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

August 9, 2004

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 QUARTERLY INTEGRATED INSPECTION REPORT 05000259/2004007

Dear Mr. Singer:

On July 10, 2004, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection associated with recovery activities at your Browns Ferry 1 reactor facility. The enclosed integrated inspection report documents the inspection results, which were discussed on July 26, 2004, with Mr. Jon 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 and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel. A significant portion of your engineering activities, Special Program implementation, and modification activities were reviewed during this inspection period and found to be effective with no significant problems identified. However, based on the results of this inspection, one Severity Level IV violation of NRC requirements that resulted from failure to follow Radiation Work Permits was identified. However, the NRC is treating this finding as a non-cited violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy. If you contest the NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at Browns Ferry Nuclear Plant.

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 05000259/2004007
w/Attachment: Supplemental Information

cc w/encl: (See page 3)

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

James E. Maddox, Vice President
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

Thomas J. Niessen, Acting General Manager
Nuclear Assurance
Tennessee Valley Authority
Electronic Mail Distribution

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

Mark J. Burzynski, 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

U.S. NUCLEAR REGULATORY COMMISSION

REGION II

| | |
|---|---|
| Docket No: | 50-259 |
| License No: | DPR-33 |
| Report No: | 05000259/2004007 |
| Licensee: | Tennessee Valley Authority (TVA) |
| Facility: | Browns Ferry Nuclear Plant, Unit 1 |
| Location: | Corner of Shaw and Nuclear Plant Roads<br>Athens, AL 35611 |
| Dates: | April 11 - July 10, 2004 |
| Inspectors: | W. Bearden, Senior Resident Inspector, Unit 1<br>E. Christnot, Resident Inspector<br>J. Kreh, Emergency Preparedness Inspector (Section P1)<br>J. Lenahan, Senior Reactor Inspector (Sections E1.11, E1.12, E8.6, E8.7)<br>C. Smith, Senior Reactor Inspector (Section E1.5)<br>B. Crowley, Senior Reactor Inspector, (Section E1.6)<br>P. Fillion, Reactor Inspector (Sections E1.3, E1.4)<br>J. Fuller, Reactor Inspector (Section E1.7)<br>D. Mas-Penaranda, Reactor Inspector (Sections E1.3, E1.4 )<br>N. Staples, Reactor Inspector (Section E1.5)<br>A. Vargas, Reactor Inspector (Section E1.7)<br>M. Maymi, Reactor Inspector (Sections E1.11, E1.12, E8.6, E8.7) |
| Approved by: | Stephen J. Cahill, Chief<br>Reactor Project Branch 6<br>Division of Reactor Projects |

Enclosure

## EXECUTIVE SUMMARY

Browns Ferry Nuclear Plant, Unit 1
NRC Inspection Report 05000259/2004-007

This integrated inspection report 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 inservice inspection, emergency planning and Unit 1 Special Programs in the areas of drywell steel platforms; containment coatings; cable ampacity; cable installation; cable separation; flexible conduit; and intergranular stress corrosion cracking (IGSCC).

Inspection Results - Engineering

- Review of Unit 1 modification design packages for four modifications concluded that the design changes were appropriately developed, reviewed, and approved for implementation per licensee procedural requirements. (Section E1.1)

- Modification installation activities associated with three permanent plant design changes were performed in accordance with the documented requirements. (Section E1.2)

- Inspection of the methodology for the Cable Installation and Cable Separation special programs, together with performance based inspection in the area of cable installation issues, led to the conclusion that implementation of these programs is proceeding in accordance with licensee commitments. (Section E1.3)

- Inspection of the methodology for the Flexible Conduit special program led to the conclusion that implementation of this program is proceeding in accordance with licensee commitments. (Section E1.4)

- The methodology developed by the licensee for resolving cable ampacity issues, although different from the program used to support the restart of Units 2 and 3, was considered adequate for resolving cable ampacity issues related to the restart of Unit 1. (Section E1.5)

- The licensee's IGSCC mitigation plan continued to meet commitments established by the licensee's Regulatory Framework letters. Recirculation System piping replacement activities were meeting American Society of Mechanical Engineers (ASME) Code and other regulatory requirements. (Section E1.6)

- The licensee's Inservice/Preservice Inspection (ISI/PSI) Program was meeting applicable regulatory requirements and licensing commitments. (Section E1.7)

- A review of a temporary alteration and 10 CFR 50.59 evaluation associated with closure of sluice gates on the Residual Heat Removal Service Water (RHRSW) supply did not identify any significant impact on the operability of equipment required to support operations of Units 2 and 3. (Section E1.8)

- 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 licencee's corrective action program. (Section E1.9)

- Ongoing Unit 1 recovery activities at the Power Service Shop (PSS) exhibited a high level of professionalism. Work documents included applicable technical and quality requirements. No violations or deviations were identified. (Section E1.10)

- Based on records review and observation of recent installations of structural steel the inspectors concluded that Drywell Steel Platform special program activities continued to satisfy applicable requirements. (Section E1.11)

- Based on review of corrective action documents, acceptance criteria, and qualification records for identification, removal, and repair of unqualified coatings in the Unit 1 Torus the inspectors concluded that Containment Coatings Special Program activities satisfied applicable requirements. (Section E1.12)

- The recently implemented System Cleanliness Verification Program has provided for comprehensive inspections of systems to continue the identification of component degradation or special requirements to support the Unit 1 recovery. (Section E1.13)

- The Licensee program for oversight of Unit 1 recovery activities performed at the PSS was well planned and Quality Assurance assessors were knowledgeable of the applicable work document requirements, PSS programs, and work processes. The licensee's inspection of the Unit 1 warehouse storage identified a minor material storage deficiency. (Section E7.1)

Inspection Results - Maintenance

- Ineffective communications between the Unit 1 maintenance/modifications organization and operations work control is considered a weakness in the area of work control/conduct of maintenance. (Section M1.1)

- 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 M1.2)

Inspection Results - Plant Support

- A Severity Level IV Non-Cited Violation for failure to follow Radiation Work Permit (RWP) requirements was identified during the current inspection. The error resulted from inattention to detail and failure of craft supervision to perform a walkdown of the jobsite prior to beginning work. The error resulted in some personnel contaminations and contamination of a portion of the drywell. (Section R1.2)

- The licensee's Emergency Preparedness Program was being maintained in a state of operational readiness. Changes to the program since the last inspection were consistent with licensee commitments and NRC requirements, and did not decrease the licensee's overall state of preparedness. Based on focused Emergency Preparedness (EP) reviews for Unit 1, the inspectors did not identify any impediments to the future transition of Unit 1 EP inspections under the EP Cornerstone for the Reactor Oversight Process. (Section P1.1)

## 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 Packages (37550)

a. Inspection Scope

The inspectors reviewed permanent plant modifications to Emergency Core Cooling System (ECCS) accident signal logic, core spray system instrumentation, and reactor building 480 VAC electrical system. 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 Design Change Notice (DCN) 51016

The inspectors reviewed the Unit 1 permanent plant modification DCN 51016, ECCS accident signal logic (systems 068, 074, 075, 082, and 211). The intent of this DCN was to implement the modifications recommended for the Unit 1 and 2 emergency core cooling system, Unit 1 RHR initiation logic, and the Unit Core Spray initiation logic. This DCN, in conjunction with DCNs 51018 and 51216, will correct the following action items and commitments from Reportable Occurrence Reports (ROR), Condition Adverse to Quality Reports (CAQR), and Licensee Event Reports (LER):

- The existing AC power system and logic scheme, including load sequencing, does not fully accommodate the various sequences of the occurrence of real and credible spurious accident signals.

- A failure of a battery supplying logic for one division of RHR would result in an RHR pump attempting to start on a de-energized electrical board and in conjunction with a loss of off site power and a design basis loss of coolant accident will result in an insufficient combination of RHR pumps needed to meet the peak cladding temperature of the 10 CFR 50, Appendix K, analysis.

- A Unit 1 common accident signal cable that was damaged, and resulted in the unexpected auto start of the Unit 3 emergency diesels, will be replaced.

- The Unit 1 accident signal will remain disabled until the ability to load shed due to the loss of main plant battery 1 concurrent with a Unit 1 accident signal is provided.

The issues associated with the RHR and Core Spray 4KV pump initiation logic; and the issues associated with the 480V load shed logic will be resolved by moving the 250V DC logic power supply to Unit 1 & 2 480V load shed logic panels from main plant battery board 1 to main plant battery board 3. The inspectors reviewed criteria in licensee procedures and modification instructions to verify that the risk-significant plant modification was developed, reviewed, and approved per these procedure requirements.

b.2 Design Change Notice (DCN) 51018

The inspectors reviewed the permanent plant modification to the Unit 2 accident signal logic to support the restart of Unit 1. The intent of this DCN in conjunction with DCN 51016 was to modify the Unit 2 common accident signal logic. To support the restart of Unit 2, the accident signal logic inputs for Units 1 and 3 were disabled by plant procedures and DCN H2735. Prior to the restart of Unit 3, the accident signal logic inputs for Units 2 and 3 were modified by DCN W20217 to ensure the diesels were able to support the required emergency core cooling system loads in the event of a spurious accident signal from the non-accident unit. The initiation of the Common Accident Signal (CAS) logic from the Core Spray (CS) system initiation logic results in the starting of all eight diesel generators (DGs) (the four Unit 1 / 2 DGs and the four Unit 3 DGs). The diesel generators of the non-accident unit are required to supply power to the common systems, such as the emergency equipment cooling water, the standby gas treatment , and the control room emergency ventilation needed to support the accident unit. DCN W20217 did not address the impact of Unit 1 operation, because the accident signal logic for Unit 1 remained disabled. To support the restart of Unit 1, the accident signal logic for Unit 1 and Unit 2 will be modified by DCN 51016 and DCN 51018 respectively. These modifications ensure that consequences of the various sequences of logic operation due to the occurrence of real accident signals and credible spurious accident signals during the combined operation of Units 1, 2 and 3 will be mitigated. Part of the planned modifications is to eliminate the redundant start logic. The new logic will have a Division I logic initiation activate only Division I equipment and a Division II logic initiation activate only Division II equipment. This in conjunction with other changes will ensure that the minimum emergency core cooling system requirements are met. There are no changes to the Unit 3 accident signal logic required to support the restart of Unit 1. The implementation of DCN 51016 and DCN 51018 will create a significant operational difference between Unit 3 and the other units. The implementation of DCN 51017 will modify the Unit 3 accident signal logic to minimize these operational differences. The inspectors reviewed criteria in licensee procedures and modification instructions to verify that the risk-significant plant modification was developed, reviewed, and approved per the procedure requirements.

b.3 Design Change Notice (DCN) 51216

The inspectors reviewed the permanent plant modification to the 480V AC distribution system in the reactor building. The intent of this DCN was to implement the following: Replace, reroute, retag, and/or abandon 480 volt cables to provide documentation of qualified cable installation for 10 CFR 50.49, Environmental Qualification (EQ), and Post Accident Monitoring (PAM) requirements; to ensure that cable ratings are acceptable for voltage drop, ampacity, and/or short circuit requirements; and to correct Appendix R, General Specifications G-38, and electrical separation concerns. Among the other items within the DCN were the following: Replace breakers that require changes as determined by supporting design calculations to protect cables and alleviate inadvertent trips; replace existing transformers TS1A and TS1B, power supplies for 480V shutdown boards 1A and 1B, to correct electrical overload, PCBs, and EQ concerns; remove all four Low Pressure Coolant Injection (LPCI) motor-generator sets; and abandon-in-place 480V Reactor Motor Operated Valve (RMOV) boards 1E and 1D and transfer their loads to the respective Division I and II RMOV boards 1A and 1B. The inspectors reviewed criteria in licensee procedures and modification instructions to verify that risk-significant plant modification was developed, reviewed, and approved per the procedure requirements.

b.4 Design Change Notice (DCN) 51238

The inspectors reviewed the permanent plant modification to Core Spray (CS) system in the reactor building. The intent of this DCN was to modify the instrumentation used in the CS system. The following were part of the modification: Replace obsolete equipment that replacement and spare parts are no longer available to support upkeep; replace various flow transmitters, flow switches, and pressure switches with instruments that meet environmental equipment qualification requirements; ensure that new equipment will meet the 120 percent power uprate, the 24 month refueling cycle, and the use of hydrogen water chemistry requirements; and the refurbishment of various instrumentation panels which include the replacement of isolation valves, equalization valves, drain valves, and instrument tubing. The inspectors reviewed several equipment changes including: Replacement of obsolete Barksdale Controls type B2T-A12SS pressure switches with type B2T-M12SS switches; replace Robertshaw type 613B pressure indicators with Ashcroft type 1379 indicators; replace Mercoid type DA-7043-804-R21E pressure switches with Class 1E environmentally qualified Static-O-Ring type 5N6-B3-U8-C1A-JJTTNQ switches; replace obsolete General Electric type 50-551032EAAL1 pressure transmitter with a class 1E environmentally qualified Rosemount type 1153GF8TB transmitter; replace ITT-Barton type 289A flow switches with Class 1E environmentally qualified Static-O-Ring type 103AS-B212-NX-JJTTX6 switches; and replace General Electric type 50-555111BDAA3PBJ flow transmitters with

Class 1E environmentally qualified Rosemount type 1154DP5RB transmitters. The inspectors reviewed criteria in licensee procedures and modification instructions to verify that risk-significant plant modification was developed, reviewed, and approved per the procedure requirements.

c. Conclusions

Review of Unit 1 modification design packages associated with four DCNs concluded that design changes were appropriately developed, reviewed, and approved for implementation per procedural requirements.

