- The licensee's Special Program to resolve previous problems with Flexible Conduit continued to support Unit 1 restart. Inspection of the methodology for the Flexible Conduit special program and inspection of installed flexible conduit in the control complex led to the inspector's conclusion that implementation of this program is proceeding in accordance with licensee commitments. (Section E1.4)

- The licensee's intergranular stress corrosion cracking (IGSCC) mitigation plan continued to meet commitments established by licensee Regulatory Framework letters. Recirculation (RECIRC) System, Residual Heat Removal (RHR) System, Reactor Water Cleanup (RWCU) System, and Core Spray (CS) System piping replacement activities were meeting ASME Code and other regulatory requirements. (Section E1.5)

- The licensee's Inservice/Preservice Inspection (ISI/PSI) Program was meeting applicable regulatory requirements and licensing commitments. (Section E1.6)

- The licensee's program for restoration of the coatings in the Unit 1 torus was determined to comply with NRC requirements. Repairs to identified coating problems were completed by qualified individuals and accomplished in accordance with approved procedures. (Section E1.7)

- Modifications implemented to resolve Cable Ampacity issues were acceptable to support Unit 1 restart. (Section E1.8)

- The Control Room Design Review program was adequate to resolve previously identified Human Engineering Deficiencies. (Section E1.9)

- Inspectors reviewed one new temporary alteration to support a temporary air supply for Unit 1 torus activities and four temporary alterations issued to convert Unit 1 modifications, previously implemented under existing plant procedures. Inspectors determined the alterations did not cause any significant impact on the operability of equipment required to support operations of Units 2 and 3. (Section E1.10)

- The licensee's System Return to Service (SRTS) activities continued to be performed in accordance with procedural requirements. System deficiencies were identified and appropriately addressed by the licensee's corrective action program. (Section E1.11)

- Overall implementation of restart testing activities for the Fuel Pool Cooling System was acceptable. Test deficiencies identified during performance of testing were adequately documented under the licensee's corrective action program. (Section E1.12)

- Increased management focus and oversight of the welding program has resulted in fewer documentation errors and an improvement in welder performance. (Section E7.1)

<u>Inspection Results - Maintenance</u>

- Based on review of records and observation of ongoing work, the licensee's General Electric Type HFA relay replacement program was complying with the applicable requirements. (Section M1.1)

<u>Inspection Results - Plant Support</u>

- Implementation of the Unit 1 Browns Ferry Fire Protection Improvement Plan is proceeding in accordance with licensee commitments. A modification to upgrade the fire alarm and detection system was technically acceptable and appropriately developed, reviewed, and approved for implementation per procedural requirements. Modifications to upgrade fire alarm and detection system were technically acceptable and appropriately developed, reviewed, and approved for implementation per procedural requirements. (Section F1.1)

- Implementation of the plant's fire prevention program for the control for storage of permanent and transient combustible materials and ignition sources met fire protection program procedures. Cutting and welding operations in progress were properly authorized by a permit and a fire watch with an extinguisher posted for the duration of the work. Transient fire loads associated with plant maintenance, modifications, and construction activities were being adequately controlled by the licensee's fire protection program. (Section F1.2)

- Implementation of the plant's valve position and controls for fire suppression systems was satisfactory. Valve positions were generally controlled and secured as indicated on the system drawings and operating procedures. Accessibility to the standpipe systems hose racks in the Unit 1 reactor building by fire brigade personnel was considered degraded and could hinder fire brigade access to the stations. The licensee initiated a corrective action document to review this area. Additionally, portions of the Unit 1 reactor building fire suppression water supply piping were noticeably degraded; however, future modifications to improve the material conditions were planned before restart of Unit 1. (Section F1.3)

REPORT DETAILS

Summary of Plant Status

Unit 1 has been shut down since March 19, 1985, and has remained in a long-term lay-up condition with the reactor defueled. The licensee initiated Unit 1 recovery activities to return the unit to operational condition following the TVA Board of Directors decision on May 16, 2002. During the current inspection period, reinstallation of plant equipment and structures continued. Recovery activities include ongoing replacement of reactor coolant system piping; reinstallation of balance-of-plant piping and turbine auxiliary components; and installation of new electrical penetrations, cable trays, and cable tray supports. Limited system return to service (SRTS) activities occurred during this reporting period.

II. Engineering

E1 Conduct of Engineering

E1.1 Design Change Notice (DCN) Package Reviews (37551)

a. Inspection Scope

The inspectors reviewed permanent plant modifications to the Residual Heat Removal System, Core Spray System, Reactor Water Cleanup System, Main Steam System, and 4 KV electrical breakers . The inspectors reviewed criteria in licensee procedures 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.

b. Observations and Findings

b.1 DCN 51087, 4KV Electrical Breakers, System 57-4

The inspectors reviewed the Unit 1 permanent plant modification DCN 51087, Modifications of Electrical 4KV breakers in the Control Building and Turbine Building. The intent of this DCN was to remove existing General Electric magne blast circuit breakers and replace them with new Wyle/Seimens vacuum type circuit breakers. The DCN will be implemented on 4KV Shutdown Boards A, B, C, and D; Unit Boards 1A, 1B, and 1C; Recirculation Pump Board 1; and Recirculation Pump Trip Boards 1-I and 1-II. The DCN will replace the circuit breakers for the following equipment: Core Spray pumps and RHR pumps located on the Shutdown Boards A, B, C, and D; condensate pumps, condensate booster pumps, control rod drive feed pumps, and condenser circulating pumps located on the Unit Boards; the unit start pump breakers and the normal pump breakers located on the Recirculation Pump Board; and the recirculation pump breakers located on the Recirculation Pump Trip Boards. Based on information included in the DCN the inspectors concluded that the new Wyle/Seimens Vacuum type circuit breakers were designed to fit into the same compartments as the GE circuit breakers without requiring any changes to the switchgear; this change was similar to the changes previously implemented on Units 2 and 3; and the new breakers will perform the same function as the old and would not require any special testing.

During the review of DCN 51087, Electrical Modifications in the Control Building and Turbine Building, System 57-4, the inspectors observed an administrative error involving trip timing requirements for the EOC trip. The ATWS and End of Cycle trips for the Recirculation Pump Trip Boards 1-I and 1-II require two separate trip coils for each recirculation pump. The new Wyle/Seimens circuit breakers are equipped with two separate coils to satisfy these two trip requirements. DCN 51087 states, in part, that the full opening of the breaker be less than 135 milliseconds at 100 percent reactor power. The new Wyle/Seimens circuit breakers operate in a vacuum and were designed to open in a 5 cycle maximum time and this corresponds to approximately 35 milliseconds. The inspectors determined that the correct corresponding time to the 5 cycle maximum is approximately 85 milliseconds rather than 35 milliseconds. Additionally, it was not clear if the 135 millisecond time requirement was for the current licensed 100 percent power or for the extended power uprate (EPU) 100 percent power, which corresponds to 120 percent of the current licensed power. The licensee issued PER 73291 to address this discrepancy and PER 73590 to resolve the question concerning applicability of the 135 millisecond time requirement. Corrective action documents were issued to document the minor errors which were discovered during the inspectors review of DCN 51087.

b.2 <u>DCN 51222, Reactor Building - Electrical RHR, System 74</u>

The inspectors reviewed the Unit 1 permanent plant modification DCN 51222. The intent of this DCN is to replace, reroute, and/or splice power and control cables, and rework electrical cable raceways associated with various safety related valve motor operators and other RHR components. The replacement and splicing will be with Class 1E/Environmentally Qualified cables and splices to satisfy 10 CFR 50.49 requirements and rerouting and rework of cable raceways will be to satisfy the requirements 10 CFR 50, Appendix R. The replacement of the valve motor operators and other RHR components will be through other DCNs, such as DCN 51199, Reactor Building - Mechanical RHR, System 74. This DCN replaces 30 RHR valves with new motor operators to meet the requirements of the NRC Generic Letter 89-10, 10 CFR 50.49, and other industry issue upgrades. The DCN will install interlocks between valve 1-FCV-74-57 and valves 1-FCV-74-2 and 13, RHR Loop I, as well as valve 1-FCV-74-71 and valves 1-FCV-74-25 and 36, RHR Loop II. These interlocks will prevent the inadvertent draining of the reactor vessel to the suppression pool. The DCN also disconnects and abandons existing cables in 480 VAC RMOV Boards 1D and 1E and installs new cables to 480 VAC RMOV Boards 1A and 1B for respective MOVs. RMOV Boards 1D and 1E will be abandoned in place by other DCNs.

b.3 <u>DCN 51223, Reactor Building - Electrical, Core Spray, System 75</u>

The inspectors reviewed the Unit 1 permanent plant modification DCN 51223. The intent of this DCN was to implement changes to meet the requirements of NRC Generic

Letter (GL) 89-10, 19CFR50.49 Environmental Qualification (EQ), design configuration for divisional separation, and GL 95-07. The DCN includes four stages. Stage 1 covers components within System 75 Division I. Stage 2 covers components within System 75 Division II. Stage 3 addresses the deletion of cables, local control stations, and raceways associated with the two System 75 drain pumps. The drain pumps were scheduled to be removed under DCN 51200, Reactor Building - Mechanical, Core Spray, System 75. Stage 4 addresses the Master Equipment List (MEL) for all components that did not require any physical work. Changes included determination and abandoning cables associated with the testable check valves for both divisions 1-FCV-75-26 and 54; remove, replace, and re-route, with EQ cables, the power to the four, System 75 pump suction valves 1-FCV-75-2, 11, 30, and 39 (two per division); replace the contact blocks and internal wiring with EQ blocks and EQ wiring in the junction boxes associated with the control switches located in Panel 25-1; remove, replace, and re-route, with EQ cables, the power to the two, one per division, System 75 minimum flow insolation valves 1-FCV-75-9 and 37; and the removal, replacement, and re-routing, with EQ cables, was also addressed for additional System 75 valves and switches as appropriate. The DCN also addressed cabling for the System 75 instrumentation located in both divisions, such as: Pressure switches 1-PS-75-7, 16, 35, 42, and 52; pressure transmitters 1-PT-75-20 and 48; flow transmitters 1-FT-21 and 49; and flow switches 1-FS-75-21 and 49. The instrumentation and the valve motor operators were scheduled to be replaced by other DCNs such as DCN 51238, Reactor Building - Instrumentation, Core Spray, System 75.

b.4 <u>DCN 51194, Reactor Building - Mechanical, Reactor Water Cleanup (RWCU), System 69</u>

The inspectors reviewed the Unit 1 permanent plant modification DCN 51194. The intent of this DCN was to mitigate the effects of Intergranular Stress Corrosion Cracking (IGSCC) by upgrading the RWCU piping to a non IGSCC susceptible piping; improve pump availability by providing a rerouting of the cold leg suction piping; replace the Regenerative Heat Exchangers and the piping around the new heat exchangers as necessary; replace the small bore vent, drain, and test connection piping branching of the new RWCU piping and the valves located on these connections; replace the instrument lines branching of the new RWCU piping up to the root shutoff valves, a separate DCN 51235, RWCU, System 69, and Water Quality, System 43, replaces instrumentation tubing and installs new instrumentation; replace the flow elements located on the RWCU pump discharge piping; replace thermo wells and temperature elements as necessary; and redesign the RWCU piping support configuration to reflect the new piping routes and stress analysis.

b.5 <u>DCN 51211, Reactor Building - Electrical, Main Steam, System 01</u>

The inspectors reviewed the Unit 1 permanent plant modification DCN 51211. The intent of this DCN was to implement the requirements 10CFR50.49, Equipment Qualification (EQ), Design Criteria BFN-50-728, Physical Independence of Electrical System, and Design Criteria BFN-50-7001, Fire Protection. Equipment and components affected by this DCN included the replacement of the safety related limit switches for the outboard Main Steam Isolation Valves (MSIV) with qualified limit switches; replacement of electrical cables for the inboard and outboard MSIVs with qualified cables, Raychem splices, and EGS type quick disconnects; replacement of MSIVs Reactor Protective System (RPS) position indicating switch cables with qualified cables and Raychem splices; replacement and rerouting of electrical cables for the MSIV drain interlock to meet the cable separation requirements defined in Design Criteria BFN-50-728; replacement of electrical cables for the inboard and outboard Main Steam Isolation Drain Valves (MSDIV) with qualified cables and Raychem splices; replacement and rerouting electrical cables in conjunction with DCN 51221, High Pressure Coolant Injection System (HPCI) - Electrical, System 73, to separate the Automatic Depressurization System (ADS) and HPCI cables per Design Criteria BFN-50-728; and replace and reroute additional electrical cables for other equipment and components as required to meet applicable design criteria.

c. Conclusions

The inspector's review of Unit 1 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. Corrective action documents were issued to document minor errors which were discovered during the inspectors review of DCN 51087.