E1.2 Permanent Plant Modifications (71111.17, 37550)

a. Inspection Scope

The inspectors reviewed permanent plant modifications for 500 KV switchyard breakers, Reactor Protection System (RPS), and replacement of fuses. The inspectors evaluated the adequacy of the modification and observed field work to verify the design basis, licensing bases, and TS-required performance for the system had not been degraded as a result of the modifications.

b. Observations and Findings

b.1 DCN 51476, Replace Power Circuit Breaker (PCB) 5204/Install New PCB 5208

The inspectors reviewed permanent plant modification activities associated with DCN 51476, Replace Power Circuit Breaker (PCB) 5204 and install new PCB 5208, to make changes to the 500 KV switchyard breaker 5204 and add new 500 KV switchyard breaker 5208 and motor operated disconnect (MOD) 5209. The intent of this DCN is to replace Breaker 5204 to address maintenance and reliability issues and to install new breakers to convert the 500 KV switchyard, west point line from a “breaker and half” configuration to a “double breaker” configuration.

The inspectors observed portions of the installation of the permanent plant modification to pull cables to relay boards, 0-PNLA-9-22 and 0-PNLA-9-23, per WO 03-005851-15 and terminate cables in relay boards, 0-PNLA-9-22 and 0-PNLA-9-23, per WO 03-005851-16.

b.2 DCN 51080, Reactor Protective System

The inspectors reviewed permanent plant modification activities associated with DCN 51080. This modification will implement changes to the RPS System and consists of alarms, relay trip logic, switches, and other components necessary to initiate a scram to the reactor core. The inspectors observed portions of the installation of the permanent plant modification to replace General Electric Type HFA relays in Control Panel 9-15 located in Unit 1 Instrument Room.

b.3 DCN 60073, Replace fuses in 480V Turbine Building Motor Operated Valve (TMOV) Board 1C

The inspectors reviewed permanent plant modification activities associated with DCN 60073, Replace Fuses in 480V Turbine Building Motor Operated Valve (TMOV) Board 1C  This DCN replaces quality related and non-safety related fuses in the Unit 1 Turbine Building with a like-for-like equivalent fuse.  In addition, this DCN performs drawing changes for these fuses and is intended to eliminate discrepancies between installed fuses and design output documents as part of the BFN Fuse Control Program for Unit 1

The inspectors observed portions of the installation of the permanent plant modification in conjunction with WO 03-024401-75 to replace control power and breaker logic fuses for the normal and alternate feeder breakers for TMOV Board 1C.

c. Conclusions

Modification activities associated with 500 KV switchyard breakers, Reactor Protection System (RPS), and fuse replacement activities were performed in accordance with the documented requirements.

E1.3 Cable Installation and Cable Separation Special Program Activities (37550)

a. Inspection Scope

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

The inspectors reviewed the Unit 1 special programs addressing the issues of cable installation and cable separation with the intent of determining whether they are the same as the corresponding programs for Unit 3.  Selected areas of the plant were inspected to identify a sample of conduits having attributes indicating they should have been addressed within the cable installation special program.  Some independent verifications were made related to the issue of cable jamming in conduit during the pulling-in process and the issue of 600 V power cables being damaged by using a condulet as a pull point where proper bend radius could not be maintained.

b. Observations and Findings

The cable installation special program for Unit 1 has nine sub-programs, and each of these is defined by a "calculation."  For example, four of the sub-programs are defined in Calculation No. EDQ1 999 2003 0015, Analysis of Unit 1 Cable Installation - Miscellaneous Issues. The methodology for corresponding special programs implemented on Unit 3 is described in a report titled, “Methodology Report for Cable Installation Issues - Browns Ferry Nuclear Plant Unit 3 - Bechtel Job No. 21042" which was approved on December 14, 1992. The inspectors found that the Unit 1 programs and the Unit 3 programs for the cable installation and cable separation were essentially the same. In general, the engineering work for these special programs has been completed, i.e. walkdowns, design document reviews and preparation of design change packages. Actual physical implementation of the design change packages had just begun at the time of this inspection.

As stated in the scope section, the inspectors made walkdowns to identify a sample of conduits having attributes indicating they should have been addressed within the cable installation special program. Eleven conduits were identified: B78, 1B7-B1, 1PL5257-II, 1PL5232-I, 1PL5233-I, 1V2799, 1V2824, 1PL762, 1PL5932-II, 1PL5805-II, and 1PL5808-II. The inspectors later observed that the special program calculation correctly addressed these conduits. The following plant areas were included in the walkdown:

- Control bay 1C hallway
- Cable spreading room
- 4160 V shutdown board room A
- 480 V shutdown board room 1A, EL 621
- Air handling unit 1A room
- Reactor building EL 593 at the R1, R2, P-line

During the walkdown, the inspectors observed that a few conduit labels were incomplete or illegible, and the licensee initiated PER 60893 to address the observation.

From the cable mark number/cable description list, the inspectors selected two cables (as a sample for inspection) having a diameter that is a concern for jamming in the conduit. Further review indicated that these particular cable types were not installed at Browns Ferry, therefore no problem with the special program was identified in this area. The Mark Nos. selected were WPH-1 and WPK.

The inspectors requested and reviewed a list (by cable number) of all safety-related cables used on 600 V systems having a conductor size 300 MCM or larger. This list was compared to the licencee’s walkdown listing for cables related to the issue of condulet associated with pull points for large 600 V cable. The two lists matched indicating the licensee had identified all the potential problem cables in the special program for this issue.

c. Conclusions

Inspection of the methodology for the Cable Installation and Cable Separation special programs, together with performance based inspection in the area of cable installation issues, led to the conclusion that implementation of these programs is proceeding in accordance with licensee commitments.

E1.4 Flexible Conduit Special Program Activities (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, documented walkdown inspections of installed flexible conduit, and correction of any conduits not meeting the criteria.

b. Observations and Findings

The inspectors observed the licensee had completed walkdowns of Unit 1 flexible conduits containing safety-related cable. The program called for inspection of 100 percent of all safety-related flexible conduit, except for those conduits already identified for re-work or abandonment. 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.

The inspectors examined samples of the specific type of flexible conduit installed in the plant and the manufacturer's engineering application information was also reviewed. The inspectors confirmed the bending radius shown on the manufacturer's data sheet was consistent with the minimum bend radius specified in General Engineering Specification G-40, Installation, Modification and Maintenance of Electrical Conduit, etc. In addition, the inspectors examined a small sample of installed flexible conduits focusing on the concerns of this special program and did not identify any issues.

c. Conclusions

Inspection of the methodology for the Flexible Conduit special program led to the conclusion that implementation of this program is proceeding in accordance with licensee commitments.

E1.5 Cable Ampacity Special Program Activities (37550)

a. Inspection Scope

The inspectors reviewed the licensee's corrective actions that were developed and implemented for resolution of Unit 1 cable ampacity issues. The corrective actions were evaluated for conformance with the cable ampacity program that was used for resolving cable ampacity concerns prior to re-start of Units 2 and 3. Additionally, differences in the cable ampacity program used for re-start of Units 2 and 3 and that being used for Unit 1 were evaluated for technical adequacy.

b. Findings and Observations

TVA in their letter dated December 13, 2002, " Browns Ferry Nuclear Plant (BFN)-Unit 1- Regulatory Framework for the Restart of Unit 1", provided TVA's proposed regulatory framework for the restart of Unit 1. The licensee stated that TVA's plan for the restart of Unit 1 was based on the regulatory requirements, corrective action special programs, commitments, technical specification improvements, and internally identified deficiencies and concerns that were resolved prior to Units 2 and 3 restart. The licensee also stated that TVA's program for the resolution of the ampacity issue was completed during Unit 3 restart using the Unit 2 precedent.

The inspectors reviewed the corrective actions developed and implemented for resolving Units 2 and 3 cable ampacity issues and determined the program consisted of three evaluations which were performed to establish adequate cable ampacity. Phase 1 evaluation used the acceptance criteria specified in TVA electrical design standard DS-E-12.6.3. If a cable failed to meet the acceptance criteria of the design standard a phase IIa and IIb evaluation was performed at which time additional inputs concerning the cable loading based on actual operational loads and load diversity was incorporated in the evaluation. Cables that failed to satisfy the acceptance criteria of the phase IIa and IIb evaluations were further evaluated using the phase III acceptance criteria. The phase III evaluation took credit for time diversity as well as the loading diversity of the phase IIb evaluation. Cables which failed to meet the acceptance criteria of the phases IIa, IIb, and III evaluations were replaced in accordance with the ampacity requirements delineated in design standard DS-E12.6.3. A detailed description of the cable ampacity program implemented for restart of Units 2 and 3 is documented in NRC Inspection Reports 50-260/89-59 dated February 23, 1990 and 50-259, 260, 296/94-35, dated March 16, 1995.

Based on review of the licensee's corrective actions for resolving the cable ampacity issue prior to Unit 1 restart, the inspectors determined that phases IIa, IIb, and III evaluations would not be used. A phase I evaluation using the acceptance criteria of design standard DS-E12.6.3 would be used for establishing cable ampacity requirements for all safety related medium and low voltage power cables required to support Unit 1 restart. Cables that failed to meet the acceptance requirements of DS-E12.6.3 would be replaced. Additionally, an administrative decision was made by TVA management to replace all the safety-related cables in the drywell which included 25 power cables. The scope of the corrective actions developed by the licensee for resolving the cable ampacity issue for Unit 1 is as follows:

- All safety-related medium voltage and low voltage (V5 and V4) power cables required for Unit 1 restart will be identified along with their raceway type, eg. conduit vs. tray.

- All safety-related power cables in trays will be will be sized and replaced using the requirements of DS-E12.6.3.

- All safety-related power cables in conduit will be evaluated using the requirements of DS-E12.6.3.

- Safety-related control (V3) cables will be evaluated in order to identify those cables that exceed the 10 amperes load current criteria for control cables. Cables not meeting this acceptance criteria will be resized using the requirements of DS-E12.6.3.

The inspectors reviewed design standard DS-E12.6.3 in order to evaluate the acceptance criteria specified for sizing cable conductors. Based on this review, the inspectors determined that conductor sizes are calculated on the basis of either an allowable load current or a minimum required ampacity given a specific load current. The technical bases for the cable ampacity ratings were identified and are traceable back to accepted industry standards and engineering practices. Additionally, correction factors have been incorporated in the evaluation based on the details of the cable installation. Cable ampacity calculations which incorporated the acceptance criteria of DS-E12.6.3 were prepared to determine the cable ampacity ratings of replacement cables. Samples of these calculations were also reviewed by the inspectors. The calculations were identified as design inputs to design change notices that will be used for installing the following replacement cables:

- 98 safety-related power cables in trays will be replaced and re-routed in new cable trays or conduits.

- 8 safety-related power cables in conduit will be sized using the acceptance requirements of DS-E12.6.3 and will be replaced.

- From a total of approximately 2400 control (V3) cables reviewed by the licensee, the load current of 12 cables exceeded the 10 amperes load criteria and the cables were re-classified as power (V4) cables. The cables were re-analyzed for ampacity requirements using the criteria of DS-E12.6.3 and re-routed in V4 raceways.

A listing of the cable ampacity calculations reviewed are documented in the attachment to the report.

The inspectors performed a walk down of Unit 1 reactor building in order to assess the scope of design changes involving installation of cable trays and replacement cables. The inspectors were informed by TVA engineers that the low pressure coolant injection motor generator sets planned to be replaced and that scope of the design changes included transfer of electrical loads from Unit 1 480 V RMOV Boards 1D and 1E to the respective Unit 1 Division I and II 480 V RMOV Boards 1A and 1B. The replacement cables would be sized to meet the ampacity requirements of design standard DS-E12.6.3. The inspectors performed a field inspection of the installed equipment and cable trays that were within the scope of the plant modification in order to assess the scope of design changes involving installation of cable trays and replacement cables. Technical specification change request number 427, prepared for deletion of the low pressure coolant injection motor generator sets, was also reviewed by the inspectors.

c. Conclusions

The inspectors concluded that the corrective actions developed by the licensee, for resolving the cable ampacity issue prior to Unit 1 restart, was different from the program used to support the restart of Units 2 and 3. TVA in their letter dated December 13, 2002, stated they may modify implementation of the ampacity program to incorporate efficiencies or experience gained from the restart of Unit 3 or changes in TVA's processes. The omission of phases IIa, IIb, and III evaluations was not determined to be a reduction of the licensee's commitment for resolving Unit 1 cable ampacity issue. The inspectors considered the use of the phase I evaluation for establishing cable ampacity in accordance with the acceptance requirements DS-E12.6.3, to be technically conservative when compared to the evaluations performed in phases IIa, IIb, and III for determining cable ampacity. The implemented corrective actions of replacing cables that failed to meet the acceptance requirements of the design standard were also considered adequate for resolving the cable ampacity issue related to the restart of Unit 1. No violations or deviations were identified.

E1.6 Intergranular Stress Corrosion Cracking (IGSCC) - Welding of Replacement Reactor Water Recirculation (RECIRC) System Piping (55050)

a. Inspection Scope

As part of the IGSCC Special Program, TVA is replacing the RECIRC system piping with Type 316NG material. The applicable Codes for this work are: (1) ASME Section XI, 1995 Edition, 1996 Addenda and (2) 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

See Inspection Reports 50-259/2003-09, 50-259/2003-10, 50-259/2003-11, and 50-259/2004-06 for documentation of previous inspections in this area. During the current inspection, the inspectors observed/reviewed the following to verify compliance with the applicable Codes listed above:

- For completed RECIRC System Loop A Welds RWR-1-001-001, -002, -003, -004, -024, and -025, and Loop B Welds RWR-1-002-012, -013, and -053, the inspectors visually inspected final weld surfaces (outside diameter (OD) only); reviewed Weld Data Sheets and Liquid Penetrant (PT) Examination Reports; and reviewed radiographic (RT) film (including Welds 003, 004, and 053 that the licensee rejected for lack of fusion and initiated repairs). The RT of Weld RWR-1-002-053 was for information (not final) prior to preparing the final weld surface for ISI.

- For RECIRC System Welds RWR-1-001-005 and RWR-1-002-048, the inspectors observed in-process welding and reviewed in-process Weld Data Sheets.

- The inspectors observed in-process PT examination of the final surface for Weld RWR-1-002-048 and the repair excavation surface for Weld RWR-1-001-004. The repair of Weld -004 was for lack of fusion identified by RT examination.

- For the RECIRC System welds listed above, the inspectors verified the use of qualified Detailed Welding Procedure Specifications (DWPSs) and certified welding materials. The procedures and materials were the same as those verified and documented in NRC Inspection Report 50-259/2004-06.

- The inspectors reviewed Welder Qualification Records, including continuity records, as applicable, for the seven welders who welded the RECIRC System welds listed above.

- Qualification Records for five QC Examiners (Welding and PT) who performed welding and nondestructive examinations (NDE) examinations of the RECIRC System welds listed above were reviewed.

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

  Penetrant - Lots 01H05K and 03K08K
  Remover - Lots 03B03K, 01G11K, 03F21K and 03F22K
  Developer - Lots 01K02K, 03H10K, and 02J15K

- In addition to observation of in-process welding and inspection 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

No violations, deviations, or significant problems were identified during this review of the licensee's Intergranular Stress Corrosion Cracking Special Program and pipe welding activities for Unit 1.

E1.7 Inservice Inspection - Review of Program (73051), Inservice Inspection (73753), 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 the licensee has completed, or has plans to complete prior to restart, ISI/PSI requirements in accordance with regulatory requirements and licensee commitments.

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

ISI/PSI Program

As detailed in TVA's letter to the NRC dated March 1, 1988, the first Unit 1 ten year ISI interval, which began August 1, 1974, is currently in its third period and will end one year following the restart of the unit from its extended outage. The licensee's ISI program provides the basis for completion of the first ten year inspection interval.

Pursuant to 10 CFR 50.55a, TVA Surveillance Instruction (SI) 1-SI-4.6.G, Browns Ferry Unit 1 ISI Program, Revision 5, adopts the 1995 Edition with Addenda through 1996 of ASME Section XI for the remainder of the first ten year ISI Interval, with the exception of ISI weld selection. Weld selection for ISI activities remains in accordance with ASME Section XI 1974 Edition with Addenda through Summer 1975. The licensee's ISI plan was revised October 11, 2002, to incorporate the specified provisions of ASME Section XI, 1995 Edition with Addenda through 1996. The licensee notified the NRC of this change by letter dated November 8, 2002. Although most of the examinations required to be performed during the first ten year inspection interval were completed during the first two periods, the licensee is in the process of identifying the remaining examinations required to complete the first inspection interval, and plans to complete these examinations prior to the restart of Unit 1.

In addition to completing the examinations required for the first interval, the licensee committed by letter dated November 8, 2002, to conduct a re-baseline examination of a sample of Classes 1, 2, and 3 components, which are not being repaired or replaced, in order to ascertain their structural integrity given the extended shutdown of Browns Ferry Unit 1. The re-baseline examination of selected components will be performed in accordance with the percentages stated in SI 1-SI-4.6.G and the ISI acceptance criteria of Section IWX-3000 of the 1995 Edition through Winter 1996 Addenda of ASME Section XI. The scope of the re-baseline effort is as follows:

- Class 1: 25% of piping welds accessible without removal of supports or permanent plant features for those systems not being replaced, 100% of component supports, Reactor Pressure Vessel (RPV) head and longitudinal shell welds, 100% of RPV bolting, and 100% of accessible RPV interior and interior attachments.

- Class 2: 7.5% sample of pipe welds in each system and 100% component supports.

- Class 3: 100% of component supports including attachments.

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.

In addition, the inspectors reviewed and discussed the application of code cases, potential or pending relief requests, the ISI examination plan, and any augmented examinations that need to be complete prior to the restart of Unit 1 to ensure the licensee’s ISI program is in conformance with regulatory requirements and licensing commitments.

Observation/Review of ISI Activities

The inspectors observed/reviewed a sample of examination activities which were conducted as part of the licensee’s re-baseline effort. 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 a Magnetic Particle (MT) dry powder examination of Core Spray (CS) 16 inch diameter carbon steel pipe weld attachments CS-1-33 and CS-1-29

- Observed the calibration of Ultrasonic (UT) equipment and the UT examination for welded attachments CS-1-33 and CS-1-29

- Reviewed nondestructive examination (NDE) reports of the following completed Liquid Penetrant (PT) and Visual (VT) examinations:

| | |
|---|---|
| PT Report R-068 | Unit 1 Feedwater Component FW-1-SSA5-1A |
| VT Report R-226 | Unit 1 Control Rod Drive Component 1-47B468-261 |

The inspectors reviewed qualification and certification records for examiners, and NDE procedures for the above mentioned re-baseline examinations. Verification of personnel qualification and certification included reviewing documentation for two Level II VT examiners, three Level II MT examiners, one PT Level II examiner, two Level II UT examiners and two Level III examiners. The inspectors also verified ASME Code Section XI compliance for equipment and consumables by reviewing documentation for the items listed below:

Equipment/Consumables:

UT: Scope #: E36302
1/4“ 5 MHz 45 Degree Transducer #00W415
½" 2.25 MHz 45 Degree Transducer #00W41R
Straight Beam 4MHz Transducer #04194
Couplant Ultragel Batch #03125
Carbon Steel Calibration Block #WB78
Rhombus Calibration Block #5100
Thermometer #561993

MT: Yoke #11987
10 lb Test Weight #E06218
Visual Card #057

Calibration of NDE equipment:
UT Scope
UT Transducers
MT Yoke

The inspectors also reviewed re-baseline examination results for components with rejectable indications, and for examinations that identified indications located outside of the Section XI examination boundary. The licensee had identified unacceptable linear indications during a MT examination on two Class 2 main steam welds, DSMS-1-13 and DSMS-1-30. The licensee is planning to obtain a boat sample of the indications for further examination and determination of the cause. The inspectors verified the required examination sample expansion resulting from identification of the rejectable indications was conducted in accordance with ASME Section XI, 1995 Edition with Addenda through 1996. The required sample expansion had resulted in identification of linear indications outside the Section XI examination boundary on welds DSMS-1-43 and DSMS-1-44. The licensee issued PER 04-003621-000 to address all of the indications and is currently evaluating the significance of the indications. At the time of the NRC inspection, the root cause investigation had not been completed.

Examination of approximately 66 percent of the Class 1 piping welds and 57 percent of the Class 2 piping welds selected as the re-baseline sample has been completed. The re-baseline examinations of other systems, conducted to date, have not identified any other rejectable indications. Inspector Follow-up Item (IFI) 50-259/04-07-01, Main Steam Line Linear NDE Indications, will be identified to follow resolution of the linear indication in the Main Steam piping.

The inspectors discussed several augmented examinations that are required by licensing commitments to be completed prior to the restart of Unit 1. These augmented examinations include requirements from the Technical Requirements Manual (TRM), Surveillance TSR 3.4.3.3, which addresses the evaluation of corrosion damage of piping components exposed to residue from the March 22, 1975 fire. Other augmented examinations include the reactor vessel interior examinations and weld inspection for pipe whip protection per the TRM, Surveillance TSR 3.4.3.2. The licensee plans to complete all required augmented examinations prior to returning any of the affected systems to service.

c. Conclusions

The inspectors determined that the licensee's (ISI/PSI) activities and plans to date, met applicable code requirements and licensing commitments. An IFI is opened to follow resolution of the linear indication in the Main Steam piping. No violations or deviations were identified during this ongoing review of the licensee's Inservice/Preservice Inspection Program for the restart of Browns Ferry Unit 1.

E1.8 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed licensee procedure SPP-9.5, Temporary Alterations and a temporary alteration associated with closure of sluice gates on the RHRSW supply to ensure that procedure and regulatory requirements were met. This temporary alteration was installed under Temporary Alteration Control Form (TACF) 0-04-004-023 to support work activities associated with Unit 1 Condenser Circulating Water (CCW) - System 27. 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

TACF 0-04-004-023, was issued to revise Note 7 on Drawing 1-47E858-1 to allow the closure of two sluice gates on the same RHRSW supply during two unit operation. Note 7 on the drawing stated: To ensure proper RHRSW pump operation, the following limitations shall apply. This was followed by notes 7A, 7B, and 7C. Note 7A stated, during three-unit operation, none of the six sluice gates of the three RHRSW pump suction supply pit supply lines may be closed at any time; Note 7B stated, during two-unit operation, two gates, but not two on the same supply line tee, may be closed at any time; and Note 7C stated, during one-unit operation, any three gates may be closed at any time. The TACF revised Note 7B and added an additional Note 8. The revised Note 7 B stated: During two-unit operation, two gates, but not two on the same supply line tee, may be closed at any time, except as stated in Note 8. The additional Note 8 stated as follows: To ensure capability to safely shutdown and proper RHRSW pump operation, following a down stream dam break with two units operating and two sluice gates closed on the same RHRSW pump pit supply line, all of the following limitations shall apply:

- Each unit shall be operated at less then or equal to 3458 Mwt
- This alignment shall only be utilized while river temperature is less then or equal to 91 degrees F
- RHRSW flow is restricted to less then or equal to 4100 gpm per pump, 2 pumps per unit
- EECW is reduced to two pumps at less then or equal to 4500 gpm per pump prior to starting RHRSW pumps

The inspectors reviewed the 10 CFR 50.59 screening questions and the evaluation. The evaluation concluded the proposed TACF did not require NRC approval. The evaluation also concluded that the proposed TACF could be implemented per plant procedures without obtaining a License Amendment.

c. Conclusions

The inspectors determined that the temporary alteration and 10CFR 50.59 evaluation associated with closure of sluice gates on the RHRSW supply did not create any significant impact on the operability of equipment required to support operations of Units 2 and 3. No violations or deviations were identified.