E1.2 Implementation of Permanent Plant Modifications (71111.17, 37550, 37551)

a. Inspection Scope

The inspectors reviewed and observed permanent plant modifications for the Unit 1 common accident signal logic, change out of transformers TS1A and TS1B, the changes to control room panel 1-9-4, fuel pool cooling system, and the replacement of the Unit 1 main transformers. The inspectors evaluated the adequacy of the modifications and observed field work to verify the design basis, licensing basis, and TS required systems had not been degraded as a result of the modifications.

b. Observations and Findings

b.1. DCN 51016, Unit 1 Emergency Core Cooling System (ECCS) Accident Signal Logic

The inspectors continued to review ongoing work activities associated with the Unit 1 permanent plant modification DCN 51016. The intent of this DCN is to restore the Unit 1 common accident signal logic for Unit 1 recovery. The inspectors observed completed activities involving the lifting of electrical cable conductors in panels 1-9-32 and 33, located in the auxiliary instrument room, and in panel 1-9-3, located in the control room. The activity was documented on work order, WO 02-011715-13. The inspectors observed that during this activity degraded conductors were discovered on electrical cables 1ES9551 and 1ES0571. The insulation on the conductors had become brittle and were cracked. The licensee issued PER 69626 to document the corrective action for repair of the degraded conductors. The work order was revised to use nuclear grade Raychem heat shrink tubing on the conductors. The repairs were made to a total of six conductors using nine inch lengths of heat shrink for each conductor. The repairs were performed in accordance with procedure MAI 3.3, Cable Terminating and Splicing for Cables Rated Up to 15,000 Volts.

b.2. DCN 51216, Stage 1, Replace Transformer TS1A, System 57-4

The inspectors reviewed the Unit 1 permanent plant modification DCN 51216. The intent of this DCN is to replace existing 4160V to 480V PCB oil filled 750KVA transformer with a Class 1E, dry type, 1000/1333KVA transformer to support increased capacity for Unit 1 restart. The transformer supplies normal power to the safety related 480V Shutdown Board 1A. Activities observed by the inspectors included placement of cement for support of the transformer per WO 03-01001-04, removal and replacement of 4KV current transformers and meter per WO 02-09460-23, and final terminations of 4KV cables per WO 03-01001-11. Portions of an additional DCN were implemented in support of DCN 51216. DCN 51087 replaced the 4160V side current transformers with higher turns ratio transformers, replaced the AC ammeter to compensate for the higher turns ratio, and changed applicable relay settings.

b.3. DCN 51470, Unit 1 Main Bank Transformers, System 57-1

The inspectors reviewed the Unit 1 permanent plant modification DCN 51470. The intent of this DCN is to replace existing Unit 1 24 KV to 500 KV Main Bank Transformers 1A, 1B, and 1C, with new higher kilo volt-ampere (KVA) rated transformers. This modification was part of the licensee's Extended Power Uprate (EPU) program. Activities observed included temporary and final placement of the transformers, installation of the fire protection and auxiliary equipment, and testing of the final installation.

b.4. DCN 51095, Control Room Design Review (CRDR) Panel 1-9-4, System 78

The inspectors reviewed the Unit 1 permanent plant modification DCN 51095. The intent of this DCN is to change the existing hand switches on Control Room panel 1-9-4 for the fuel pool cooling system. This DCN in conjunction with DCN 51203, Spent Fool Pool Cooling - Reactor Building, System 78, modified system 78 to reflect the as-built configuration of Units 2 and 3. Activities observed included installation of panel 1-25-16 and the replacement of valve and pump control panel switches.

b.5. DCN 51216, Stage 2, Replace Transformer TS1B, System 57-4

The inspectors reviewed the Unit 1 permanent plant modification DCN 51216. The intent of this DCN is to replace existing 4160V to 480V PCB oil filled 750KVA transformer with a Class 1E, dry type, 1000/1333KVA transformer to support increased capacity for Unit 1 restart. The transformer supplies normal power to the safety related 480V Shutdown Board 1B. Among the activities observed by the inspectors were the following: Remove existing transformer per WO 03-01001-15, remove and replace 4KV current transformers and meter per WO 02-09460-25, pull new cable per WO 03-01001-20 and test transformer per WO 03-01001-74. Portions of an additional DCN were implemented in support of DCN 51216. DCN 51087, Stage 14, replaced the 4160V side current transformers with higher turns ratio transformers, replaced the AC ammeter to compensate for the higher turns ratio, and changed applicable relay settings. During the cable pulling activities, licensee personnel adequately monitored for bend radius.

c. Conclusions

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

E1.3 Unit 1 Restart Special Program Activities - Cable Installation and Cable Separation (37550)

a. Inspection Scope

The programs for investigating and resolving the issues of cable installation and cable separation are described in TVA's letter to the NRC dated May 10, 1991. This letter describes programs as essentially the same as described in the Browns Ferry Nuclear Performance Plan which outlined the corrective actions to be implemented before restart of Unit 2, and repeated for restart of Unit 3. NRC Inspection Manual MC 2509, Browns Ferry Unit 1 Restart Project Inspection Program, acknowledges the special programs utilized on Units 2 and 3 as sufficient to address corresponding issues on Unit 1 if implemented in the same manner.

This inspection focused on the following sub-programs within the cable installation special program:

- Sidewall pressure
- Pullbys
- Supports for cables in vertical conduit
- Pulling cable through a 90E condulet or mid-run flexible conduit.

Inspection of the sidewall pressure sub-program consisted of reviewing a sample of the licensee's pulling tension calculations related to the Unit 1 restart as well as making independent calculations using published industry guidelines in order to compare results. The review included verification of all input data including a walkdown to verify the configuration sketches on one cable.

Inspection of the pullby and supports for cables in vertical conduit sub-programs consisted of reviewing the licensee's methodology, results and dispositions. A sample of dispositions were confirmed by reviewing the appropriate documentation. For example, the inspectors confirmed that an approved modification package included provisions to replace a particular cable before restart.

Inspection of the pulling cable through a 90E condulet or mid-run flexible conduit sub-program consisted of a detailed review of the licensee's methodology and a walkdown inspection of the Unit 1 control complex to look for examples of this issue.

Inspection of the cable separation special program consisted of reviewing the details of the issue and the relevant design criteria. Also, the licensee was requested to demonstrate that the computerized cable and raceway system blocked cables from being routed in a raceway that would violate the design criteria for separation. This was done by use of test data specified by the inspector. The inspector witnessed entering the test data at the computer terminal.

b. Observations and Findings

The licensee's analysis of the sidewall pressure issue consisted of performing a pulling tension calculation on certain originally installed cables. A very limited number of originally installed cables required a pulling tension calculation. This was primarily because many cables are being replaced prior to restart and many power distribution cables had already been addressed under the Units 2 & 3 restart programs. These calculations were performed using special computer software, and review of the software users manual showed that established formulas were incorporated. The results indicated there was a comfortable margin between calculated expected tensions and limits. The inspectors determined that coefficients of friction used were conservative and that weight correction factors were correctly applied. In a few cases, the calculations contained errors that resulted in overestimating the expected pulling tension. While these errors were of minor significance, the licensee initiated PER 71496 to address this issue.

In general, reviews of documentation of the sub-programs mentioned in the scope section showed a careful and consistent implementation. Walkdown inspections by the inspectors did not identify any examples of conditions that should have been documented but were not.

The cable separation special program is intended to address the problem where cables having a certain separation code were routed in raceways having a separation code that was not compatible. This led to cable separation criteria problems where cables of different separation code were routed and installed in the same raceway. A contributing factor to this problem was the original design utilized hand written cable and raceway schedules, which did not provide any automatic check or block of incorrect routing. Subsequently, the cable and raceway schedule was transferred to a computer based system having the separation code checks of cable routing. As such, the inspectors reviewed the computer separation code check as described in the scope section, and the proper checks were found to be in place.

c. Conclusions

Inspection of the methodology for the Cable Installation and Cable Separation special programs together with performance based inspection led to the inspector's conclusion that implementation of these programs is proceeding in accordance with licensee commitments.

E1.4 Unit 1 Restart Special Program Activities - Flexible Conduit (37550)

a. Inspection Scope

The Special Program associated with the installation of flexible conduit is designed to ensure that flexible conduits are installed in a manner that will allow differential movement that would result from seismic events or anticipated pipe movement without damage to the conduit or installed cables. Specifics for this program are defined in Supplemental Safety Evaluation Reports transmitted on March 19, 1993, and October 3, 1995, which applied to all three units at Browns Ferry. The essential elements of the Special Program for flexible conduits are development of installation criteria and methodology, documented walkdown inspections of installed flexible conduit, and correction of any conduits not meeting the criteria.

b. Observations and Findings

The flexible conduit Special Program was documented in Calculation No. EDQ1 999 2003 0014, Analysis of Flex Conduit to Devices for Unit 1, Rev. 1.

This inspection consisted of a walkdown of all areas in the Unit 1 control complex focusing on flexible conduit installation. The attributes of bending radius, length, tightness of fittings and ground wire were observed. During the walkdown, the inspector observed that, in the southwest corner of the control room, one flexible conduit had a loose connection (could easily be turned by hand) and the ground wire at one flexible conduit was not attached at one end. The licensee initiated PER 72394 to address the inspectors observations. Otherwise, the inspectors did not identify any discrepancies with the Special Program methodology or implementation.

c. Conclusions

Inspection of the methodology for the Flexible Conduit special program and inspection of installed flexible conduit in the control complex led to the conclusion that implementation of this program is proceeding in accordance with licensee commitments.

E1.5 Unit 1 Restart Special Program Activities - Intergranular Stress Corrosion Cracking (IGSCC) - Welding of Replacement RECIRC System, RHR System, RWCU, and CS System Piping (55050)

a. Inspection Scope

As part of the IGSCC special program, TVA is replacing the RECIRC system piping and portions of the RHR and RWCU piping systems with Type 316NG material. In addition, the CS System stainless steel piping is being upgraded by replacing it with carbon steel. The applicable Codes for this work are: (1) Code for Power Piping USAS B31.1, 1967 Edition, (2) ASME Section XI, 1995 Edition, 1996 Addenda, and (3) ASME Section III, 1995 Edition, 1996 Addenda.

The inspectors observed completed and in-process welds and reviewed completed weld records, procedures, personnel qualification records, and material certification records, as detailed below to verify compliance with applicable requirements.

b. Observations and Findings

Inspection Reports 50-259/2003-009, 50-259/2003-010, 50-259/2003-011, 50-259/2004-006 and 50-259/2004-007 documented previous inspections in this area. During the current inspection, the inspectors observed/reviewed the following to verify compliance with the applicable Codes listed above:

- The inspectors visually inspected final weld surfaces (outside diameter (OD) only) for the following RECIRC and CS System Welds:

RWR-1-001-047A	CS-1-002-033A
RWR-1-001-047	CS-1-W002-012
RWR-1-002-038	CS-1-W002-011
RWR-1-002-045	CS-1-002-032
RWR-1-001-046	CS-1-W002-017
RWR-1-002-017	CS-1-002-008

  In addition, RECIRC System Weld RWR-1-002-032 was inspected during grinding to prepare the final surface for preservice inspection (PSI).

- For RHR System Welds RHR-1-014-013 and RHR-1-012-001, the inspectors observed in-process welding and reviewed in-process Weld Data Sheets. For Weld RHR-1-013-001, the inspectors observed the final fitup after tacking.

- For the following RECIRC, CS, RHR, and RWCU System welds, the inspectors reviewed the completed Weld Data Sheets and associated Welding Procedures, nondestructive examination (NDE) and Quality Control (QC) personnel qualification records, welding material certification records, welder qualification records, and NDE examination (visual (VT), liquid penetrant (PT), magnetic particle (MT) as applicable) records, as detailed below.

RWR-1-001-047	CS-1-002-012	RWCU-1-001-019
RWR-1-001-046	CS-1-002-010	RWCU-1-003-023
RWR-1-002-045	CS-1-002-011	RWCU-1-005-029
RWR-1-001-047A	CS-1-002-027	RHR-1-019-001
RWR-1-002-038	CS-1-002-019	
RHR-1-019-002	RHR-1-013-002	

- The inspectors reviewed the radiographic (RT) film for the following welds:

RWR-1-002-032	RWCU-1-005-023	CS-1-002-010
RWR-1-001-046	RWCU-1-005-025	CS-1-002-032
RWR-1-002-038	RWCU-1-009-015	CS-1-002-033A
RWR-1-001-047	RHR-1-019-001	CS-1-002-008
RWR-1-002-045	RHR-1-013-002	
RWR-1-001-047A		
RWR-1-002-017		

- Certified Material Test Reports (CMTRs) were reviewed for the following heats/lots of welding material: 0.035 ER308/308L Spooled Wire - Lot XF7995; 0.035 ER309/309L Spooled Wire - Lot XM7860; 1/8" ER70S-6 Bare Wire - Lot D-8145; 1/8" ER70S-6 Bare Wire - Heat 286390; 3/32" ER70S-3 Bare Wire - Heat C208240; 0.035" ER70S-6 Spooled Wire Heat 186921; and 3/32" ER309L Bare Wire - Lot CM6603.