E1.9 System Return to Service Activities (37550)

a. Inspection Scope

The inspectors reviewed and observed portions of the licensee's system return to service (SRTS) activities. The licensee has recently initiated the first SRTS activities for the Unit 1 recovery effort, focusing mainly on support systems. The SRTS activities were performed in accordance with Technical Instruction 1-TI-437, System Return to Service (SRTS) Turnover Process for Unit 1 Restart, associated with systems System 8, Turbine Drains and Miscellaneous Piping, System 39, CO2 Storage and Fire Protection/Generator Purge, System 40, Station Drainage, and System 79, Fuel Handling.

b. Observations and Findings

The SRTS process consisted of three parts as follows: The System Plant Acceptance Evaluation (SPAE), which consists of design changes, engineering programs analysis, drawings, calculations, corrective action items, and licensing issues; the System Pre-Operability Checklist (SPOC) I, which consists of the completion of items required for system testing; and the SPOC II, which consists of the completion of system testing and the completion of items that affect operational readiness. Instruction 1-TI-437 allows for the omission of the SPOC I process for various systems, such as System 40, Station Drainage, and go directly to the SPOC II process. The inspectors reviewed and observed portions of the licensee's SRTS activities for the following:

- System 8, Turbine Drains and Miscellaneous Piping, which included the SPAE process
- System 39, CO2 Storage and Fire Protection / Generator Purge, which included the SPAE process and the SPOC I process
- System 40, Station Drainage, which included the SPAE process
- System 79, Fuel Handling, which included the SPAE process and the SPOC I process

The activities observed for the systems include meetings to discuss the SRTS status of the systems, the status of the SPOC I checklists, the status of the SPOC II checklists, and portions of the systems in the plant. The activities for system 39 included the completion of the SPOC I process and the system was placed in the status control category by operations on April 26, 2004.

The inspectors also reviewed selected PERs initiated during the SRTS process as follows: PER 47372 documented that for System 39, CO2 Storage and Fire Protection / Generator Purge, design change ECN-51724 was not shown on the Open Item Punchlist; PER 41617 documented that for System 79, Fuel Handling, design change

DCN-51367 was listed as open against Unit 1 when this DCN only affected Units 2 and 3 and is closed; and PER 47538 which documented the failure to recognize that there are programmatic items involving certain systems that need to be addressed in order to support Unit 1 in an operating condition. Specifically this PER stated that the guidelines for system scoping for Unit 1 recovery indicated that portions of the current inservice systems on Unit 1 were outside the scope of the Unit 1 SRTS process. Among these systems were the following: System 24, Raw Cooling Water; System 32, Control Air; System 33, Service Air; and System 78, Spent Fuel Pool Cooling. The inspectors evaluated the licensee's identification and proposed corrective actions for these issues and determined they were appropriate and consistent with the intent of the SRTS process.

c. Conclusions

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

E1.10 Power Service Shop Activities (71111.17 and 37550)

a. Inspection Scope

The inspectors visited the licensee's machine shop and electrical repair shop at the Power Service Shop (PSS) in Muscle Shoals and observed work in progress on Unit 1 components.

b. Observations and Findings

The licensee is performing initial QC receipt inspection of replacement piping components and weld joint preparation activities for replacement piping at the PSS. The inspectors observed in-progress machining activities performed on the Recirc A loop five port fitting. The inspectors observed housekeeping, PSS work control processes, PSS NDE procedures, verification of weld joint geometry, and verified that work documents incorporated technical and quality requirements included in the contract. Machining was performed in accordance with WO 02-010314-078. The assigned machinist and machine shop foreman were knowledgeable of technical and quality requirements. The inspectors noted that work documents included specific hold points for QC inspections by PSS and Browns Ferry Unit 1 QC personal. Additional hold points for inspections specified by the American Nuclear Insurers (ANI) representative were also included.

The inspectors also visited the electrical repair shop and observed work locations for Unit 1 recirc motors, RHR pump motors and various MOV actuators which were awaiting refurbishment at the PSS. However, no actual work was ongoing at the electrical repair shop. All Unit 1 components were segregated and labeled. MOV actuators were located within a radiological control area with restricted access and contained adequate warning signs.

c. Conclusions

Ongoing work in the PSS machine shop involved a high level of professionalism. Work documents included applicable technical and quality requirements. No violations or deviations were identified.

E1.11 Drywell Steel Platforms Special Program Activities (62002 and 37550)

a. Inspection Scope

During investigations performed in the 1980's by the licensee and NRC related to restart of Unit 2, numerous deficiencies were identified in design and construction of safety-related structural steel platforms. These included cracking of clip angles which connect structural members, failure to construct the platforms in accordance with design documents, deficiencies in welding (primarily undersized fillet welds), seismic design issues, and configuration management issues ( i.e., failure to control addition of more loads to platforms). The Unit 1 drywell structural steel platforms have been redesigned to meet current design criteria. Design changes were issued to modify the structural steel platforms to correct the deficiencies. The majority of the original structural steel members for the platforms on elevation 563 and 584 were removed and replaced. The connections on the platforms on elevations 604, 616, and 628 were modified by replacing bolts, addition of clip angles and stiffeners, and reinforcement of existing welds. The modified structural platforms are intended to meet current design criteria and have a design margin for addition of future loads, if necessary.

The licensee's commitments for resolution of issues associated with the drywell structural steel platforms are stated in TVA letter dated December 13, 2002, Subject: Browns Ferry Nuclear Plant - Unit 1 - Regulatory Framework for the Restart of Unit 1. The letter references previous commitments for restart of Units 1 and 3 were stated in a letter dated July 10, 1991, Subject: Regulatory Framework for the Restart of Units 1 and 3, and NRC approval of the licensee's plans in a letter dated April 1, 1992. Design criteria for design and seismic qualification of the drywell structural steel platforms were submitted to NRC in TVA letters dated June 12, 1991, June 13, 1991, and February 6, 1992. Acceptance of the licensee's design criteria for the structural steel platforms by NRC is documented in a Safety Evaluation Report dated July 13, 1992, Design Criteria for Lower Drywell Steel Platforms and Miscellaneous Steel. The modifications for the Elevation 563 and 584 drywell structural steel platforms were previously inspected during inspections documented in NRC Inspection Reports 50-259/2003-09 and 50-259/2003-11.

The inspectors reviewed design calculations, design drawings, work orders issued to implement modifications to the drywell structural steel platforms, work control instructions, and quality control inspection procedures. During the current inspection, the inspectors examined selected modifications to the Unit 1 drywell structural steel platforms on elevation 563 between azimuth 351° to 15°; on elevation 584 between azimuth 330° to 351°; on elevation 604 between azimuth 15° to 45° and azimuth 145° to 165°; on elevation 616 between azimuth 295° to 15°and azimuth 115° to 195°; and on elevation 628 between azimuth 299° to 0° to verify that modifications were completed in accordance with design requirements, which included instructions in the work order packages, and the design drawings.

During the walkdown inspection, the inspectors examined the following attributes for compliance with the requirements shown on the design drawings: member sizes, configuration, installation of cover plates on radial beams, weld sizes, type, and length, connection details, and recent installation of correct type bolts in existing connections.

The inspectors also reviewed completed quality records to verify compliance with applicable requirements. Records reviewed included: inspection records for installation of new structural steel bolts, weld travelers, weld maps, personnel qualification records, and material certification records. Specific records examined included detailed welding procedure specification (DWPS) SM11-B-1-N, Revision 1, and supporting procedure qualification record (PQR), GT-SM11-0-3C; welder qualification records (WQR) and welder performance qualification records (WPQRs) for ten welders qualified to weld to the above DWPS, and verified compliance to applicable welder requirements; certified material test reports (CMTRs) for 1/8" E 7018 Carbon Steel Covered Electrode - Heat 93051; 3/32" E 7018 Carbon Steel Covered Electrode - Heat 40511; 3/32" E 7018 Carbon Steel Covered Electrode - Heat 148157. In addition the inspectors reviewed qualification records for two QC inspectors (Post-Weld Visual Examination) and four Pre-Weld Visual Examination qualified individuals (QI), who had signed off on a sample of weld data sheets selected for review.

b. Conclusions

Based on records review and observation of recent installations of structural steel the inspectors concluded that Drywell Steel Platform Special Program activities continued to satisfy applicable requirements. No violations or deviations were identified.

E1.12 Containment Coatings Special Program Activities/Repairs to Coatings in Unit 1 Torus (62002 and 37550)

a. Inspection Scope

The inspectors reviewed the licensee’s program to identify unqualified coatings in the Unit 1 Torus, and the planned corrective actions to remove unqualified coatings and apply new Service Level I coatings qualified to ANSI standards. The inspectors reviewed TVA Specification G-55 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 reviewed the exceptions to G-55 and the specification change process to verify changes to Specification G-55 were controlled in accordance with TVA procedure NEDP-10 and NRC design change controls specified in 10 CFR 50, Appendix B, Criterion III. The inspectors examined storage of protective coatings material, controls to ensure shelf life was not exceeded prior to application of the materials, and the paint shop. The inspectors also examined the testing program (including the test panels) used to qualify the coating applicators, and the qualification records of the qualified applicators to verify the licensee's program for qualifying applicators complied with ASTM D4228, Standard Practice for Qualification of Journeyman Painters for Application of Coatings to Steel Surfaces of Safety-related Areas in Nuclear Facilities. In addition, the inspectors reviewed TVA Specification G-29, Part B, Standard Materials Specifications Section 2A, to verify the controls for purchase and use of consumables such as paint removers, masking tape, and paint thinners were in accordance with applicable requirements.

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.

b. Conclusions

Based on review of corrective action documents, acceptance criteria, and qualification records for identification, removal, and repair of unqualified coatings in the Unit 1 Torus the inspectors concluded that Containment Coatings Special Program activities satisfied applicable requirements. No violations or deviations were identified.

E1.13 System Cleanliness Verification (37550)

a. Inspection Scope

The inspectors verified that the licensee was following the prescribed program established for cleanliness verification of Unit 1 safety-related equipment. The inspectors reviewed licensee Technical Instruction 1-TI-474, Cleanliness Verification Program, to verify that system cleanliness inspections and monitoring were completed as required.

b. Observations and Findings

The licensee had recently implemented the System Cleanliness Verification Program to support the current modification phase of the Unit 1 restart effort. This program is intended to replace the previous Equipment Layup Program that had been in effect since the unit was shutdown. Transition to the newer program was still in progress at the start of the inspection period. Under the new program, the assigned system and component engineers, along with chemistry personnel, would perform a series of inspections of Unit 1 systems to identify any system degradation or special requirements to support Unit 1 recovery. The inspectors held discussions with the System Cleanliness Coordinator and attended several routine System Cleanliness Status Meetings to determine the adequacy of this program. System layup tags will remain in place and be under control of the assigned system engineer until system alignments would be performed as required by the SPOC process.

The inspectors observed conditions in the ECCS ring header to evaluate the adequacy of system cleanliness inspections as specified by the applicable procedure. This was performed by viewing a video tape exam of the entire ring header interior surface. The video showed minimum corrosion of the ring header interior surface and the presence of no debris or foreign material. The inspectors also reviewed a sampling of PERs to verify that equipment problems were identified and corrected as required by procedure SPP-3.1, Corrective Action Program.

c. Conclusions

No violations or deviations were identified. The licensee has continued to utilize the recently implemented System Cleanliness Verification Program to provide comprehensive inspections of systems to continue the identification of degradation or special requirements to support the Unit 1 recovery.

## 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 observed licensee Nuclear Assurance (NA) oversight activities associated with machining of sections of Unit 1 recirc piping at the Power Service Shop (PSS). The inspectors observed NA inspection of a Browns Ferry warehouse facility located on the Muscle Shoals Reservation nearby the PSS. This warehouse is used for bulk storage of Unit 1 components awaiting receipt inspection or work at PSS. Also, the inspectors' review was to assess whether any issues were processed in accordance with licensee Procedure SPP-3.1, Corrective Action Program.

b. Observations and Findings

The inspectors reviewed the licensee's inspection plan prior to the scheduled NA observation visit at the PSS. The inspectors verified the licensee's inspection plan included specific guidance for review of housekeeping, PSS work control processes, PSS NDE procedures, verification of weld joint geometry, and that work documents incorporated technical and quality requirements included in contract. The inspectors accompanied licensee NA assessors to the PSS and observed work in progress on Recirc Loop A five port fitting. The assessors were knowledgeable of the applicable work document requirements and PSS programs and work processes. The inspectors noted that work documents included specific hold points for QC inspections by PSS and Browns Ferry Unit 1 QC personal. Additional hold points for ANI inspections were also included. The inspectors reviewed selected PSS Corrective Action Reports (CARs) to verify the adequacy of the PSS program for identification and resolution of problems.