  CMTRs for other welding materials being used were previously reviewed as documented in NRC Inspection 50-259/2004-007.

- Detailed Welding Procedure Specifications (DWPSs) GTA88-C-2-N, Revision 1, and GT-O-1-N, Revision 5, including applicable Procedure Qualification Records (PQRs), were reviewed.

- The inspectors reviewed Welder Qualification Records, including continuity records, as applicable, for 18 welders who welded on the above listed RECIRC, RHR, CS, and RHR System.

- Qualification Records for 14 Level II MT Examiners, 15 Level II PT Examiners, and 17 Welding - VT Examiners, two Level III RT Examiners, and one Level II RT Examiner, who performed QC and NDE of the RECIRC, RHR, CS, and RWCU System welds listed above, were reviewed.

- For the completed PT examination reports reviewed, the inspectors reviewed certification records for the following PT materials:

Remover - Lot 04A02K
Developer - Lots 03A03K and 03K02K

Certifications for other PT materials identified on the reviewed PT reports were reviewed during the inspection documented in NRC Inspection Report 50-259/2004-007.

- In addition to observation of in-process welding activities and review of records, the inspectors reviewed welding control and nondestructive examination procedures as listed in the List of Documents Reviewed in the Supplemental Information attached to this report.

c. Conclusions

The licensee's intergranular stress corrosion cracking (IGSCC) mitigation plan continued to meet commitments established by licensee Regulatory Framework letters. Recirculation (RECIRC) System, Residual Heat Removal (RHR) System, Reactor Water Cleanup (RWCU) System, and Core Spray (CS) System piping replacement activities were meeting ASME Code and other regulatory requirements. No violations or deviations were identified.

E1.6 Inservice Inspection (73753), and Inservice Inspection Data Review and Evaluation (73755)

a. Inspection Scope

The inspectors reviewed the Browns Ferry Unit 1 Inservice/Preservice Inspection (ISI/PSI) activities as detailed below to verify compliance with ISI/PSI requirements in accordance with regulatory requirements and licensee commitments. See Inspection Report 50-259/2004-007 for a previous inspection 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 program are: (1) ASME Section XI, 1995 Edition, 1996 Addenda, and (2) ASME Section XI 1974 Edition with Addenda through Summer 1975. For PSI of replaced components, the applicable Code is ASME Section XI, 1995 Edition, 1996 Addenda.

b. Observations and Findings

Repaired or replaced welds and components will receive a PSI in accordance with the requirements of ASME Section XI prior to returning the system to service. The PSI examination of repaired or replaced welds susceptible to IGSCC will not be conducted until after a Mechanical Stress Improvement Process has been performed.

Observation/Review of ISI/PSI Activities

The inspectors observed/reviewed a sample of PSI examination activities as detailed below. The observations and records were compared to the Technical Specifications (TS) and the applicable Code (ASME Boiler and Pressure Vessel Code, Section XI, 1995 Edition with Addenda through 1996) to verify compliance. The examinations observed/reviewed are listed below:

- Observed UT examination and post-examination calibration check for CS System Weld CS-1-S002-018
- Visually inspected CS System Welds CS-1-W002-017, CS-1-W002-012, and CS-1-W002-011
- Reviewed NDE examination reports for completed UT examinations of Welds: CS-1-002-014 and CS-1-S002-028

For Welds CS-1-S002-018, CS-1-002-014 and CS-1-S002-028, the inspectors reviewed qualification/certification records for the Level 2 UT examiners, examination equipment and examination consumables.

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.

E1.7 Unit 1 Restart Special Program Activities - Containment Coatings - Repairs to Coatings in Unit 1 Torus (37550)

a. Inspection Scope

The inspectors reviewed the licensee's program to identify unqualified coatings in the Unit 1 torus, the planned corrective actions to remove unqualified coatings, and the program for application of new Service Level I coatings qualified to ANSI standards. The inspectors also reviewed the licensee's program for procurement of coating materials, the testing program performed to qualify the coating materials, the licensee's painter qualification program, and surface preparation prior to application of new coatings.

b. Findings and Observations

The inspectors reviewed TVA Specification G-55, Technical and Programmatic Requirements for the Protective Coatings for TVA Nuclear Plants, Revision 13, to verify technical requirements were specified for application of the new Service Level 1 coatings, including qualification of the materials, qualifications and testing requirements for the coating applicators, environmental conditions during coating application, surface preparation, coating application, curing, inspection, and repairs and touch-up. The inspectors also reviewed specification change notices, SRN-G-55-26 and SRN-G-55-27.

The inspectors also reviewed TVA procedure MAI-5.3, Protective Coatings, which specifies acceptance criteria for coatings, the quality control inspection requirements, and requirements for records for documentation of completed coatings to verify the licensee's program for inspection of completed coating complied with NRC requirements. The inspectors reviewed the qualification records of the QC inspectors, samples of completed Service Level I coatings' QC inspection records, and Problem Evaluation Reports (PERs) which documented and evaluated deficiencies in the coatings' program.

The inspectors reviewed the licensee's program for procurement of protective coatings. The program requirements are specified in TVA procedure SPP-4.1, Procurement of Material, Labor, and Services, and TVA procedure NEDP-8, Technical Evaluations for Procurement of Materials and Services. The licensee's quality assurance program, implemented by these procedures, requires the performance of technical evaluations to establish quality requirements for the specification and procurement of materials for use in safety related activities. The inspectors reviewed a purchase order for procurement of the coating system designated as SL-I-109 in Specification G-55, Keeler and Long Kolor Poxy 6548/7107.

The inspectors examined the paint shop and the controls for issuing the coating materials to ensure shelf life was not exceeded prior to application of the materials. The inspectors also reviewed the qualification records of selected applicators and the testing program (including the test panels) used to qualify the coating applicators for System SL-I-109. The test panels used to qualify applicators for spray application of the coatings comply with those specified in ASTM D4228, Standard Practice for Qualification of Journeyman Painters for Application of Coatings to Steel Surfaces of Safety-related Areas in Nuclear Facilities. An alternate method is utilized by the licensee to qualify applicators for use of brushes. This method is for limited qualifications, as specified in ASTM D4228.

The inspectors determined that the original coatings in the Unit 1 torus consisted of two different systems. One system, proprietary name Plasite, was used to paint the vapor zone, which is the area above the torus high water line. Plasite is no longer manufactured. In the immersion zone, which is the area below the high water line, the torus was originally painted with three coats of a product called Valspar 78. Valspar 78 is currently manufactured under the brand name Vyguard 78, which is equivalent to Valspar 78. The design basis accident (DBA) testing performed on the Valspar 78 to qualify this system for immersion service required that the Valspar 78 be heat cured by heating the torus steel to between 200 and 250E F for a period of 48 hours after application of the third and final coat of paint. The DBA test results were also used to establish surface preparation requirements, number of coats, curing time between coats, dry film thickness (DFT) of each coat and DFT of the finished coating. The requirements for application and curing of the Valspar 78 are specified in Appendix D of G-55.

The inspectors determined that the licensee's program for restoration of the coatings in the torus for the Unit 1 recovery program was originally based on repair of defects in the existing damaged coatings. After the torus was drained, and prior to start of the Unit 1 recovery coatings activities, inspections were performed by engineers and quality control personnel to identify defects in the coatings. The inspections identified damaged (defective) coatings, areas where coatings were applied to stainless steel components, and hardware with vendor applied coatings which were considered uncontrolled.
The defects were documented in the inspection records attached to the inspection procedures. Additional surveys/inspections were then performed by various individuals, including craftsman, inspectors, and engineers, to identify other defects which were recorded on maps. Several problem evaluation reports (PERs) were initiated as a result of inconsistent criteria and questions on inspection methods. However, when the extent of the defective/damaged coatings became apparent, licensee management made the decision to remove all the coatings in the immersion zone by sandblasting and to re-paint the area. Licensee engineers initially proposed to use Vyguard 78 (System SL-I-108) as the coating system in the restored immersion zone with curing at 70E F for 14 days. DBA testing was performed on the Vyguard 78 with the revised curing method. The test results showed the application of Vyguard 78 with curing at 70E F for 14 days did not perform well in the immersion zone, but was acceptable for use in the vapor zone. The licensee then selected a coating system, Keeler & Long Kolor Poxy 6548/7107, for use in the immersion zone. One advantage of this system was that it did not require heat curing. The Keeler & Long 6548/7107 is designated as Coating System SL-I-109 in TVA Specification G-55. The inspectors reviewed the results of DBA testing performed on the Keeler & Long 6548/7107. The test results, documented in a report dated 3/2/81, showed the materials are qualified for immersion service. The inspectors compared the criteria used in the testing program for application of the coating materials to the test plates to verify that the criteria specified in G-55 for coating application duplicated test conditions. These criteria included surface preparation, thickness and number of coats, curing time between coats, and final curing time.

The licensee had no test data to demonstrate that the Keeler & Long 6548/7107 can be applied over the Plasite at the interface between the two coating systems. Since the Plasite is no longer manufactured, it was not possible to perform a test to demonstrate the qualification of the application of 6548/7107 over Plasite. PER 04-046609 was initiated to document that the Keeler & Long 6548/7107 coating were incorrectly used to repair the Plasite coatings in the Bay 1 vapor space. The inspectors reviewed the PER. The licensee has DBA test results which show that Vyguard 78 is qualified to apply over the Plasite, and that the Keeler & Long 6548/7107 can be applied over Vyguard 78. The inspectors reviewed the results of these DBA tests. Based on the test data, the licensee concluded that it was necessary to perform the application of the new protective coatings by removing via sand blasting existing coatings from a strip one foot wide above the immersion zone. The Vyguard 78 coatings were applied to this area to serve as a transition zone between the Plasite and Keeler & Long 6548/7107 coatings. After completion of the application and curing of the Vyguard coatings in the transition zone, the existing coatings (Valspar 78) in the immersion zone were removed by sandblasting. Repairs to existing damaged Plasite coatings in the vapor zone were performed using Vyguard 78. The inspectors determined that these actions were appropriate to fulfill DBA criteria and resolve the Unit 1 torus material condition.

The inspectors examined the completed surface preparation for application of the Vyguard 78 coatings in the transition zone. The inspection was performed while the licensee's final quality control acceptance inspection was in process. The inspectors examined portions of the torus shell, inside and outside of four downcomer pipes, portions of supports in the transition zone, and other hardware in the transition zone. The inspectors independently examined the surface profile by use of a Keane-Tator surface profile comparator.

c. Conclusions

The licensee's program for restoration of the coatings in the Unit 1 torus was determined to comply with NRC requirements. Repairs to identified coating problems were completed by qualified individuals and accomplished in accordance with approved procedures. No violations or deviations were identified.

E1.8 Unit 1 Restart Special Program Activities - Cable Ampacity (37550)

a. Inspection Scope

The inspectors reviewed two DCNs, that the licensee prepared for resolving cable ampacity issues, in order to evaluate the technical adequacy of the plant modifications, and, specifically, the adequacy of the cable ampacities. The inspectors also performed field inspections of one of the DCN packages in order to verify that the installed cable was consistent with the design change requirements specified in the DCN package.

b. Findings and Observations

The inspectors reviewed two DCN packages that the licensee had prepared for resolving cable ampacity issues involving 5000 volts type V5 power cables outside the drywell. DCN 51222, included as part of its scope, the replacement of cable number ES2625-II because of its failure to meet the cable tray derating criteria for ampacity. The power cable was also to be changed because of its failure to satisfy 10 CFR 50.49 Environmental Qualifications requirements. The design change involved the procurement and installation of new cables in conduits from 4160 V shutdown board C located in the reactor building at elevation 621' to RHR pump 1B located in the Reactor building at elevation 519'.

The inspectors determined that post issue change (PIC) number 62222 was written by the licensee to revise the scope of DCN No. 51222, and retain the as-built installation for cable ES2625-III to support the re-start of Unit 1. The installed cable is a 400 MCM, 5000 Volt, 3-1/conductor cable required for operation of RHR pump 1B. The inspectors reviewed PIC No. 62222 and verified that the licensee had obtained walkdown data for the existing cable installation which demonstrated that the cable test data and as-built installation satisfied the requirements of 10 CFR 50.49. The PIC, however, failed to address the cable ampacity concerns that were initially identified in DCN 51222. A resolution of this concern would not have been provided by using the as-built cable installation to support restart of Unit 1. The inspectors brought this omission to the licensee's attention and PER Number 72276 was written by the licensee to initiate corrective action for this omission.