During the inspection of the bulk warehouse facility the NA assessor concluded the facility satisfied Level C storage requirements with one exception. The concrete floor had an excess buildup of dirt and required cleaning. PER 64271 was issued to address this concern

c. Conclusions

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

## E8 Miscellaneous Engineering Issues (92701)

E8.1 (Closed) Apparent Violation 50-259, 260, 296/92-29-01: Apparent Violation of Plant Records.

NRC Information Notice 92-30 alerted licensees of a potential problem with falsification of plant records. In response to NRC concerns about falsified plant logs, the licensee had conducted an audit to determine if similar problems existed at Browns Ferry. Although no examples of falsification were found during the licensee's audit, several discrepancies were identified. These discrepancies were associated with failure by auxiliary unit operators to satisfy management expectations during performance of routine operator rounds. This apparent violation was opened pending review of the potential problem at Browns Ferry and other licensee facilities. Subsequent to this the licensee conducted additional training for non-licensed personnel to emphasize the importance of operator routines. In addition, the licensee implemented a computerized system to record and trend information during performance of operator routines. The inspectors considered the licensee's actions adequately addressed the concern. Further examples similar to the discrepancies previously identified during the above audit have not recurred since recovery of Units 2 and 3. Because the same procedures and processes are used on Unit 1 for performance of operator routines, no further action is required. Furthermore, because any implementation performance deficiencies would likely be detected by licensee oversight programs and have only minor consequences, this item meets the closure criteria established for Unit 1 recovery issues. This issue is closed for Unit 1.

E8.2 (Closed) LER 50-259/88-12: Battery Failure Concurrent With LOOP/LOCA Prevents Automatic Start of Residual Heat Removal (RHR) Pump

In March 1988, during a review of the 250V DC system design engineers identified that a condition existed that could prevent the automatic start of a RHR pump. The failure of a battery supplying logic power for one division of RHR would prevent one of the two pumps for automatic starting. When the loss of offsite power and loss of coolant accident occurs concurrent with a battery failure the logic to start the RHR pump will not function properly in that a RHR pump breaker could close onto a de-energized emergency bus. The breaker would immediately open and go into a lock out condition. This would prevent the automatic starting of the RHR pump. This would place the plant in an unanalyzed condition. The condition impacted all three units. The cause of the condition was a modification to the logic that was installed in 1977. There was an insufficient review of the design for single failure criteria prior to the installation. This item was closed for Unit 2 in inspection report IR 50-259, 260, 296/89-35 by implementing DCR 3549 through ECN E-2-P7136 and documented in Work Package 2182-88. This item was closed for Unit 3 in inspection report IR 50-259, 260, 296/95-60 when the inspectors verified the logic change to the RHR pump breakers. The inspectors reviewed DCN 51016A which included the design details for the ECCS accident logic modification and will correct the item for Unit 1. Therefore, because this item is effectively being tracked in the licensee's corrective action program, is being corrected identically to the Unit 3 solution with the same process and design change, and because any implementation performance deficiencies would likely be detected by licensee oversight programs and have only minor consequences, this item meets the closure criteria established for Unit 1 recovery issues. Because this problem was originally identified while the unit was shutdown and defueled and will be corrected prior to re-start, no violation of NRC requirements occurred. This issue is closed for Unit 1.

E8.3 (Closed) LER 50-259/86-18: Neutron Monitoring Surveillance Test Deficiencies

In April 1986, during a performance of an Intermediate Range Monitoring (IRM) surveillance instruction (SI) test it was determined that the test did not fulfill the Technical Specification (TS) requirements. Specifically, the TS required that the IRM high flux scram be functionally tested once per week, during refueling, and once per month. The TS also required the rod block function be checked once per month. The SI was performed with the IRM channel in the INOP mode, which in itself causes a trip signal. Therefore, the test was inadequate in that it did not actually verify the individual trip signals to completion for half scram and rod block. This LER was closed in Inspection Report (IR) 259, 260, 296/88-33 for Unit 2 in August, 1988, when the licensee installed an INOP mode bypass switch on the IRM channels. This allowed direct testing of the half scram and rod block functions. This LER was closed in Inspection Report (IR) 259, 260, 296/94-36 for Unit 3 in December, 1994, when the licensee initiated and implemented design change DCN 18726A. This DCN installed an INOP mode bypass switch on the Unit 3 IRM channels. The licensee initiated design change DCN 51079A, Nuclear Measurement Analysis and Control - Power Range Monitor (NUMAC-PRNM) Upgrade for Unit 1. Stage 3 of the DCN was initiated to install minor modifications to the Source Range Monitors and Intermediate Range Monitors. The inspectors reviewed DCN 51079A including PIC 61566 which was issued to implement the Stage 3 minor modifications for Unit 1. Therefore, because this item is effectively being tracked in the licensee's corrective action program, is being corrected identically to the Unit 3 solution with the same process and design change, and because any implementation performance deficiencies would likely be detected by licensee oversight programs and have only minor consequences, this item meets the closure criteria established for Unit 1 recovery issues. Because this problem was originally identified while the unit was shutdown and defueled and will be corrected prior to re-start, no violation of NRC requirements occurred. This issue is closed for Unit 1.

E8.4 (Closed) IFI 50-259/86-40-03: Intermediate Range Monitoring (IRM) Power Supply

In December 1986, as result of a review of General Electric Service Information Letter (SIL) 445, Intermediate Range Monitor (IRM) Fuse Failure, it was determined that design changes and procedure changes would be required for the Source Range Monitors (SRM) and Intermediate Range Monitors (IRM). The procedure changes were to require the functional testing of the SRMs and the IRMs. The design changes were to replace the 3/4 amp fuses in the IRMs with 1.5 amp fuses and to provide an IRM INOP trip to the reactor protective system in response to a loss of the negative 24V DC power supply to the IRMs. This item was closed in Inspection Report (IR) 259, 260, 296/89-35 for Unit 2 in July, 1989, when the licensee initiated design change DCN H1706A and installed the change thru work plan 2583-88. The procedure change was made to procedure 0-O1-57D, DC Electrical System Operating Instructions, which applied to all three units. The procedure change required the functional testing of the SRMs and the IRMs. This item was closed in IR 259, 260, 296/94-32 for Unit 3 in November 1994, when the licensee initiated and installed design change DCN W 18726A. The change to procedure 0-O1-57D was still in effect. The licensee initiated design change DCN 51079A, Nuclear Measurement Analysis and Control - Power Range Monitor (NUMAC-PRNM) Upgrade for Unit 1. Stage 3 of the DCN was initiated to install minor modifications to the Source Range Monitors and Intermediate Range Monitors. The inspectors reviewed DCN 51079A, including PIC 61566, which was issued to implement the Stage 3 minor modifications for Unit 1. Additionally, the inspectors reviewed revision 73 of procedure 0-O1-57D and observed the change requiring the functional testing of the SRMs and the IRMs was still in effect. Therefore, because this item is effectively being tracked in the licensee's corrective action program, is being corrected identically to the Unit 3 solution with the same process and design change, and because any implementation performance deficiencies would likely be detected by licensee oversight programs and have only minor consequences, this item meets the closure criteria established for Unit 1 recovery issues. Because this problem was originally identified while the unit was shutdown and defueled and will be corrected prior to re-start, no violation of NRC requirements occurred. This issue is closed for Unit 1.

E8.5 <u>(Closed) LER 50-296/93-01: Unexpected Auto-Start of Unit 3 Diesel Generators</u>

In February, 1993, an unplanned actuation of an Engineered Safeguard Feature occurred when the Emergency Diesel Generators (EDG) 3A, 3B, 3C, and 3D automatically fast started, accelerated to full speed and rated voltage. The EDGs remained running and, because no undervoltage signal existed, did not tie onto their respective 4kV shutdown boards. The cause of this event was the partial severing of the cable that connects the Unit 1 accident signal logic initiating relay to the Common Accident Start (CAS) circuits in the Unit 3 4kV shutdown board system. An area of the cable was severed in such a way that the two conductors shorted and initiated a Unit 3 CAS signal. The licensee committed to replace the damaged CAS logic cable prior to the restart of Unit 1. The licensee initiated Unit 1 permanent plant design change modification DCN 51016, Emergency Core Cooling System (ECCS) Accident Signal Logic (systems 068, 074, 075, 082, and 211). The intent of this DCN was to implement the modifications recommended for the RHR initiation, the Unit 1 and 2 emergency core cooling system, and the Core Spray initiation. This DCN was also initiated to correct action items and commitments from Licensee Event Reports. The inspectors reviewed DCN 51016 which contained actions to replace the damaged CAS logic cable. Therefore, because this item is effectively being tracked in the licensee's corrective action program and because any implementation performance deficiencies would likely be detected by licensee oversight programs and have only minor consequences, this item meets the closure criteria established for Unit 1 recovery issues. Because this problem was originally identified while Unit 1 was shutdown and defueled and will be corrected prior to re-start, no new violations of NRC requirements occurred. This issue is closed for Unit 1.

E8.6 <u>(Closed) Unresolved Item (URI) 50-259/86-14-03: Over Stress of Drywell Beams</u>

This item was opened to document safety concerns regarding overstress of beams during an inspection in 1986 while all three units in Browns Ferry were shut down. The Unit 1 drywell structural steel platforms have been redesigned to correct the deficiencies. The majority of the original structural steel members for the platforms on elevation 563 and 584 have been removed and will be replaced. The connections on the platforms on elevations 604, 616, and 628 have been modified by strengthening structural steel connections by replacing bolts, addition of clip angles and stiffeners, and reinforcement of existing welds. The modified structural platforms are intended to meet current design criteria and have a design margin for addition of future loads, if necessary. The modifications have been examined by the inspectors during the current and previous inspections. This open item has been adequately resolved by the Unit 1 recovery projects and is closed. No new violations of NRC requirements were identified.

E8.7 (Closed) Violation 50-259/04-11-03 (EA-04-063): Failure to Implement Quality Assurance Requirements Over Torus Repair Activities, Which Resulted in Multiple Examples of Omitted Repairs

This violation was identified by NRC during inspections of repairs to the torus, and during review of documentation, including work plans, quality control inspection records, and welding records, associated with the torus repairs. The torus repairs were performed to correct deficiencies identified in PER 03-017339 concerning failure to complete torus integrity modifications required by Engineering Change notice (ECN) P-003 in the early 1980's. Previous review by NRC of the licensee's corrective actions to address the violation examples are documented in Section E1.3 of NRC Inspection Report number 50-2004/2004-011. The corrective actions reviewed previously were focused on defining the scope of the problems and planning corrective actions. The licensee also discussed their corrective actions for this violation during an Enforcement Conference held in the NRC Region II office on April 28, 2004, and in a letter to NRC dated June 2, 2004, Subject: Browns Ferry Nuclear Plant - NRC Inspection Report 50-259/2004-011, Reply to Notice of Violation (NOV) EA-04-063.

During the current inspection, the inspectors reviewed design calculations which evaluated welds which did not meet the original sizes specified in ECN P-003. Calculations reviewed included numbers CDQ130320032194, CDQ130320032356, CDQ130320032357, and CDQ130320032374. The inspectors reviewed revised drawings issued to complete the torus repairs required by PER 03-017339. The inspectors also examined welds which were repaired as corrective actions to resolve this violation. Specific welds inspected are listed in the back of this report.

Additional corrective actions reviewed by the inspectors included review of two desktop guides issued for use by planners in preparing work orders for DCN and Non-DCN work orders, records of training completed for planners on the desktop procedures, results of reviews completed on other Non-DCN work orders issued prior to identification of this violation, and records of training completed for quality control inspectors to correct the violation deficiencies.

The licensee's corrective actions to resolve this violation were adequate to address the causes and have been completed. Violation 50-259/04-11-03 is closed.