The licensee revised calculation EDQ1-999-2002-0023, in response to the cable ampacity omission discussed above, and performed an analysis for portions of the cable route where the cable was installed in conduit. The inspectors reviewed this calculation and verified that the analysis was consistent with the guidance of electrical design standard DS-E12.6.3, and that the analysis demonstrated that the cable had adequate ampacity for conduit installation. For those portions of the cable route where the cable was installed in cable trays, the licensee took credit for the phase IIa analysis that was performed during Units 2 and 3 restart. The methodology used in the phase IIa cable ampacity analysis was previously accepted by the NRC for Unit 2 in Safety Evaluation Report, NUREG -1232, Volume 3, Supplement 2.

The inspectors reviewed cable ampacity calculation ED-Q0000-870135 that was prepared in 1992 to evaluate the ampacity of safety related cables routed in V4 and V5 trays that were required for three unit operation. The inspectors specifically evaluated the analysis for cable ES2625-II routed in cable tray segments 2AQ-ES11 and 2BK-II. Based on this review the inspectors determined that the cable failed the phase I calculation, but was demonstrated to have adequate ampacity upon performance of the phase IIa calculation. The methodology used to demonstrate the adequacy of the cable ampacity was also determined to be consistent with that accepted by the NRC in NUREG -1232, Volume 3, Supplement 2.

The inspectors also reviewed DCN 51216 which installed a 5000 Volt power feeder cable from the primary side of transformer TS1A to 4KV Shutdown board 1A. The review included an evaluation of the ampacity calculation for Shutdown Board 1A power feeder cable 1PP9857b 1A. The inspectors verified that the cable ampacity analysis was consistent with the guidelines delineated in design standard DS-E12.63. Additionally, the inspectors evaluated the overall technical adequacy of the cable installation by reviewing the voltage drop calculations, the short circuit calculation, and other supporting engineering documents used to specify and procure the cable. The inspectors also performed a field inspection of the installed cable and verified that the as-built cable installation was installed in accordance with the requirements of the DCN.

c. Conclusion

The inspectors concluded that the DCNs implemented as corrective actions to resolve cable ampacity concerns involving cables ES2625-II and 1PP9857B are acceptable to support Unit 1 restart. The licensee's use of the as-built installation for cable ES2625-II required for RHR pump 1B motor feeder, in lieu of procuring and installing new cables, was considered acceptable based on the cable ampacity analysis and NUREG -1232. The inspectors also concluded that the cable ampacity of new power feeder cable 1PP9857B, installed for shutdown board 1A, was adequate.

E1.9 Control Room Design Review CRDR Program (37551)

a. Inspection Scope

The inspectors performed a review of the Control Room Design Review (CRDR) Program. This review also included a review of portions of the Human Engineering Deficiencies (HED) Program which is covered under the CRDR Program. The majority of the HEDs involved the control room panels.

b. Observations and Findings

The CRDR Program and the HED program plan for Unit 1 are similar to the CRDR and the HED programs for Units 2 and 3. Differences are mostly due to obsolete equipment since the Unit 2 and Unit 3 recoveries. The CRDR program consisted of a number of DCNs, each of which was assigned to a specific Control Room Panel. The individual HEDs for Unit 1 were assigned to specific DCNs. Within each specific DCN, a number of HEDs were documented, the item was described, and the corrective action was also documented. The corrective actions included operators, escutcheons, handles, and indicating components. DCNs and assigned HEDs reviewed by the inspectors included the following:

- DCN 51094, CRDR, Implement Modifications to Control Room Panel 1-9-3 to Resolve Identified HEDs. Stage 1 implemented the specific HEDs for System 1 - Main Steam, System 3 - Reactor Feedwater, and System 43 - Sampling and Water Quality; Stage 2 implemented the specific HEDs for System 64 - Primary Containment and System 76 - Containment Indenting; Stage 10 implemented the specific HEDs for System 73 - High Pressure Coolant Injection; Stage 12 implemented the specific HEDs for System 74 - Residual Heat Removal; and Stages 17 through 28 implemented the specific HEDs for the 12 RHRSW pumps. Each Stage listed the interfacing DCNs that also affected the systems Changes to Panel 1-9-3 to implement the specific HEDs included rearrangement and relocation of control switches and instruments to provide improved operator interfaces and to support switch and equipment movement from other control room panels; installation of improved component labeling and switch position escutcheons; replacement of hand switch handles with black shape coded handles; replacement of indicating meters and associated scales to meet system functional requirements; replacement of panel indicating lights in accordance with systems engineering and human factors requirements; replacement of specific obsolete recorders, controllers, and power supplies; installation of equipment associated with the Hardened Wet-well Vent modification; installation of new system mimics and general panel surface enhancements; and support activities associated with annunciator DCN 51107, Unit 1 common accident signal DCN 51016, and Unit 2 common accident signal DCN 51018.

- DCN 51096, CRDR, Implement Modifications to Control Room Panel 1-9-5 to Resolve Identified HEDs. Stage 1 implemented the specific HEDs for System 4 - Hydrogen Water Chemistry, System 64 - Primary Containment Isolation, and System 261 - Computer and Computer Trend Recorders 1-9-41-37A and 37B; Stage 2 implemented the specific HEDs for System 1 - Main Steam, System 3 - Reactor Feedwater, and System 46 - Reactor Feedwater Controls; Stage 3 implemented the specific HEDs for System 57 - Electrical Distribution and System 98 Reactor Water Recirculation Flow Control; Stage 4 implemented the specific HEDs for System 63 - Standby Liquid control; and Stages 5 through 10 implemented the specific HEDs for the such systems as System 68 - Reactor Water Recirculation, System 85 - Control Rod Drive, System 92 - Neutron Monitoring, and System 99 - Reactor Protection. Each Stage listed the interfacing DCNs that also affected the systems. The changes to Panel 1-9-5 to implemented the specific HEDs were similar to activities associated with Panel 1-9-3.

- DCN 51111, CRDR, Implement Modifications to Control Room Panel 1-9-6 to Resolve Identified HEDs. Stage 1 implemented the specific HEDs for System 66 - Off-Gas, System 90 - Radiation Monitoring, System 47 - Turbine/ Generator Electro-Hydraulic Control, System 55 - Control Room Annunciators, and System 57 - Electrical Distribution; Stage 2 implemented the specific HEDs for System 35 - Main Generator Hydrogen and Stator Cooling, System 43 - Sampling and Water Quality, System 57 - Electrical Distribution, and System 242 - 500 KV Switchyard; Stage 3 implemented the specific HEDs for System 82 - Standby Diesel Generators; and the remainder of the stages were mainly for the various 4160 VAC and 480 VAC distribution systems. Each Stage listed the interfacing DCNs that also affected the systems. The changes to Panel 1-9-6 to implemented the specific HEDs were similar to activities associated with Panel 1-9-3

- DCN 51098, CRDR, Implement Modifications to Control Room Panel 1-9-8 to Resolve Identified HEDs. Systems affected by this DCN included System 1 - Main Steam, System 2 - Condensate, System 3 - Reactor Feedwater, System 5 - Extraction Steam, System 6 - Feedwater Heaters and Drains, and System 46 - Reactor Feedwater Control. Each Stage of the DCN listed the interfacing DCNs that also affected the systems. The changes to Panel 1-9-8 to implemented the specific HEDs were similar to activities associated with Panel 1-9-3

The implementation of some of the surface enhancements to the control room panels to resolve identified HEDs were assigned to DCN 51109. These enhancements included painting the surface areas of the Control Room panels with a specific color paint; install handrails on the Control Room front panels 1-9-3, 1-9-4, 1-9-5, 1-9-6, 1-9-7, and 1-9-8;

and install color coded mimics, demarcation lines, special figure tags, and applicable graphics . Other DCNs issued to implement identified HEDs to Control Room back panels and panels located outside the Control Room included DCN 51102 for panel 1-9-25, Control Room back panel; DCN 51103 for panel 1-9-46 and panel 1-9-47, Control Room back panels; and DCN 51106, for panel 1-25-3, Remote Shutdown Panel, located in the Electrical Board Room 1A.

c. Conclusions

Based on review of selected modifications the inspectors concluded that the licensee's Control Room Design Review program was adequate to resolve previously identified Human Engineering Deficiencies. No violations or deviations were identified.

E1.10 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed licensee procedure SPP-9.5, Temporary Alterations. The inspectors also reviewed the following temporary alterations: TACF 1-04-010-099, U1 Reactor Protective System (RPS), Primary Containment Isolation System (PCIS), and Backup Scram Circuitry Power Supplies; TACF 1-04-011-064D, U1 PCIS Isolation Signals to Various Systems; TACF 1-04-012-074, U1 Residual Heat Removal (RHR) Isolation PCIS Over Ride; TACF 1-04-013-069, U1 Reactor Water Cleanup (RWCU) Isolation PCIS Over Ride; and TACF 1-04-014-064, Temporary Air Supply for Unit 1 Torus Sand Blasting and Coating Activities. 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 TACF. In addition, special emphasis was placed on the potential impact of this temporary modification on operability of equipment required to support operations of Units 2 and 3.

b. Observations and Findings

b.1 TACFs issued to convert Unit 1 modifications, previously implemented under plant procedures

POI-200.4, Defeating RPS/PCIS Logic, was used in the defeating of Reactor Protective System (RPS), System 99, and Primary Containment Isolation System (PCIS), System 64, logic on various Unit 1 systems to support the restart of Unit 2 and Unit 3. The conversion of the modifications was to allow the depictions of the jumpers, fuse pulls, open circuit breakers, and lifted leads in the various systems, implemented under the procedure, to be placed onto drawings. The conversion also allowed for the removal of the tags placed on the various system components, lifted leads, fuse pulls, open circuit breakers, and jumpers by plant procedure POI - 200.4, and their replacement by TACF tags. This was done in accordance with the corrective action of PER 40603.

- TACF 1-04-010-099, was initiated for open circuit breakers and pulled fuses in the RPS, PCIS, and backup scram circuitry power supplies. Systems other than RPS and PCIS, that were affected by this TACF included Main Steam, System 1; Recirculating Water Sampling, System 43; and Control Rod Drive, System 85. The breakers that were opened were Breaker 903 and Breaker 953, both located on Battery Board 1, Panel 9. The breakers supplied power to RPS Logic Channel A and RPS Logic Channel B, respectively. Breakers 903 and 953 also supplied power to PCIS Trip Channels A1, A2, B1, and B2. The fuses that were pulled were Fuse 5A-F21A and Fuse 5A-F22A, located in Panel 1-9-15, and Fuse 5A-F21B and Fuse 5A-F22B, located in panel 1-9-17. The fuses supplied power to Backup Scram System A and Backup Scram System B, respectively.

- TACF 1-04-011-064D, was initiated for various components in systems that were disabled by the PCIS logic system being de-energized. The affected components were in systems such as the following: Drywell Control Air, System 31; Primary Containment, System 64A; Secondary Containment, System 64C; Standby Gas Treatment, System 65; Core Spray, System 75; Radwaste, System 77; and Transverse Incore Probe (TIP), System 94. The TACF documented the location of jumpers installed around relay contacts to make the various components functional for operational purposes. Components affected included 1-FCV-64-29, Drywell Vent Valve; 1-FCV-64-32, Torus Vent Valve; 1-FCV-76-18, Drywell Nitrogen Inboard Isolation Valve; 1-FCV-76-19, Torus Nitrogen Inboard Isolation Valve; 1-FCV-64-30, Drywell Vent to Reactor Building Ventilation Valve; 1-FCV-64-30, Torus Vent to Reactor Building Ventilation Valve; Secondary Containment Isolation Inboard and Common Valves (DIV I) and Outboard Valves (DIV II); and Plant Refuel Zone Isolation Initiation Logic.

- TACF 1-04-012-074, was initiated for the lifted leads, jumpers, and pulled fuses in Panels 1-9-42 and 1-9-43. This was to allow for the operation of certain RHR valves with PCIS de-energized. The fuses that were pulled were 16A-F15 and 16A-F16. The lifted leads and jumpers were on relay 16A-K29, located in panel 1-9-42, and on relay 16A-K30, located in panel 1-9-43. Valves affected included 1-FCV-74-47, RHR Shutdown Cooling Outboard Isolation and 1-FCV74-48, RHR Shutdown Cooling Inboard Isolation; 1-FCV-74-53, RHR Loop I Inboard Injection and 1-FCV-74-67, RHR Loop II Inboard Injection; 1-FCV-74-77, RHR Shutdown Cooling Outboard Head Spray Isolation, which is to be disconnected and abandoned in place by DCN 51151; and 1-FCV-74-102, 103, 119, and 120, Torus Vents, which are to be deleted by DCNs 51199, Reactor Building - Mechanical, 51222, Reactor Building - Electrical, and 51094, Control Room Design Review.