## III. Maintenance

### M1 Conduct of Maintenance

M1.1 Maintenance Work Control Activities (37550)

a. Observations and Findings

a.1 Failure to Notify Operations with Volatile Organic Compounds (VOCs) Greater than 20 Pounds

The Standby Gas Treatment (SBGT) System was operated on May 20, 2004, for a short period of time while greater than 20 pounds of VOCs were in use. The use of VOCs in the combined zone secondary containment are limited due to the ability of VOCs to contaminate the activated charcoal present in the SBGT System. This limit is controlled by MAI-5.3, Protective Coatings for Service Level I, II, III and Corrosive Environments and Technical Instruction (TI) 1-TI-65, Control of Unit 1 Torus Coating Activities when Release of VOCs is Expected to Exceed 20 Pounds in a 24 Hour Period. MAI-5.3, Section 6.1.12.4 requires that operations work control be notified if total VOCs are expected to exceed 20 pounds in a 24 hour period. VOCs include protective coatings, solvents and thinners used during Unit 1 torus modifications activities. 1-TI-65 provides operations with guidance for providing additional ventilation in the Unit 1 Reactor Building Zone and methods to aid in mitigation of the effects of VOCs in the secondary containment. The inspectors noted that examples had existed where tracking and control of VOCs had not met management expectations. Specifically PER 62867 documented that SBGT was operated for 3 minutes on May 20, 2004, with 23 pounds of VOCs logged in on data sheet MAI-5.3, Attachment 7, Daily Painting Tracking Log. Additionally, PER 62211 documented that during several days in May 2004 the Daily Painting Tracking Log showed total VOC release greater than 20 pounds without evidence of operations notification.

The inspectors reviewed the licensee's operability evaluation associated with the failure to implement special requirements in MAI-5.3 when VOCs exceeded 20 pounds. During this evaluation the licensee determined that there had been no significant effect on the ability of SBGT to perform its intended safety function. This was due to the low concentration of VOCs (23 pounds) and short duration of SBGT operation. The NRC review of this operability evaluation was documented in NRC Inspection Report 50-260, 296/2004-03.

a.2 Removal of Radiation Monitor from Service Without Associated Calibration Procedures to Allow Return to Service

On April 1, 2004, Raw Cooling Water (RCW) liquid effluent radiation monitor, 1-RM-90-132D, was removed from service by DCN T37520 without a detailed schedule of all activities to return the radiation monitor to service. LCO 1-090-ODCM-2004-0022 had existed since March 24, 2004, because of an unrelated problem (low RCW sample flow). Grab samples were collected and analyzed as required by Action D for Table 1.1-1, Radioactive Liquid Effluent Monitoring Instrumentation, from the licensee's Offsite Dose Calculation Manual (ODCM). DCN T37520 had been implemented on Units 2 and 3 during previous unit recoveries to replace obsolete liquid effluent radiation monitors. Modifications personnel began implementation of DCN T37520 Stage 2 on Unit 1 by installing a new drawer module for radiation monitor 1-RM-90-132D under modifications work order 96-011771-001 without the associated calibration procedures being revised to allow return to service of the new instrumentation. This resulted in an unplanned delay in return to operation while the newly installed drawer monitor was removed and the older model re-installed and verified to be functioning on April 14, 2004. The radiation monitor was then turned back over to plant instrumentation technicians to perform required surveillance testing to allow return to operability within the required LCO time limit. The licensee documented this problem in B level PER 04-003389-000. The inspectors reviewed the licensee's root cause evaluation and corrective actions. The licensee determined that the problem was due to the failure of personnel to recognize the importance of verifying all required procedures were ready before placing the work order in ready to work status.

b. Conclusions

The two examples were determined not to be a violation of NRC requirements. However, the licensee's ineffective communications between the Unit 1 maintenance/modifications organization and operations work control is considered a weakness in the area of work control/conduct of maintenance.

## M1.2 Replacement of General Electric Type HFA Relays (37550)

a. Inspection Scope

The inspectors continued to observe and review the licensee's activities involved with General Electric type HFA relays. The licensee issued a B level PER, 03-18287-000, to document the extent of condition for all three units. Relays on Units 2 and 3 were previously changed out and the NRC inspections of those activities were documented in various NRC inspection reports. The Unit 1 project has identified a total of 349 HFA relays to be changed out. The change out was due to recent concerns, as identified in the Unit 2 and 3 inspection reports, along with problems discussed in NRC Bulletins 84-02 and 88-03.

b. Observations and Findings

At the end of this report period the Unit 1 recovery project had completed change out of 132 of 349 HFA relays identified for replacement. 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 03-002001-07, 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) such as the following: 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 PERs were considered historical issues and due to the past practice of not updating or Red Lining Unit 1 drawings to reflect the as built status of electrical systems. If any problems, such as inadequate installation of a relay, were to occur during the present change out process, then a new PER would be written.

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

### R1 Radiological Protection and Chemistry Controls

R1.1 Follow-up of Unit 1 Reactor Building Contamination Event (71153)

a. Observations and Findings

On June 9, 2004, portions of the Unit 1 Reactor Building were contaminated due to an event which occurred as the result of an improperly tested High Efficiency Particulate Air (HEPA) unit. Laborers were decontaminating the highly contaminated support frame for the 1C Reactor Water Cleanup (RWCU) Regenerative Heat Exchanger (RHX) located in the RWCU heat exchanger room on the 593' elevation of the Unit 1 Reactor Building. Decon efforts were being performed inside a containment bag with a 500 CFM HEPA filter unit attached to provide negative air flow for the containment bag. The HEPA filter unit was staged outside of the contamination zone and Radiological Controls (radcon) personnel were providing continuous coverage for the decontamination evolution. The decontamination evolution was started shortly before craft personnel were expected to exit the RCA for lunch. As Unit 1 personnel started to exit the RCA a large number of personnel were found to have surface contamination on their clothing or shoes. Smear survey results performed on the floor area outside the contamination zone, around the RWCU heat exchanger room, and on the HEPA filter unit indicated excessive contamination and the decontamination evolution was stopped. The inspectors reviewed interim corrective actions associated with the event. The inspectors concluded the licensee's initial response to the event was appropriate.

The licensee's preliminary investigation revealed the HEPA unit used for the above decontamination evolution did not contain a HEPA filter cartridge. This type portable HEPA filter unit design involves an internal HEPA filter cartridge which is not visible through the prefilter. The HEPA filter unit is normally locked and remains locked except when the HEPA filter cartridge servicing is required. The licensee's preliminary investigation further determined the event was due to an inadequate test procedure and insufficient guidance in the vendor manual associated with portable HEPA filtration unit maintenance. Specifically, this HEPA filter unit had satisfactorily passed testing without a HEPA filter cartridge installed. This may have occurred due to overly restrictive 1/4 inch O.D. tubing used in conjunction with the aerosol generator utilized to perform the testing. However, additional vendor information will be needed to complete the review of the event. B level PER 62944 was issued to document the licensee's review of this event.

The licensee's final evaluation and root cause determination was still in progress at the end of the report period. Concerns associated with the adequacy of the licensee's program for testing portable HEPA filter units could not be resolved due to the limited vendor and operating experience information available at the end of the report period. An Unresolved Item (URI) 50-259/2004-07-02, Adequacy of Testing of Portable HEPA Filter Units, was identified.

b. Conclusions

An Unresolved Item was identified pending the inspectors review of vendor and operating experience information associated with testing of portable HEPA filter units.

R1.2 Follow-up of Unit 1 Drywell Contamination Event (71153)

a. Observations and Findings

On June 15, 2004, two personnel and portions of the Unit 1 Drywell were contaminated due to an event which occurred as the result from the failure of personnel to follow

Radiation Work Permit (RWP) requirements. Craft personnel were attempting to wipe down the internal surfaces of Core Spray drywell piping penetration X-16A in preparation for final cleanliness inspection. Core Spray system piping in the drywell had previously been removed and is being replaced as part of Unit 1 recovery. The planned cleanliness inspection was the last step prior to final fitup and welding of replacement core spray piping in the drywell. The craft personnel involved were signed onto RWP 04130754. This RWP provided dose control measures only, authorized general work activities in the drywell, but did not allow entry into a contamination zone, high radiation area or airborne area. The inspectors determined the individuals should have used RWP 04130751 which provided more detailed instructions for working within a contamination zone. Instructions contained in RWP 04130754 required personnel to review planned work with radcon prior to each entry and to check with radcon personnel prior to any system breach or for any unusual radiological conditions. The individuals had not reviewed planned work with radcon prior to starting the job and had not been briefed on an RWP for contaminated work. Craft supervision had failed to perform a walkdown of the jobsite prior to beginning work and the exact scope of work assignment was not well understood. Craft personnel removed the outer foreign material exclusion (FME) cover from the drywell penetration and wiped the interior of the penetration back to the contamination barrier. The interior portion of the penetration up to the barrier had been previously decontaminated to support the planned pipe replacement. The FME covers had a placard instructing workers that contamination was present inside the penetration. Workers then failed to stop work when the internal contamination barrier was reached and removed that barrier from the drywell penetration, which resulted in contamination of portions of the Unit 1 drywell. Licensee radcon personnel took prompt measures to stop the work, control the spread of contamination, and both individuals were surveyed and found to be contaminated. This event was documented by the licensee in PER 63286.

10 CFR 50, Appendix B, Criteria V, Instructions, Procedures, and Drawings, in part, states that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Licensee procedure RCI-9.1, Radiation Work Permit Preparation and Administration Section 6.6 states the RWP is the primary means by which radcon documents and controls work in radiologically hazardous areas. On June 15, 2004, contrary to the above, craftworkers failed to follow requirements as required by RWP 04130754. Based on a review of the licensee's PER investigation results and associated corrective actions and discussions with licensee personnel, the inspectors determined the failure by the craftworkers to comply with the RWP requirements was of low safety significance. Actual doses received by the individuals were low. The error resulted from inattention to detail and failure of craft supervision to perform a walkdown of the jobsite prior to beginning work. A Severity Level IV Non-Cited Violation (NCV) 50-259/2004-07-03, Failure to Follow RWP Requirements, was identified. This violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy.

The inspectors reviewed the licensee's root cause determination and held discussions with Unit 1 management. In addition, the inspectors reviewed interim corrective actions. The inspectors concluded the licensee's response to the event was appropriate.

b. Conclusions

The inspectors determined the licensee failure to follow RWP requirements was identified as a Severity Level IV Non-Cited Violation (NCV) 50-259/2004-07-03, Failure to Follow RWP Requirements

## P1 Conduct of Emergency Preparedness (EP) Activities

P1.1 Operational Status of the Emergency Preparedness (EP) Program (82701)

a. Inspection Scope

The inspectors reviewed plant-wide EP program activities, which included Unit 1, to determine whether the licensee's emergency response capability was maintained in a state of operational readiness, and to determine whether changes to the program since the last such inspection (in April 2002) met commitments, NRC requirements, and affected the licensee's overall state of preparedness.

An additional objective of this inspection was to determine the potential readiness to transition future Unit 1 inspections in EP to the Reactor Oversight Process (ROP). The ROP is the NRC's inspection program for operating reactors, and selected inspection areas (designated as Cornerstones) of the ROP can be incorporated for Unit 1 once NRC inspections conclude the area can be adequately monitored under the ROP.

b. Observations and Findings

Since April 2002, the licensee had issued Revisions 70 through 73 to the Radiological Emergency Plan (REP). The inspectors selectively reviewed and discussed with licensee representatives, the changes made in these revisions. The revisions to the REP were submitted to the NRC in accordance with regulatory requirements and were determined to have had no adverse effect on the licensee's level of emergency preparedness. No emergency declarations were made since the last inspection.

The inspectors reviewed the licensee's commitments with respect to the testing and maintenance of the alert and notification system (ANS), which comprised 100 sirens in the ten-mile-radius emergency planning zone. The testing program, delineated in Sections 8.5 and A.4.1 of the licensee's Radiological Emergency Plan (REP), included biweekly silent tests, monthly full-volume tests, and annual growl tests (the latter in conjunction with preventive maintenance). ANS changes during the past two years, post-maintenance testing methodology, and siren test records (with an emphasis on identification of any repetitive individual siren failures) were reviewed and discussed with cognizant management and maintenance personnel.

Overall ANS operability for the 12-month period ending March 31, 2004 was 99.5 percent, exceeding the FEMA acceptance criterion of 90 percent.

Emergency facilities, equipment, instrumentation, and supplies were evaluated during the September 24, 2003 biennial emergency response exercise for onsite facilities (Technical Support Center and Operational Support Center) as well as, the Central Emergency Control Center in Chattanooga, TN and found to be fully functional and well maintained.

The inspectors reviewed the emergency response organization (ERO) exercise/drill program. During the 24-month period ending March 31, 2004, the licensee conducted a sufficient number of drills to allow for participation by 97.9 percent of the 48 individuals designated as key ERO personnel. The licensee's ERO exercise/drill program was judged to be a strength.

The inspectors evaluated the efficacy of licensee programs that addressed weaknesses and deficiencies in EP. The procedure governing the plant corrective action program was reviewed for applicability to the EP program. Reports on one audit, performed in accordance with 10CFR 50.54(t), and five self-assessments were reviewed. The inspectors evaluated selected drill scenarios and associated critiques to determine whether the licensee had properly identified failures to implement regulatory requirements and planning standards. Meaningful issues were identified by these audits, self-assessments, and critiques, and resultant corrective actions were found to be thorough and timely. The inspectors also attended the licensee's Nuclear Safety Review Board Meeting No. 291 on June 10, 2004, in which emergency preparedness performance issues and corrective actions were discussed, including specific matters related to the Unit 1 Recovery Project.

EP ROP Transition Inspection

The inspectors evaluated the readiness to transition future Unit 1 inspections in EP to the ROP. This transition process is described in NRC Manual Chapter 2509. This was accomplished by inspecting the EP program for Units 2 and 3 under the ROP, and during the process evaluating Unit 1 specifics with respect to emergency preparedness and the areas of inspection defined in the EP ROP inspection procedures and the NRC EP ROP Performance Indicators. The inspectors noted that the main difference between Unit 1 and the currently operating Units 2 and 3 in the area of EP was the emergency action levels (EALs) used in determining the emergency classification of an abnormal event. Those Unit 1-specific EALs will differ only in parametric values such as radiation monitor readings and room temperatures, just as some EALs had different values currently specified for Units 2 and 3. The inspectors concluded the licensee was aware of these differences and Unit specific differences in the area of EP would be adequately addressed in EP response procedures and training for Unit 1.

The inspectors also evaluated the areas of EP typically inspected under the ROP including: alert and notification testing; drill and exercise evaluation; EAL and EP plan changes; and performance indicator reporting. In addition, the inspectors reviewed the licensee's corrective action documents to assess performance issues were being identified and corrected within the licensee's process. Based on this review and the lack of any significant findings or weaknesses in the area of EP, the inspectors did not identify any impediments to the future transition of Unit 1 EP inspections under the EP Cornerstone for the ROP. This transition will likely be reflected in the 2005 ROP Inspection Plan.

c. Conclusion

The licensee's emergency preparedness program was being maintained in a state of operational readiness. Changes to the program since the last inspection were consistent with licensee commitments and NRC requirements, and did not decrease the licensee's overall state of preparedness.

Based on focused Emergency Preparedness (EP) reviews for Unit 1, the inspectors did not identify any impediments to the future transition of Unit 1 EP inspections under the EP Cornerstone for the ROP.

## V. Management Meetings

### X1 Exit Meeting Summary

On July 26, 2004, the resident inspectors presented the inspection results to Mr. Jon Rupert and other members of his staff, who acknowledged the findings. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

ATTACHMENT: SUPPLEMENTAL INFORMATION

## SUPPLEMENTAL INFORMATION

### KEY POINTS OF CONTACT

Licensee personnel

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 | 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 |
| IP 62002 | Inspection of Structures, Passive Components, and Civil Engineering Features at Nuclear Power Plants |
| IP 70370 | Testing Piping Support and Restraint Systems |
| IP 71111.17 | Permanent Plant Modifications |
| IP 71111.23 | Temporary Plant Modifications |
| IP 73051 | Inservice Inspection, Review of Program |
| IP 73753 | Inservice Inspection |
| IP 73755 | Inservice Inspection Data Review and Evaluation |

| | |
|---|---|
| IP 92701 | Follow-up |
| IP 71153 | Event Follow-up |
| IP 82701 | Operational Status of the Emergency Preparational Program |

**LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED**

Opened and Closed

| | | |
|---|---|---|
| 50-259, 260, 296/ 04-07-03 | NCV | Failure to Follow RWP Requirements (Section R1.2) |

Opened

| | | |
|---|---|---|
| 50-259/04-07-01 | IFI | Main Steam Line Linear NDE Indications (Section E1.7) |
| 50-259/04-07-02 | URI | Adequacy of Testing of Portable HEPA Filter Units (Section R1.1) |

Closed

| | | |
|---|---|---|
| 50-259/92-29-01 | AV | Apparent Violation of Plant Records (Section E8.1) |
| 50-259/88-12 | LER | Battery Failure Concurrent With LOOP/LOCA Prevents Automatic Start of Residual Heat Removal (RHR) Pump (Section E8.2) |
| 50-259/86-18 | LER | Neutron Monitoring Surveillance Test Deficiencies (Section E8.3) |
| 50-259/86-40-03 | IFI | Intermediate Range Monitoring (IRM) Power Supply (Section E8.4) |
| 50-296/93-03 | LER | Unexpected Auto-Start of Unit 3 Diesel Generators (Section E8.5) |
| 50-259/86-14-03 | URI | Overstress of Drywell Beams (Section E8.6) |
| 50-259/04-11-01 | AV | Failure to Implement QA Requirements Over Torus Activities (Section E8.7) |

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 51016 - ECCS Accident Signal Logic, Unit 1
DCN 51018 - ECCS Accident Signal Logic, Unit 2
DCN 51216 - Reactor Building 480 VAC
DCN 51238 - Core Spray System Instrumentation

## Section E1.2 Plant Modifications

Procedures and Standards

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

DCNs

DCN 51159 - Drywell Electrical Penetrations Assemblies
DCN 51177 - RHRSW

Modifications Work Orders (WOs)

WO 03-001371-001, RHRSW large bore piping

## Section E1.3 Cable Installation and Cable Separation Special Program Activities

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
DS-E13.1.4, Electrical Design Standard, Maximum Cable Diameter for Various Rigid Steel Conduits, Rev. 1

Drawings

SD-E 15.3.3, Conduit, CA & W Indent Tags (Browns Ferry NUC PLT & All Non-NUC PLT), Rev. 4

Procedures

BFT 3-92/05277, TVA Contract TV-83425V Bechtel Job 21042 Methodology Report, dated December 18, 1992

Calculations

EDQ1 999 2003 0015, Analysis of Unit 1 Cable Installation - Miscellaneous Issues, Rev. 1
EDQ1 999 2003 0019, Analysis of Unit 1 10CFR50.49 Cables in Conduits with Missing Bushings, Rev.1
EDQ1 999 2002 0074, Analysis of Unit 1 Large 600 Volt Cables in Standard Condulets, Rev. 1
EDQ1 999 2003 0026, Analysis of Splices in 10 CFR 50.49 and Safety-Related Cables in Unit 1 Areas of Potential Flooding for Unit 1 Restart, Rev. 1
EDQ1 999 2003 0025, Evaluation of Use of Brand Rex Cable, Contract 80K6-825419, in Unit 1, Rev. 1
EDQ1 999 2003 0009, Analysis of Bend Radius for Unit 1 Safety-Related Medium-Voltage Power Cables, Rev. 2
EDQ1 999 2003 0016, Analysis of Cable Support in Vertical Raceway for Unit 1

**Section E1.4 Flexible Conduit Special Program Activities**

Calculations

EDQ1 999 2003 0014, Analysis of Flex Conduit to Devices for Unit 1, Rev. 1

Standards

DS-E13.1.7, Dimensions of Rigid and Flexible Metal Conduit Bends

Manufacturer’s Data

Electri-flex Company Publication on Type LA liquidtight flexible steel conduit

**Section E1.5 Cable Ampacity Special Program Activities**

Ampacity Calculations

EDQ1-999-2002-0039, Cable Ampacity Calculation for Safety Related V3 Power cables Outside the Drywell.
EDQ1-999-2003-0006, Mini-Calculation for EDQ1-999-2002-0039, Cable Ampacity Calculation for Safety Related V3 Power cables Outside the Drywell.
EDQ1-999-2002-0024, Cable Ampacity Calculation for safety Related V4 Power Cables Outside the Drywell
EDQ1-999-2003-0005, Mini-Calculation for EDQ1-999-2002-0024, Cable Ampacity Calculation for Safety Related V4 Power Cables Outside the Drywell
EDQ1-999-2002-0023, Cable Ampacity Calculation for Safety Related V5 Power cables Outside the Drywell
EDQ1-999-2002-0019, Cable Ampacity Calculation for Drywell Power Cables
EDQ1-999-2002-0048, Mini-Calculation for EDQ1-999-2002-0019, Cable Ampacity Calculation for Drywell Power Cables

Procedures and Standards

Electrical Design Standard DS-E12.6.3, Auxiliary and Control Power Cable Sizing up to 15,000 Volts, Rev. 10
General Design Criteria Document No. BFN-50-758, Power, Control, and Signal Cables for use in Class 1 Structures, Rev.12
Electrical; Design Standard DS-E12.1.1, Cable Conductor Current Carrying Capacity Polyethylene Insulated (0-8000V), Rev. 0

**Section E1.6, Intergranular Stress Corrosion Cracking (IGSCC) -Welding of Replacement Recirculation System Piping**

Procedures and Standards

General Engineering Specification G-29B, Material Fabrication and Handling Requirements for Austenitic Stainless Steel
TVAN Standard Department Procedure MMDP-10, Controlling Welding, Brazing, and Soldering Processes, Revision 4
MMDP-8, Controlling Welding Filler Material (WFM), Revision 0
MMDP-9, Qualification, Certification, of Personnel Performing Welding Processes, Revision 0
WASPP-201, Controlling Welding, Brazing, and Soldering (WBS) Processes, Revision 3
WASPP-202, Controlling Welding, Brazing, and Soldering (WBS) Materials, Revision 3
WASPP-203, Qualification, Certification, and Continuity of Personnel Performing Welding, Brazing and Soldering (WBS), Revision 1
SSP-4.2, Material Receipt Inspection, Revision 11
SSP-4.3, Material Storage and Handling, Revision 4
G29 Specification for Welding Materials Control - P.S.1.M.3.1, Revision 8
Detail Welding Procedure Specification (DWPS) GTA88-C-2-N, Revision 0
Nondestructive Examination Procedure N-RT-1, Radiographic Examination of Nuclear Power Plant Components, Revision 25
Inspection and Examination Procedure IEP-205, Radiographic Film Processing, Handling, and Storage of TVA Radiographs and Acceptance of Contractor Radiographs

DCNs and Work Documents

DCN 51045A, U1 Recovery Drywell mechanical Lead System C68
DCA 51045-124
Work Orders (WOs) 02-010314-02, 02-010314-03, 02-010314-04, and 02-010314-05, Replace RECIRC Piping and Components with Stainless Steel Type 316NG Material

Problem Evaluation Reports (PERs)

04-001337-000 - Welder for RECIRC Pipe Welding did not possess Filler Material Requisition Form
04-001804-000 - Welding Material Control Discrepancies in Weld Material Issue Centers
62488 - Unacceptable RT Indications in RECIRC System Weld RWR-1-002-053
62423 - Unacceptable RT Indications in RECIRC System Weld RWR-1-001-003
62290 - Unacceptable RT Indications in RECIRC System Weld RWR-1-002-004
62095 - RECIRC Pipe Spool 68-19 Bumped and Dented While Being Moved in Drywell
61067 - Welder Issued Two 90 Series Welding Rods with 316L Rods for RBCCW Welding

61135 - Welder Issued Five 90 Series Welding Rods with 316L Rods for RBCCW Welding
61655 - For Cross-Under Pipe Welding, Unattended E9018B3 Rods Found on Foreman's Desk

**Section E1.7: Inservice/Preservice Inspection (ISI/PSI)**

Procedures and Standards

1-SI-4.6.G, Inservice Inspection Program Unit 1, Revision 5
1-MMI-46 PT, Liquid Penetrant Examination of Piping and Components Which were Exposed to Residue From Plant Fire Unit 1
N-UT-76, Generic Procedure for Ultrasonic Examination of Ferritic Pipe Welds, Rev 4
N-MT-6, Magnetic Particle Examination for ASME and ANSI Code Components and Welds, Rev 25
N-VT-1, Visual Examination for ASME and ANSI Section XI Pre-Service and Inservice, Rev 35
N-PT-9, Liquid Penetrant Examination of ASME and ANSI Code Components and Welds, Rev 25