- TACF 1-04-013-069, was initiated for the lifted leads, jumpers, and pulled fuses in Panels 1-9-42 and 1-9-43. This was to allow for the operation of certain RWCU valves with PCIS de-energized. The fuses that were pulled were 16A-F17 and 16A-F18. The 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. Valves affected included 1-FCV-69-1, 2, and 12, RWCU Primary Containment Isolation Valves.

The modifications that were installed under plant procedure POI-200.4 were tested in accordance with that procedure. The TACFs replaced the tags and did not change any configuration, therefore, additional testing was not necessary. The TACFs will be removed and the systems will be restored at the point in time when the RPS and PCIS can be re-energized and the affected systems ready for functional testing. The affected systems will be restored using the System Preoperability Checklist (SPOC) and System Plant Acceptance Evaluation (SPAE) process. The Restart Test Program will demonstrate proper function of reactor protective and isolation logic of the affected systems. The test program will also include controls of the affected system components. Therefore, specific restoration testing will not be required for these TACFs.

b.2 TACF issued to support current Unit 1 recovery activities

TACF 1-04-014-064, Temporary Air Supply for Unit 1 Torus Coatings Activities, was initiated to support contractor sandblasting and coatings activities in the Unit 1 Torus. The TACF used two spare 8" secondary containment penetrations, Z15651024-EL 588 ft and Z15651025-EL 586 ft, located along the south wall of the Unit 1 Reactor Building, to install three 3" air line hoses. Two new flanges, one per penetration, were installed with gaskets and bolted to the interior penetration flanges. The EL 588 ft penetration had two pipes running through and the EL 566 penetration had one pipe running through. Each pipe was welded to the new flange and had a ball valve installed, for shut off purposes, in each pipe end protruding into the Unit 1 Reactor Building. The air hoses ran from three diesel driven 1500 cubic feet per minute (CFM) air compressors, through the two penetrations, through the shut off valves, into the Unit 1 Reactor Building, and down into the Unit 1 Torus. The air hoses were rated to 400 psig working pressure. The maximum standby pressure for the compressors was 153 psig per vendor specifications. Relief valves were installed on each compressor with a set lift pressure no greater than 153 psig. Secondary Containment was not breached in that the inner flanges were first removed, the new flanges with the shutoff valves closed were installed, the outer flanges were removed, and the hoses from the air compressors were hooked up. After completion of the torus work the removal of the hoses will be in the reverse order. Specific instructions were given to personnel for the operation of the system. These instructions included that only the Control Room Operators may authorize opening of the air system, a monitor must be present when the system is open, the monitor must have means of direct communications with the control room personnel, the monitor must not leave the area without closing the shutoff valves or being properly relieved by another monitor, and the monitor must have no other duties.

c. Conclusions

The inspectors determined that one new temporary alteration to support temporary air supply for Unit 1 torus activities and four temporary alterations issued to convert Unit 1 modifications, previously implemented under existing plant procedure associated with the PCIS, RPS, RHR and RWCU systems did not cause any significant impact on the operability of equipment required to support operations of Units 2 and 3. No violations or deviations were identified.

E1.11 System Return to Service Activities (37550)

a. Inspection Scope

The inspector reviewed and observed portions of the licensee's system return to service (SRTS) activities. The SRTS activities were performed in accordance with Technical Instruction 1-TI-437, System Return to Service (SRTS) Turnover Process for Unit 1 Restart. .

b. Observations and Findings

SRTS activities increased during the reporting period. This included the completion of the SPOC II process for System 40, Station Drains which was placed in the status control category by Operations. Operations assumed total control of System 40 which was returned to Operations in a fully operable status. Additionally, SRTS activities observed by the inspectors included some system testing on System 78, Spent Fuel Pool Cooling. This review is documented in Section E1.12. The inspectors reviewed and observed the licensee's SRTS activities for the following:

- System 8, Turbine Drains and Miscellaneous Piping, including completion of the SPAE and SPOC I processes
- System 12, Auxiliary Boilers, including completion of the SPAE process and completion of the SPOC I process
- System 30, Control Bay Heating, Ventilation, and Air Conditioning, including completion of the SPAE process
- System 33, Service Air, including completion of the SPAE and SPOC I processes
- System 37, Gland Seal, including completion of the SPAE and SPOC I processes
- System 39, CO2 Storage, Fire Protection, and Generator Purge, including completion of the SPAE and SPOC I processes
- System 40, Station Drainage, including completion of the SPAE and SPOC II processes

- System 78, Spent Fuel Pool Cooling, including portions of the SPAE and SPOC I processes

- System 79, Fuel Handling, including the completion of the SPAE and SPOC I processes

Activities observed included periodic meetings to discuss the SRTS status of various systems, which included the status of the SPOC I checklists, the status of the SPAE process, and the status of the SPOC II checklists. The activities also included observation of licensee walkdowns of portions of plant systems and review of PERs initiated during the SRTS process. Specific PERs reviewed by the inspectors are listed in the report attachment. Each of these PERs were adequately addressed by the Unit 1 Restart corrective action program.

c. Conclusions

SRTS activities continued to be performed in accordance with procedural requirements. System deficiencies were identified and appropriately addressed by the licencee's corrective action program.

E1.12 System Restart Testing Program Activities (37551)

a. Inspection Scope

The inspectors reviewed and observed on-going activities associated with the Restart Test Program (RTP). Specifically, the inspectors reviewed and observed portions of the performance of Post Modification Test Instruction (PMTI), 1-PMTI-BF-078.023, Functional Testing of Fuel Pool Cooling Handswitches on Control Room Panel 1-9-4. The PMTI was associated with a portion of DCN 51095, Control Room Design Review (CRDR), Control Room Panel 1-9-4, for the resolution of identified Human Engineering Deficiencies (HED).

b. Observations and Findings

The PMTI was developed, written, approved, and issued to test portions of DCN 51095, Control Room Design Review (CRDR), Panel 1-9-4, and DCN 51203, Spent Fuel Pool Cooling - Reactor Building, System 78. The areas involved in the test were the Unit 1 Control Room, the fuel pool cooling area of the Unit 1 Reactor Building, and the Refueling Floor. A pre-test briefing was held, assignments were made, and communication was established. During the test, due to miscommunication, a valve power supply circuit breaker was opened by mistake. Section 6.1, Testing of the Fuel Pool Cooling Valve Handswitches and Indicating Lights, Step 6.1.3, Testing of Hand Switch 1-HS-78-63A for Valve 1-FCV-78-63, sub-step [4] stated, in part, if valve 1-FCV-78-63 is not opening, then open circuit breaker 1-BKR-78-63. After the circuit breaker was inadvertently opened, Step 6.1.3 had to be repeated. The licensee identified a total of six Test Deficiencies (TD) associated with this activity and WO 04-017152-01 was initiated to trouble shoot and repair the disagreement white light for the 1A pump. The test personnel also initiated PER 72436 to address the conduct of the test and the deficiencies.

c. Conclusions

Overall implementation of restart testing activities for the Fuel Pool Cooling System was acceptable. Test deficiencies identified during performance of testing were adequately documented under the licensee's corrective action program.

E7 Quality Assurance in Engineering Activities

E7.1 Licensee Quality Assurance Oversight of Recovery Activities (Identification and Resolution of Problems) (71152)

a. Inspection Scope

The inspectors continued to evaluate the adequacy of the licensee's corrective actions to address various documented deficiencies indicating problems with the licensee's welding program. The inspectors reviewed selected PERs and observed field activities. In addition, the inspectors held discussions with TVA and Stone & Webster Engineering Corporation (SWEC) management personnel, Nuclear Assurance (NA) personnel, welding engineers, and craft personnel. The inspectors evaluated the adequacy of licensee management, effectiveness of recent self-assessments in this area, and corrective actions associated with documented deficiencies.

b. Observations and Findings

During the previous reporting period the inspectors determined there were potential adverse trends in four specific areas related to the TVA welding program (weld material issue and control, weld engineering documentation errors, welder performance deficiencies, and Qualified Individual errors). Multiple examples of weld filler material control errors, weld documentation errors and weld performance errors required evaluation to determine if any common cause factors required specific actions by the licensee. Additionally, the acceptance rate for welding Quality Control (QC) inspections had been approximately 95%, which had not satisfied the licensee's goal of 98% QC acceptance rate. The NRC review of previous licensee actions to resolve welding program deficiencies was documented in Inspection Report 50-259/2004-08.

The inspectors reviewed selected NA oversight reports and a special focused self-assessment report which covered Unit 1 welding activities. The inspectors also reviewed selected new PERs which documented recent problems in this area. Licensee Performance monitoring of craft welders and foremen and daily oversight of filler material issue activities by a weld engineer continued. The inspectors noted that only minor discrepancies were found, indicating the licensee's oversight of welding activities was effective. Additionally, the QC acceptance rate for welds improved and has remained above 98%. No deficiencies were identified by the inspectors during these observations.

c. Conclusions

Increased management focus and oversight of the welding program has resulted in fewer documentation errors and improvement in welder performance. Additionally, the acceptance rate for welding QC inspections remained greater than 98%. No violations or deviations of significance were identified.

E8 Miscellaneous Engineering Issues (92701)

E8.1 (Closed) IFI 50-259/04-07-01: Main Steam Line Linear NDE Indications

During re-baseline NDE examination of Class 2 Main Steam (MS) System piping welds, the licensee had identified unacceptable linear MT indications in the base material adjacent to Welds DSMS-1-13 and DSMS-1-30. PER 04-003621-000 was issued to identify and document corrective action for these indications. Sample expansion in accordance with ASME Section XI resulted in additional similar indications in the base material adjacent to other MS welds. The indications were all in the base material oriented predominantly in the axial direction and did not extend into the welds. The largest indication (adjacent to weld DSMS-1-30) was approximately 8" in length and ultrasonic depth sizing indicated a maximum depth of 0.320". “Boat” samples were taken from locations adjacent to three of the welds for metallurgical analysis. One of the three locations sampled was the indication with a UT measured depth of 0.320". The metallurgical analysis showed that all indications were caused by shallow manufacturing “Laps”. The actual thru-wall depth of the lap measured to be 0.320" by UT was determined, based on metallurgical analysis, to be 0.102". Based on the licensee’s investigation, the discrepancy between the UT measured flaw depth and the actual flaw depth was caused by the use of a conservative sizing procedure that could accurately size flaws orientated perpendicular to the material surface, but was not accurate for sizing flaws (such as laps) orientated parallel to the surface. These types of manufacturing flaws (surface laps) are typical for carbon steel material manufactured at the time of Browns Ferry’s original construction, and as documented in the licensee’s corrective action PER, do not cause structural integrity concerns. This item is closed.

E8.2 (Closed) Bulletin 84-02, General Electric Type HFA Relays Failures

The inspectors reviewed Bulletin 84-02: Failures Of General Electric Type HFA Relays In Use In Class 1E Safety Related Systems. The purpose of this bulletin was to inform licensees and CP holders of HFA relay failures reported in several General Electric (GE) Service Advice Letters (SALs) and Service Information Letters (SILs) which were issued in 1980 and 1982. The review of Bulletin 84-02 indicated that these failures involved relays with coils manufactured with standard Class A insulation, nylon or Lexan coil spools, and with standard temperature wire. The failures were due to temperature and aging. The Unit 1 Recovery Project has scheduled the replacement of all HFA Type

Latching Relays used in Class 1E applications. The replacement relays were designed with coils that were not susceptible to this type of failure. Bulletin 84-02 was closed for Unit 2 in Inspection Reports 50-259, 260, 296/88-28 and 50-259, 260, 296/88-32. Bulletin 84-02 was closed for Unit 3 in Inspection Report 50-259, 260, 296/95-43. Relay replacement activities were observed by the inspectors and documented in Section M1.1 of this report. As of December 2004, a majority of the total population of 349 relays had been changed out and the licensee's replacement project was on track for completion. 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 establish for the Unit 1 recovery issues. This issue is closed for Unit 1.

E8.3 (Closed) Bulletin 88-03, General Electric Type HFA Relays Inadequate Latching

The inspectors reviewed Bulletin 88-03: Inadequate Latch Engagement In HFA Type Latching Relays Manufactured By GE. The purpose of this bulletin was to request that licensees perform inspections to ensure that all relays installed in class 1E, safety related, applications have adequate latch engagement. The bulletin also requested that those relays which failed to meet acceptance criteria be repaired or replaced. The review of Bulletin 88-03, indicated that these failures involved relays that failed to mechanically latch adequately after being energized. Bulletin 88-03 also indicated that relays manufactured prior to November 1987 were suspect. The Unit 1 Recovery Project has scheduled the replacement of all HFA Type Latching Relays used in Class 1E applications. The inspectors determined that the replacement relays were manufactured after November 1987. The licensee's response to this bulletin, dated July 12, 1990, stated that plant procedures require that replacement relays be calibrated prior to installation to ensure proper latching. The inspectors reviewed plant procedure EMI-100, Replacement of HFA Relay Components and/or Calibration of HFA Relays. Bulletin 88-03 was closed for Unit 2 in IR 50-259, 260, 296/90-37. Bulletin 88-03 was closed for Unit 3 by SER dated August 2, 1990, NUREG 1435, from NRR and NRR determined that the bulletin was closed. Relay replacement activities were observed by the inspectors and documented in Section M1.1 of this report. As of December 2004, a significant number of the relays had been changed out. 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 wi th the same process, and because any implementation deficiencies would likely be detected by the licensee's oversight programs, this item meets the closure criteria establish for the Unit 1 recovery issues. This issue is closed for Unit 1.

E8.4 (Closed) Bulletin 88-04, Potential Safety Pump Loss

The inspectors reviewed Bulletin 88-04: Potential Safety-Related Pump Loss. The purpose of this bulletin was to request all licensees to investigate and correct two miniflow design concerns. The first concern involved the dead-heading of one or more pumps in systems that have a miniflow line common to two or more pumps. The first concern also involved other piping configurations that did not preclude pump-to-pump interaction during miniflow operation. A second concern was whether or not the installed miniflow was adequate for even a single pump operation. Licensee actions associated with this bulletin had been previously reviewed and found acceptable by the NRC. Bulletin 88-04 was closed for Unit 2 in Inspection Report 50-259, 260, 296/90-27. Closure of this bulletin for Units 1 and 3 was documented in NRR letter dated May 8, 1989, and Inspection Report 50-259, 260, 296/95-31. The inspectors reviewed the TVA letter reporting completion of actions associated with Bulletin 88-04, dated May 7, 2004. This letter described the licensee's verification of adequacy of the miniflow line sizing for Unit 1 Emergency Core Cooling Systems (ECCS), Residual Heat Removal Service Water (RHRSW), and Emergency Equipment Cooling Water (EECW) Systems. The inspectors noted that this licensee review was conducted similar to previously performed during recoveries of Units 2 and 3 and that no problems with dead-heading or miniflow capacity were identified. The inspectors determined that no further actions were required for Unit 1. This issue is closed for Unit 1.

III. Maintenance

M1 Conduct of Maintenance

M1.1 Replacement of GE Type HFA Relays (37551)

c. Inspection Scope

The inspectors continued to observe and review the licensee's ongoing activities associated with GE type HFA relays. The Unit 1 recovery personnel, at the end of this report period, had changed out a total of 192 of the 349 HFA relays initially identified as needing to be replaced.

d. Observations and Findings

The inspectors observed the ongoing replacement activities in the Unit 1 Auxiliary Instrument Room. The replacement of the relays was considered to be like - for - like and only required the use of Work Orders (WOs) for replacement. Some of the relays were being replaced in conjunction with the implementation of modifications, such as DCN 51080, System 99 Reactor Protective System (RPS). Among the WOs used with this DCN were 3-002001-07, 13, 14, 24, 25, and 30 for HFA relays installed in Control Panel 9-17, located in Unit 1 Auxiliary Instrument Room. The relays were also being replaced by WOs alone such as 03-001980-23, 25, and 27 for relays installed in Control Panel 9-15, and WOs 04-716376-00, 05, and 10 for relays installed in Control Panel 9-30. Both of these control panels were also located in the Unit 1 Auxiliary Instrument Room. The following activities were reviewed and observed:

- WO 03-011980-25, GE HFA relay BFR-1-RLY-099-05AK10 in panel 9-15
- WO 03-001980-09, GE HFA relay BFR-1-RLY-064-16A-K7 in panel 9-15

- WO 03-002001-12, GE HFA relay BFR-1-RLY-099-05AK01 in panel 9-17

- WO 04-716376-00, GE HFA relay BFR-1-RLY-001-2E-K10 in panel 9-30

The inspectors also reviewed PERs issued by the licensee documenting conditions adverse to quality observed during the change out process. The majority of the PERs were for historical issues involving drawing discrepancies (DD). Examples of DDs were included in PER 46428, which documented DD's on drawing 1-730E9291- R02, GE Type HFA Relay Tabulation; PER 63241 which documented DD's on drawing 1-791E245, Sheet 2, R01, and on drawing 1-791E245RE, Sheet 1, R01, for relays installed in Control Panel 9-15; and PER 63503 for DD's on drawing 1-791E247, Sheet 1A, RB, and on drawing 1-791E247RE, Sheet 3, R00, for relays installed in Control Panel 9-17. These items were considered historical issues due to the practice in the past of not adequately updating drawings, referred to as Red Lining, to reflect the as built status of electrical systems. If a relay were to be inadequately installed during the present change out process, then a PER would be written and the PER would not be considered as a historical issue.

c. Conclusions

Based on review of records and observation of ongoing work the inspectors concluded the licensee's HFA relay replacement program was complying with the applicable requirements.

IV. Plant Support

F1 Fire Protection Program

F1.1 Unit 1 Restart Special Program - Fire Protection - Appendix R (64704)

a. Inspection Scope

Regarding Unit 1, the inspectors reviewed the scope of the licensee's corrective actions that were developed for resolution of issues associated with the Browns Ferry fire protection plan (FPP) and Appendix R compliance issues. The corrective actions were evaluated for conformance with the objectives of the BFN Fire Protection Improvement Plan that was used for fire protection concerns prior to re-start of Units 2 and 3.

b. Findings and Observations

The Regulatory Framework letters for the restart of Unit 1, dated December 13, 2002 and February 28, 2003, were reviewed and found that the licensee's commitments for resolution of issues associated with Unit 1 fire protection program (FPP) and Appendix R compliance were based on completing the same activities on Unit 1 that were completed for Units 2 and 3 restart. In the BFN Nuclear Performance Plan (Revision 2), TVA described its overall fire protection program objectives and corrective actions for all known weaknesses in the Browns Ferry Fire Protection Program. The fire protection objectives of the Browns Ferry Fire Protection Improvement Plan are to comply with the requirements of Appendix R, National Fire Protection Code (NFPA) and commitments to Appendix A of Branch Technical Position (BTP) Auxiliary and Power Conversion Systems Branch (APCSB) 9.5-1. These objectives are accomplished by the implementation of design changes in the: (1) detection system upgrades, (2) suppression system upgrades, and (3) compartmentalization upgrades. Acceptance of the licensee's Browns Ferry Fire Protection Improvement Plan objectives by NRC is documented in a Safety Evaluation Report dated April 14, 1989, Subject: Safety Evaluation Report on the Browns Ferry Nuclear Performance Plan - NUREG-1232, Volume 3.

The inspectors reviewed the scope of corrective actions developed for resolving Fire Protection and Appendix R issues provided by the licensee and determined the Unit 1 Fire Protection Improvement Plan was accomplished by the following four design change notices:

- DCN 51368 - Detection system modifications.
- DCN 51180 - Suppression system modifications.
- DCN 51190 - Ventilation system modifications.
- DCN 51208 - Penetration seal modifications (includes fire doors).

The inspectors reviewed the Unit 1 permanent plant modification DCN 51368, Unit 1 Reactor Building Fire Detection and Alarm System (system 026). The intent of this DCN was to upgrade the fire alarm and detection system for the Unit 1 reactor building including the Unit 1 4160 VAC Board Rooms A and B, and 480 VAC Board Rooms 1A and 1B to meet NFPA Standard 72 (2002). Components installed included smoke detectors, thermal detectors, manual pull stations, alarm horns, monitor and control modules and the interconnecting wiring of local fire alarm and control panels. Critical fire detection trouble status and alarm conditions monitored from the local panels is to be integrated with the existing Units 2 and 3 fire alarm and detection systems.

The inspectors reviewed criteria in licensee procedures SPP-9.3, Plant Modifications and Engineering Change Control; SPP-7.1, Post-Modification Testing; and BFN-50-7308, General Design Criteria for the Fire Alarm and Detection System, to verify that the fire detection system modification was developed, reviewed, and approved per the licensee's procedure requirements. In addition, the inspectors evaluated the adequacy of the modification and observed field work to verify the design basis, licensing basis, and fire protection program performance for the system to actuate in the early stage of a fire. The inspectors also reviewed criteria in the licensee's fire protection report, calculations, engineering drawings for fire detector types, spacing, locations, installation procedures and modification instructions for installation of the new fire detection system.

The inspectors performed a walk down of the accessible portions of the fire detection systems in the Unit 1 Reactor Building elevations 593.0', 621.25', and 639.0', including the Unit 1 4160 VAC Board Rooms A and B, and 480 VAC Board Rooms 1A and 1B to confirm that the system equipment and detector locations were consistent with Section 4.3.4, of the Fire Protection Plan, the licensee's engineering drawings, calculations, and each area was protected by fire detectors in accordance with the Code of Record requirements - NFPA Standard 72 (2002). During the walk downs, the inspectors observed that the detector spacing was consistent with design location drawings and met NFPA code requirements and licensing commitments. In addition, the inspectors observed that penetration fire barrier seals for new installed fire detection system conduits were being properly maintained during the system modification.

A check of the calculation NDN 0026920065, Selection, Location, and Spacing of Fire Detection Devices, Rev. 8, associated with DCN 51368 was found to be performed in accordance with the documented fire protection program requirements. However, the inspectors identified a lack of clear acceptance criteria for application of NFPA Standard 72 engineering judgement in locating photoelectric products of combustion detectors near heating ventilation and air conditioning (HVAC) supply-air-diffuser outlets (located close below the detector). The inspectors were concerned that inadequate placement could result in fire plume and smoke movement away from the detector device. PER 73587 was written to address the detector location issue.

c. Conclusions

The scope of corrective actions developed by the licensee, for resolving issues associated with the Fire Protection Program and Appendix R compliance prior to Unit 1 restart, together with performance based inspection of the Unit 1 fire detector system modification, led to the conclusion that implementation of the Unit 1 Browns Ferry Fire Protection Improvement Plan is proceeding in accordance with licensee commitments. 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 conducted to verify that the overall fire protection program was implemented adequately.

The inspectors' review of Unit 1 modification design package DCN 51368 to upgrade the fire alarm and detection system concluded that the design change was technically acceptable and appropriately developed, reviewed, and approved for implementation per procedural requirements. The modification contained adequate 10 CFR 50.59 safety evaluations; considered appropriate design input information; and identified affected documents including drawings, calculations, Final Safety Analysis Report (FSAR), the licensee's Individual Plant Examination for External Events (IPEEE), and equipment database, etc. No violations or deviations were identified.

F1.2 Unit 1 Restart Special Program - Fire Protection Administrative Controls (64704)

a. Inspection Scope

The inspectors reviewed licensee procedures, SPP-10.10, Control of Transient Combustibles, SPP-10.11, Control of Ignition Sources (Hot Work), and SPP-10.9, Control of Fire Protection Impairments, which established the controls and practices to prevent fires and to control the storage of permanent and transient combustible materials and ignition sources for Units 1, 2, and 3 during Unit 1 recovery activities. The inspectors walked down ten plant areas to verify a selected sample of the licensee controls for storage of permanent and transient combustible materials and ignition sources. Also, the inspectors observed two hot work operations in the Unit 1 Reactor Building to verify that the work activities were identified and controlled in accordance with procedure SPP-10.11.

b. Findings and Observations

The inspectors conducted walk downs of the plant areas listed below:

- Unit 1 Reactor Building Elevation 565', 593', 621', and 639'
- Units 1 & 2 Emergency Diesel Generator Building
- Unit 1 4-kV Shutdown Board Rooms (SDBR) 1A&1B
- Unit 1 480 V Reactor Motor Operated Valve (RMOV) Board Rooms 1A&1B
- Intake Structure (Electric Driven Fire Pumps)

The inspectors observed that the licensee was controlling hot-work activities, and controlling combustible materials in a manner consistent with the UFSAR, administrative procedures and other fire protection program procedures. Cutting and welding operations in progress were properly authorized by a permit and a fire watch with an extinguisher posted for the duration of the work. Transient fire loads associated with plant maintenance, modifications, and construction activities were being adequately controlled by the licensee's fire protection program.

c. Conclusions

Implementation of the plant’s fire prevention program for the control of storage of permanent and transient combustible materials and ignition sources met fire protection program procedures. No violations or deviations were identified.

F1.3 Unit 1 Restart Special Program - Fire Protection Systems, Features, and Equipment

a. Inspection Scope

The inspectors reviewed the fire pumps, fire protection water piping, portable fire extinguishers and manual suppression standpipe and fire hose system to verify adequate design, installation, and operation. The inspectors reviewed NFPA deviations, fire extinguisher placement drawings, and fire hose station drawings to ensure that the systems and equipment were installed in accordance with their design, and that their design was adequate given the current equipment layout and Unit 1 plant configuration. The inspectors performed in-plant walk-downs and observed the material condition; accessibility; and, functionality of the fire pumps, fire water system isolation valves, standpipe systems hose racks, and CO2 fire suppression system controls. The inspectors also examined design flow calculations and flow measurement/pressure test data to verify that the required fire hose water flow for each protected area was available.

b. Findings and Observations

Using plant surveillance procedure, 0-SI-4.11.B.1.b, Fire Protection System Valve Position Verification, and general operating procedure, 0-GOI-300-3, General Valve Operations, the inspectors conducted walk downs of the system isolation valves without control room indications for the intake structure fire pumps, portions of fire protection water supply system to the Unit 1 reactor building, and selected CO2 fire suppression system controls for the Units 1 & 2 Emergency Diesel Generator Rooms. The inspectors observed that the fire suppression systems' flow path isolation valve positions were generally controlled and secured using small chains with break-away seals as indicated on the system drawings and surveillance procedure. The inspectors found CO2 system isolation valve 0-39-604, (gas supply valve to the master control pilot valve located in the CO2 compressor housing) in the open position; however, this valve had no control room indication nor locked valve control requirement. Section 5.4 of procedure, 0-GOI-300-3 requires, in part, that valves without control room indication located in the main flow path be locked when, if mis-positioned, could prevent system function. The inspectors determined that, although valve 0-39-604 is not in the main CO2 system flow path, closure of this valve would prevent system function (loss of the system automatic actuation function) since isolation of pilot gas supply to the master flow control valve disables the pneumatic force required to open the master pilot valve allowing CO2 to flow to the hazard. It therefore warranted consideration to be locked. When identified by the inspectors, the licensee initiated problem evaluation report (PER) 73907 to evaluate the need to lock valve 0-39-604 consistent with other TVA facilities. The licensee indicated that procedural changes would be incorporated to lock the subject valve and this was documented into their corrective action program.

The inspectors observed that a number of pathways to fire standpipe systems hose racks in the Unit 1 reactor building were heavily obstructed by gang boxes, movable carts, scaffold carts, and other construction equipment. The inspectors determined that accessibility to the standpipe systems hose racks by fire brigade personnel was possible because most of the obstructions were moveable. However, this condition would hinder fire brigade access to the stations and possibly delay hose off racking and movement for fire fighting. This minor error was documented by the licensee in PER 73330 to address the oversight.

The inspectors observed that the material condition of Unit 1 reactor building fire suppression water supply piping was noticeably degraded. Fire protection piping to several hose stations particularly on 593' downstream of isolation valve 1-26-982, were found to have at least five piping patches installed along a 10-15 feet length of piping that were sealing existing pipe leaks. The inspectors did not identify any current leaks on the subject piping which would have questioned the operability of the fire protection water system. The inspectors noted that replacement of this piping was not included in the scope of the suppression system replacement DCN 51180. This condition could potentially adversely affect implementation of the DCN due to corrosion that may have weakened the pipe wall where pipe welds are required. The licensee indicated that this section of piping was planned to be replaced via WO 02-007391-000 which was planned for July 2005.

c. Conclusions

Implementation of the plant’s valve position and controls for fire suppression systems was satisfactory and valve positions were generally controlled and secured using small chains with break-away seals as indicated on the system drawings and operating procedures. CO2 system isolation valve 0-39-604, was found in the open position; however, this valve had no control room indication nor locked valve control requirement.

Accessibility to the standpipe systems hose racks in the Unit 1 reactor building by fire brigade personnel was degraded. A number of pathways to the fire standpipe systems hose racks were heavily obstructed by gang boxes, movable carts, scaffold carts, and other construction equipment. This condition would hinder fire brigade access to the stations and possibly delay hose off racking and movement for fire fighting. This minor error was documented by the licensee in PER 73330 to review for corrective actions.

Examples were identified indicating that portions of the Unit 1 reactor building fire suppression water supply piping was noticeably degraded. Fire protection piping to several hose stations particularly on elevation 593' downstream of isolation valve 1-26-982 [R4-P] were found to have at least 5 piping patches installed along a 10'-15' length of piping that were sealing existing pipe leaks. The licensee had previously identified the condition and had planned actions to improve the material condition of the piping.

No violations or deviations were identified.

F7 Quality Assurance in Fire Protection Activities

F7.1 Licensee Quality Assurance Oversight of Recovery Activities (Identification and Resolution of Problems) (71152)

f. Inspection Scope

Corrective action program (CAP) problem evaluation reports (PERs) resulting from fire, smoke, sparks, arcing, and equipment overheating incidents for the last 18 months were reviewed to assess the frequency of fire incidents and to identify any maintenance or material condition problems related to fire incidents. The inspectors also reviewed other CAP documents, including completed corrective actions documented in selected PERs, and operating experience program (OEP) documents to verify that industry-identified fire protection problems potentially or actually affecting Browns Ferry were appropriately entered into and resolved by the CAP process. Items included in the OEP effectiveness review were NRC Information Notices, industry or vendor-generated reports of defects and noncompliance under 10 CFR Part 21, and vendor information letters. The inspectors reviewed Unit 2 and 3 fire protection systems health reports for the last six quarters to evaluate the prioritization for resolving fire protection related deficiencies and the effectiveness of corrective actions. In addition, the inspectors reviewed a sample of licensee self-assessments of the fire protection program to verify that items related to fire protection and to SSD were appropriately entered into the licensee's CAP in accordance with the Browns Ferry quality assurance program and procedural requirements. The items selected were reviewed for classification and appropriateness of the corrective actions taken or initiated to resolve the issues.

b. Findings

No violations or deviations were identified.

V. Management Meetings

X1 Exit Meeting Summary

On February 3, 2005, the resident inspectors presented the inspection results to Mr. John 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

T. Abney, Nuclear Site Licensing & Industry Affairs Manager
M. Bali, Electrical Engineer (Bechtel)
R. Baron, Nuclear Assurance Manager, Unit 1
D. Beckley, Electrical Engineer (Bechtel)
M. Bennett, QC Manager. Unit 1
R. Bentley, NDE Level III
D. Burrell, Electrical Engineer, Unit 1
T. Butts, SWEC Mechanical Supervisor
P. Byron, Licensing Engineer
J. Corey, Radiological and Chemistry Control Manager, Unit 1
W. Crouch, Mechanical/Nuclear Codes Engineering Manager, Unit 1
R. Cutsinger, Civil/Structural Engineering Manager, Unit 1
R. Drake, Maintenance and Modifications Manager, Unit 1
B. Hargrove, Radcon Manager, Unit 1
R. Jackson, Bechtel
S. Johnson, TVA Welding Engineering Supervisor, Unit 1
R. Jones, Plant Recovery Manager, Unit 1
S. Kane, Licensing Engineer
J. Lewis, ISI Program Engineer, Unit 1
G. Lupardus, Civil Design Engineer, Unit 1
J. Ownby, Project Support Manager, Unit 1
J. Pettitt, Pipe Replacement Task Manager
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 55050	Nuclear Welding General Inspection Procedure
IP 57050	Visual Testing Examination
IP 57060	Liquid Penetrant Testing Examination
IP 57090	Radiographic Examination Procedure Review/Work Observation/Record Review
IP64704	Fire Protection Program Inspection Procedure
IP 71111.17	Permanent Plant Modifications
IP 71111.23	Temporary Plant Modifications
IP 73753	Inservice Inspection
IP 73755	Inservice Inspection Data Review and Evaluation
IP 92701	Follow-up

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

None

Opened

None

Closed

50-259/04-07-01	IFI	Main Steam Line Linear NDE Indications (Section E8.1)
84-02	BUL	General Electric Type HFA Relays Failures (Section E8.2)
88-03	BUL	General Electric Type HFA Relays Inadequate Latching (Section E8.3)
88-04	BUL	Potential Safety Pump Loss (Section E8.4)

Discussed

None.

LIST OF DOCUMENTS REVIEWED

Section E1.1 Design Change Packages

Procedures and Standards

SPP-9.3, Plant Modifications and Engineering Change Control, Rev. 9

DCNs

DCN 51080, Reactor Protective System (RPS), System 099
DCN 51152, Core Spray (CS) - Drywell, System 075
DCN 51189, Primary Containment System (PCS), System 064A and Primary Containment Isolation System (PCIS), System 64D
DCN 51199, Residual Heat Removal (RHR) - Reactor Building, System 074
DCN 51163, Reactor Vessel Level Indicating System (RVLIS) - Drywell, System 03
DCN 51231, Reactor Vessel Level Indicating System (RVLIS) -Reactor Building, System 03

Section E1.2 Plant Modifications

Procedures and Standards

SPP-9.3, Plant Modifications and Engineering Change Control, Rev. 9
MAI-4.2B, Piping, Rev 20
G-94, Piping Installation, Modification, and Maintenance, Rev. 2

DCNs

DCN 51016, Unit 1 Emergency Core Cooling System (ECCS) Accident Signal Logic
DCN 51018, Unit 2 ECCS Accident Signal Logic
DCN 51107, Control Annunciator Upgrade, System 55
DCN 51199, RHR, System 74
DCN 51200, CS, System 75
DCN 51195, Reactor Building Closed Cooling Water (RBCCW), System 70

Section E1.3 Unit 1 Restart Special Program Activities - Cable Installation and Cable Separation

Specifications

G-38, Installation, Modification, and Maintenance of Insulated Cables Rated up to 15,000 Volts, Rev. 19 (Section 3.2.1.6 on pulling tension calculations)

Calculations

EDQ1 999 2003 0015, Analysis of Unit 1 Cable Installation - Miscellaneous Issues, Rev. 1 and Rev. 2.
EDQ1 999 2003 0016, Analysis of Cable Support in Vertical Raceway for Unit 1, Rev. 2.

Design Criteria Documents

BFN-50-728, Physical Independence of Electrical Systems, Rev. 13.

Miscellaneous

Engineering application information by American Polywater Corporation (manufacturer of cable pulling lubricant) obtained at web site.
CBLPUL Computer Software System ID 262352 User Manual, Version 6.

Section E1.4 Unit 1 Restart Special Program Activities - Flexible Conduit

Specifications

G-40, General Engineering Specification, Installation, Modification, and Maintenance of Electrical Conduit, Cable Trays, Boxes, Containment Electrical Penetrations, Electrical Conductor Seal Assemblies, Lighting and Miscellaneous Systems, Rev. 15 (Section 3.2.6 Flexible Conduit)

Section E1.5, Unit 1 Restart Special Program Activities - Intergranular Stress Corrosion Cracking (IGSCC) -Welding of Replacement Recirculation System Piping

Procedures and Standards

MMDP-8, Controlling Welding Filler Material (WFM), Revision 0
G29A General Engineering Specification GWPS 1.M.1.2, General Welding Procedure Specification for American Society of Mechanical Engineers (ASME) and American National Standards Institute (ANSI), Revision 2
Detail Welding Procedure Specification (DWPS) GTA88-C-2-N, Revision 1, and Associated Procedure Qualification Record (PQR) GT88-C-5
Detail Welding Procedure Specification (DWPS) GT-O-1-N, Revision 5, including applicable PQR
Nondestructive Examination Procedure N-RT-1, Radiographic Examination of Nuclear Power Plant Components, Revision 25
Nondestructive Examination Procedure N-MT-6, Magnetic Particle Examination for ASME and ANSI Code Components and Welds, Revision 26

DCNs and Work Documents

DCN 51151
DCN 51152
Work Orders (WOs) 02-009273-007, Replace 20" RHR Shutdown Supply

WO 02-009233-002, Replace Valve FSV-74-54 and Replace RHR Piping/Valve and Associated Small Bore Pipe/Valve Branches to X-134

Problem Evaluation Reports (PERs)

71382 - Deletion Shielding Gas Cup Size for Gas Tungsten Arc DWPs
70510 - Errors and Omissions in Welding Documentation for WO 02-009273-007
71401 - Incorrect Welder ID Entered on Log for Automatic Welding in Weld RHR-1-013-002
04-003621-000 - MT Indications in Main Steam Piping

Section E1.6: Inservice/Preservice Inspection (ISI/PSI)

Procedures and Standards

N-UT-76 Generic Procedure for Ultrasonic Examination of Ferritic Pipe Welds, Rev 4

Examination Reports

NDE Report for UT of Weld CS-1-002-014
NDE Report for UT of Weld CS-1-002-028

Section E1.7 Unit 1 Restart Special Program Activities - Containment Coatings

Specifications & Procedures

TVA General Engineering Specification G-55, Technical and Programmatic Requirements for the Protective Coating Program for TVA Nuclear Plants, Rev. 13, dated 2/17/04.
Browns Ferry Site Exceptions to TVA Specification G-55, numbers G-55-BFN-6 through -9, -11, and -14 through -16
MAI-5.3, Protective Coatings for Service Level I, II, III and Corrosive Environments, Rev. 32, dated 1/26/04.
SPP-4.1, Procurement of Material, Labor, and Services, Rev. 13
NEDP-8, Technical evaluations for Procurement of Materials and Services, Rev. 10
Walkdown Instruction WI-BFN-1-MEB-03, BFN Unit 1 Primary Containment Coatings Inspection Plan (for identification of uncontrolled coatings and coatings that were applied to stainless steel)
TI-417, Inspection of Protective Coatings in the Interior Surfaces of Primary Containment, Rev. 1

Problem Evaluation Reports (PERs)

04-048609-000, K&L 6548/7107 was Applied to Repair Plasite in Vapor Space in Torus Bay 1. K&L 6548/7107 not Qualified for use in Repairs to Plasite
04-060438-000, 04-060588-000, 04-060673-000, 04-061052-000, 04-062762-000, & 04-062764-000, Surface Preparation Deficiencies
04-062385-000 & 04-63931-000, Inadequate Coatings application
04-046805-000, Inconsistent Criteria for Inspection of Existing Coatings in Torus Vapor Space
04-046980-000, Failure top Identify Damage in Existing Coatings in Vapor Space for Repair
04-061905-000, Questions on Inspection Techniques Used for Existing Torus Coatings
04-068475-000, Insufficient DFTs, new coatings, various areas in Torus
03-041031-000, Use of Material (Peel away 7) with Code Level III on stainless steel.

Miscellaneous Documents

General Design Criteria Document BFN-50-C-7100, Design of Civil Structures, Rev. 14, dated 3/17/04
Keeler & Long letter dated 8/31/04, and attached Test Report number 81-0302, dated 3/2/81, Qualification Test Result for Keeler & Long Kolor Poxy 6548/7107
Singleton Laboratories Test Report SL 209-066-001A, Browns Ferry DBA Testing of Protective Coatings for Coatings Service Level I
Singleton Laboratories Test Report SL 209-066-001B, Browns Ferry DBA Testing of Protective Coatings for Coatings Service Level I
Technical Evaluation Report - Singleton DBA Qualification Report
TVA Purchase Order 00010630, Keeler & Long Solvent Thinner, Epoxy Paint, White KL6548-7107, and Epoxy Paint, Gray, KL6548-8707

Section E1.8 Unit 1 Restart Special Program Activities - Cable Ampacity

Procedures, Guidance Documents, and Manuals

Electrical Design Standard DS-E12.6.3, Auxiliary and Control Power cable Sizing, up to 15,000 Volts, Revision 10.
TOM-FTM-6-INSU-003, High- Potential Testing, Revision 0.
SS-E12.6.01, 5-15kV Cables, Ethylene-Propylene Rubber Insulated, Revision 4.
SPP-4.3, Storage and Handling Instructions for Cable and Wire, Appendix D, Revision 5.

DCNs and Work Documents

DCN 51222, BFNP Unit 1 Restart-Electrical Lead DCN -System 074 (Reactor Building), Revision A.
DCN 51216, BFNP Unit 1 Recovery - Electrical Lead DCN System 57-4 (Reactor Building), Revision A.
PIC No. 62202, Change Request Form Equip. IDs. System Codes: RHR Pump 1B (1-PMP-074-28) / System 74, approved 08/09/2004.

Other Documents

EDQ1-999-2002-0023, Cable Ampacity Calculation for Safety Related V5 Power Cables Outside the Drywell Revision 9.
EDQ0999870135, Cable Ampacity Calculation - V4 and V5 Safety Related Trays Required for Units 1, 2, and 3 Operation, Revision 18.
EDQ-1999-2002-0023, Cable Ampacity Calculation for Safety- Related V5 Power Cables Outside the Drywell, Revision 8.
EDQ-0057-2002-0022, 4.16KV and 480V Bus load, Voltage Drop and Short Circuit Calculation, Revision 2.
EDQ-2000-870548, Cable and Bus Protection/ Breaker Coordination for 4KV switchgear and 480V Load Center, Revision 23.
B29 040823 010, Procurement Data Sheet, Revision 1.
Purchase Order 00029752, 09/30/2003.

R27 040331 329, Mark # WNC-52 Cable Test Report Approval, Revision 0.
R27 031002 416, Cable, J-Boxes & Splice Kits, Revision 1.
B29 040603 014, Cable Mark No. WNC-52-Okonite- Test Report Approval, Revision 0.

Section E1.9 Control Room Design Review

Modifications

DCN 51094, Control Room Design Review, Panel 1-9-3, Rev. 1
DCN 51096, Control Room Design Review, Panel 1-9-5, Rev. 2
DCN 51098, Control Room Design Review, Panel 1-9-8, Rev. 1
DCN 51107, Unit 1 Control Room Annunciators, System 55, Rev. 2
DCN 51109, Control Room Design Review, Control Panel Enhancements, Rev. 2
DCN 51111, Control Room Design Review, Panel 1-9-6, Rev. 1

Section E1.10 Temporary Modifications

Procedures, Guidance Documents, and Manuals

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

Other Documents

TACF 1-04-010-099, U1 Reactor Protective System (RPS), Primary Containment Isolation System (PCIS), and Backup Scram Circuitry Power Supplies
TACF 1-04-011-064D, U1 PCIS Isolation Signals to Various Systems
TACF 1-04-012-074, U1 Residual Heat Removal (RHR) Isolation PCIS Over Ride
TACF 1-04-013-069, U1 Reactor Water Cleanup (RWCU) Isolation PCIS Over Ride
TACF 1-04-014-064, Temporary Air Supply for Unit 1 Torus Sand Blasting and Coating Activities

Section E1.11 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, Rev. 0
0-TI-404, Unit One Separation and Recovery, Rev. 4

Problem Evaluation Reports (PERs)

63534, procedure O-TI-466, Plant Modification Related Work Order Preparation and Processing, did not provide clear guidance on how to incorporate the requirements of special plant procedure SSP- 8.3, Post Modification Test Control, into modification work orders
70417, the SPEA package for DCN 51177, RHRSW in the Reactor Building, System 23, had incorrectly identified some valves as Hand Control Valves (HCV) instead of the proper identifier of Shutoff Hand Valves (SHV)

72412, during SRTS activities for System 57-4, 480 VAC Shutdown Boards, temperature indicating thermocouples were not installed in the coils of the new transformers, per Vendor Drawing 0-CDV-F-5221-32359S

Section E1.12 Restart Test Program

Procedures and Standards

Technical Instruction 1-TI-469, Baseline Test Requirements
Post Modification Test Instruction 1-PMTI-BF-078.023, Functional Testing of Fuel Pool Cooling Handswitches on Panel 1-9-4, Rev. 2

Other Documents

Baseline Test Requirement Document 01-BFN-BTRD-078, Spent Fuel Pool Cooling, Rev. 2
System Test Specification 1-STS-078, Spent Fuel Pool Cooling, Rev. 1
DCN 51095, Control Room Design Review, Panel 1-9-4, Rev. 1
DCN 51203, Spent Fuel Pool Cooling, Reactor Building

Section E7.1 Licensee Quality Assurance Oversight of Recovery Activities

Nuclear Assurance Audit/Assessment Reports

Browns Ferry Engineering Assessment BFR-RMM-04-005, 8/23/04 - 9/23/04
Browns Ferry Unit 1 Nuclear Assurance Oversight Report NA-BF-04-002, 6/13/04 - 7/11/04
Browns Ferry Unit 1 Nuclear Assurance Oversight Report NA-BF-04-010, 7/12/04 - 8/8/04
TVA Corporate Nuclear Assurance Self-assessment Report CRP-NA-04-004, 4/12-16/04

Problem Evaluation Reports (PERs)

63896, Weld material handling and issuance deficiencies
65044, Weld engineering documentation deficiencies
65047, Weld rod stored in incorrect bin
66495, Welder in field with lapsed qualification
66559, Welder required reading deficiencies
69148, Welder performance deficiencies
69015, Welder required reading deficiencies
70046, Unit 1 welding deficiencies
70119, Weakness with foreman knowledge
70123, Craft foreman performance of QI function inadequate
70837, Issuance of incorrect weld material
73440, Undersize fillet weld
73489, Rounded indication on weld RWR-1-001-059

Section F1 Fire Protection

Procedures, Instructions, and Guidance Documents

TVAN General Operating Instruction, 0-GOI-300-3, General Valve Operations, Rev. 99
TVAN Standard Programs and Processes, SPP-9.3, Plant Modification and Engineering Change Control, Rev. 9
TVAN Standard Programs and Processes, SPP-10.9, Control of Fire Protection Impairments, Rev. 2
TVAN Standard Programs and Processes, SPP-10.10, Control of Transient Combustibles, Rev. 3
TVAN Standard Programs and Processes, SPP-10.11, Control of Ignition Sources (Hot Work), Rev. 2

Drawings

0-47W391-9, Fire Protection - 10CFR 50 Appendix R Penetration Internal Conduit Fire Seals, Rev. 3
1-47E836-1, Unit 1 Flow Diagram Fire Protection & Raw Service Water System, Rev. 39
1-47E850-2, Unit 1 Flow Diagram Fire Protection & Raw Service Water System, Rev. 22
1-47E850-5, Unit 1 Flow Diagram Fire Protection & Raw Service Water System, Rev. 10
1-47W600-282, Unit 1 Fire Detection and Alarm System, El. 593.0', Rev. 2
1-47W600-283, Unit 1 Fire Detection and Alarm System, El. 621.25', Rev. 4
1-47W600-284, Unit 1 Fire Detection and Alarm System, El. 639.0', Rev. 3

Calculations, Analyses, and Evaluations

BFN-NDN- 0026-920065, Unit 1 Selection, Location, and Spacing of Fire Detection and Alarm Devices, Rev. 9
BFN-NDN- 999-2004-0010, Unit 1 IPEEEE Fire Induced Vulnerability Evaluation, Rev. 0

Audits and Self-Assessments

Self-Assessment, BFN-OPS-03-009, dated August 26, 2003

Problem Evaluation Reports (PERs)

03-001375-000, Diesel Driven Fire Pump Temperature Records
03-002935-000, Fire Pump Failed Capacity Test
03-008165-000, Evaluate Heat Collectors Over Sprinklers Per IN 2002-24
03-009529-000, Asiatic Clams Found in Yard Fire Protection System
03-013828-000, Procedure MMDP-1 Does Not Consider Impacts on Fire Protection Administrative Controls

Design Criteria and Standards

BFN-50-7308, General Design Criteria for Fire Alarm and Detection System, Rev. 3

Completed Surveillance Procedures and Test Records

Surveillance Instruction, 0-SI-4.11.B.1.b, High Pressure Fire Protection System Valve Position Verification, Rev. 38, completed November 26, 2004
Surveillance Instruction, 0-SI-4.11.E.1.b(1), Fire Hose Station Operability/Flow Test, Rev. 3, completed August 28, 2001

Applicable Codes and Standards

NFPA 72, Standard on Automatic Fire Detectors, 1990 Edition

Other Documents

Fire Protection Report Volume 2, Section I-D, Smoking Restrictions, Rev. 0
U. S. Consumer Product Safety Commission, Invensys Building Systems Announce Recall of Siebe Actuators in Building Fire/Smoke Dampers, dated October 2, 2002

License Basis Documents

Fire Protection Report Volume 1, Fire Protection Plan, Rev. 29
Browns Ferry Nuclear Plant - Summary of Deviations from NFPA Code for BFN, dated August 3, 1988

Work Orders Generated During this Inspection

03-016883-000, BFN-0-PMP-026-0003 Pump Packing Leak, Fire Pump C
03-017102-000, BFN-0-ISV-026-0565 Valve Packing Leak, Fire Pump A Discharge Shutoff Valve
03-017292-000, Smoke detector 0-SDE-26-87JW is installed in the incorrect location from that shown on location plan 0-47W600-268 (Fire Area 13, location plan)
03-018587-000, Channel Diesel Fire Pump fill Valve was not Locked in the Open Position
03-018593-000, Generic Review of SQN PER 03-011569-0 on NRC Concerns Regarding Compensatory Measures