Problem Evaluation Reports (PERs) / Notice of Indications (NOIs)
PER 04-003621-000, DSMS-1-43 and DSMS-1-44 Linear indications outside Section XI examination boundary
PER 04-003367-000, DSMS-1-30 Gouge in pipe outside Section XI examination boundary
NOI U1C6R-012, DSMS-1-13 Unacceptable linear indication within Section XI examination boundary
NOI U1C6R-013, DSMS-1-30 Unacceptable linear indication within Section XI examination boundary

Examination Reports
NDE Report # R-068, PT FW-1-SSA5-1A
NDE Report # R-226, VT CRD 1-47B468-261
NDE Report # R-272, UT CS-1-29
NDE Report # R-271, UT CS-1-33
NDE Report # R-270, MT CS-1-29
NDE Report # R-269, MT CS-1-33
NDE Report # R-0151, MT and UT DSMS-1-13
NDE Report # R-0151, MT and UT DSMS-1-30
NDE Report # BOP-U1-R306, MT DSMS-1-43
NDE Report # BOP-U1-R305, MT DSMS-1-44
NDE Report # BOP-U1-R304, MT and UT DSMS-1-30

**Section E1.8 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 0-04-004-023

**Section E1.9 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)

PER 41617, System 79, Fuel Handling, design change DCN-51367 was listed as open against Unit 1 when this DCN only affected Units 2 and 3 and is closed
PER 47372, System 39, CO2 Storage and Fire Protection / Generator Purge, ECN-51724 was not shown on the Open Item Punchlist
PER 47538, failure to recognize that there are programmatic items involving certain systems that need to be addressed in order to support Unit 1 in an operating condition

**Section E1.10 Power Service Shop (PSS) Activities**

Work Documents

WO 02-010314-078, Machining Recirculation System Loop A five port fitting

**Section E1.11 Drywell Steel Platforms Special Program Activities**

Specifications & Procedures

TVA General Engineering Specification G-89, Requirements for Structural and Miscellaneous Steel, Rev. 3, dated 4/26/94.
TVA General Engineering Specification G-29-S01, PS 4.M.4.4, ASME Section III and Non-ASME (Including AISC, ANSI B31.1 and ANSI B31.5) Bolting Material, Rev 5, dated 11/12/02.
MAI-1.3, General Requirements for Modifications, Rev. 16, dated 9/15/03.
MAI-5.2, Bolting and Structural Connections, Rev. 14, dated 6/23/03.
MAI-5.9, Fabrication and Installation of Structural and Miscellaneous Steel, Rev. 11, dated 6/16/03.
TVA Nuclear Engineering Design Procedure NEDP-10, Design output, Rev. 8, dated 2/25/03
Surveillance Instruction 0-SI-4.7.A.2.K, Primary Containment Drywell Surface Visual Inspection, Rev. 12
Technical Instruction 0-TI-376, ASME Section XI Containment Inservice Inspection Program, Rev. 4
Technical Instruction 0-TI-417, Inspection of Service Level I, II, III Protective Coatings, Rev. 3
Technical Instruction 0-TI-466, Plant Modification Related Work Order Preparation and Processing, Rev. 4
Procedure N-VT-15, Visual Examination of Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Plants, Rev. 5
Procedure MMDP-10, Controlling Welding, Brazing, and Soldering Processes, Rev. 4

Desktop Guide for Large Scope/Non-DCN Related Work Orders, Rev. 1, dated 4/23/04
Desktop Guide for DCN Related Work Orders, Rev. 3, dated 4/19/04

Drawings

Drawing number 1-48E442-1, Structural Steel Drywell Floor Framing, Elev. 563' - 0-1/2", Rev. 7
Drawing number 1-48E442-2, Structural Steel Drywell Floor Framing at Elev. 563' - 0-1/2", Azimuth 351° to 81°, Rev. 6
Drawing number 1-48E442-6, Structural Steel Floor Framing at Elev. 563' - 0-1/2", Sections & Details, Sheet 1, Rev. 5
Drawing number 1-48E442-8, Structural Steel Floor Framing at Elev. 563' - 0-1/2", Sections & Details, Sheet 2, Rev. 7
Drawing number 1-48E443-1, Structural Steel Drywell Floor Framing, Elev. 584' - 91/2", Rev. 7
Drawing number 1-48E443-5, Structural Steel Drywell Floor Framing at Elev. 584' - 91/2", Azimuth 262° to 351°, Rev. 6
Drawing number 1-48E443-6, Structural Steel Floor Framing at Elev. 584' - 91/2", Sections & Details, Sheet 1, Rev. 6
Drawing number 1-48E443-7, Structural Steel Floor Framing at Elev. 584' - 91/2", Sections & Details, Sheet 2, Rev. 7
Drawing number 1-48E981-1, Miscellaneous Steel Access Platforms, Elev. 604' - 0", Plan, Rev. 1
Drawing number 1-48E981-2, Miscellaneous Steel Access Platforms, Elev. 604' - 0", Sections and Details - Sheet 2, Rev. 2
Drawing number 1-48E981-3, Miscellaneous Steel Access Platforms, Elev. 604' - 0", Sections and Details - Sheet 2, Rev. 1
Drawing number 1-48E981-4, Miscellaneous Steel Access Platforms, Elev. 604' - 0", Sections and Details - Sheet 3, Rev. 2
Drawing number 1-48E982-1, Miscellaneous Steel Access Platforms, Elev. 616' - 0", Plan, Rev. 3
Drawing number 1-48E982-2, Miscellaneous Steel Access Platforms, Elev. 616' - 0", Sections and Details - Sheet 2, Rev. 2
Drawing number 1-48E982-3, Miscellaneous Steel Access Platforms, Elev. 616' - 0", Sections and Details - Sheet 2, Rev. 3
Drawing number 1-48E982-4, Miscellaneous Steel Access Platforms, Elev. 616' - 0", Sections and Details - Sheet 3, Rev. 2
Drawing number 1-48E983-1, Miscellaneous Steel Access Platforms, Elev. 628' - 0", Plan, Rev. 3
Drawing number 1-48E983-2, Miscellaneous Steel Access Platforms, Elev. 628' - 0", Sections and Details - Sheet 1, Rev. 2

Design Calculations

CDQ130320020857, Drywell Floor Framing Modifications at Elevation 604, Rev. 1, dated 4/28/04
CDQ130320020858, Drywell Floor Framing Modifications at Elevation 616, Rev. 1, dated 4/28/04
CDQ130320020859, Drywell Floor Framing Modifications at Elevation 628, Rev. 1, dated 4/28/04
CDQ130320020860, Connections for Upper Drywell Floor Framing Modifications at Elevations 604, 616, and 628, Rev. 1, dated 6/14/04

CDQ130320032194, Torus Integrity - Misc Steel Torus Spray Header Supports - Review of Walkdown Packages, Rev. 3, dated 5/11/04

CDQ130320032356, Torus Integrity - Mechanical Suppression Chamber Internal Modifications - Review of Walkdown Packages, Rev. 4, dated 5/11/04
CDQ130320032357, Torus Integrity - Misc Steel Torus Walkways and Access Platforms - Review of Walkdown Packages, Rev. 5, dated 5/11/04
CDQ130320032374, Torus Integrity - Evaluation of Box Beams and Rams Head Supports Inside Torus - Review of Walkdown Packages, Rev. 4, dated 5/11/04

Miscellaneous Documents

General Design Criteria Document BFN-50-C-7100, Design of Civil Structures, Rev. 14, dated 3/17/04
DCN 51286, Provide Modifications to Drywell Structural Steel platforms at Elev. 604, 616, and 628
Work Order (WO) 02-008179-007, Re-install Drywell Steel Azimuth 30 to 60 Degrees on Elevation 584.0 Per the Marked Up Drawing
WO 02-008179-017, Install Drywell Steel Azimuth. 262 to 278 degrees, Elevation 584, Unit 1
WO 02-008180-012, Install Drywell Steel Azimuth. 122 to 148 degrees, Elevation 563, Unit 1
WO 03-017394-018, Repair Welds as Described in PER 03-017339-000 for Bay 1 thru Bay 5 and Ring Girder 1 Thru Ring Girder 5
WO 03-017394-019, Repair Welds as Described in PER 03-017339-000 for Bay 6 thru Bay 9
WO 03-017394-020, Repair Welds as Described in PER 03-017339-000 for Bay 10 thru Bay 13 and Ring Girder 10 Thru Ring Girder 13
WO 03-017394-021, Repair Welds as Described in PER 03-017339-000 for Bay 14, Bay 15 and Bay 16
Walkdown Packages WDP-BFN-1-CEB-01-09 through -14, As-Built Drywell Platform Steel for the Elevation 604, 616, and 628 platforms
DWPS SM11-B-1-N, G29 Detailed Welding Procedure Specifications (DWPSs) ASME/ANSI - GWPS 1.M.1.2
PQR GT-SM11-0-3C
PER 03-017339-000, Construction Discrepancies Identified in Unit 1 Torus

**Section E1.12 Containment Coatings Special Program Activities/Repairs to Coatings in Unit 1 Torus**

Specifications & Procedures

TVA General Engineering Specification G-29-, Part B, Standard Materials Specifications Section 2A: Consumerables and Other Miscellaneous Materials, dated 5/26/98, Purchase Specification for Solvents for Use on Stainless Steel.
TVA General Engineering Specification G-55, Technical and Programmatic Requirements for the Protective Coating Program for TVA Nuclear Plants, Rev. 12, 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.

Problem Evaluation Reports (PERs)

PER 04-062773-000, Qualification of Apprentice Painters to Apply Service Level I Protective Coatings
PER 04-060438, 04-060588, 04-060673, 04-061052, 04-062762, & 04-062764, Surface Preparation Deficiencies
PER 04-062385 & 04-63931, Inadequate Coatings application
PER 04-046805, Inconsistent Criteria for Inspection of Existing Coatings in Torus Vapor Space
PER 04-061905, Questions on Inspection Techniques Used for Existing Torus Coatings
Level B PER 04-064037-000, Administrative Deficiencies in Qualification Records of Protective Coatings Applicators, and referenced Level C PERs 04-064038, 04-064039, 04-064041, 04-064043, 04-064044, 04-064045, 04-064046, 04-0064047, 04-064048, 04-064050, 04-064053, 04-064054, 04-064055, 04-064056, 04-064057, 04-064058, 04-064059, 04-064072, 04-064073, 04-064075, 04-064076, 04-064077, 04-064078, 04-064079, 04-064080, 04-064081, 04-064083, 04-064084, 04-064085, 04-064086, 04-064087, 04-064088, and 04-064089

**Section E1.13 System Cleanliness Verification Activities**

Procedures, Guidance Documents, and Manuals

Technical Instruction 1-TI-474, Cleanliness Verification Program, Rev. 0

Problem Evaluation Reports (PERs)

PER 64968, 1A1 main steam crossaround piping failed cleanliness inspection due to presence of debris
PER 64979, 1C1 main steam crossaround piping failed cleanliness inspection due to presence of slag
PER 64980, 1C2 main steam crossaround piping failed cleanliness inspection due to presence of debris
PER 64982, 1B1 main steam crossaround piping failed cleanliness inspection due to presence of debris

**Section E7.1 Licensee Quality Assurance Oversight of Recovery Activities**

TVA Power Service Shop Documents

Corrective Action Report (CAR) 2004-1411, Piping spool identification number discrepancies
CAR 2004-1432, Lack of liquid penetrate cleaner documentation
CAR 2004-1446, Replacement recirculation pipe surface condition
CAR 2004-1455, Inconsistent procedure revision numbers

Nuclear Assurance Observation Reports

31993, Machining of Recirculation piping at Power Service Shop on 6/24-28/04
32008, Inspection of bulk storage Level C warehouse located on Muscle Shoals Reservation
TVA Corrective Action Documents

PER 64271, concrete floor in Muscle Shoals warehouse had an excess buildup of dirt and required cleaning. Did not satisfy Level C storage

**Section E8.7 Follow-up of Unit 1 Corrective Actions for Torus Weld Repairs**

Weld # PCI-1-002-017, -018, -019, -026, -027, 028, &-029, Stiffener plates on main vent line per Sketch # 30, R4 Azimuth 112° - 30‘
Weld # PCI-1-WO 0317394020-001 & -002, Ring Girder # 10, Sketch 8-2, R0, Plates 1 & 2
Weld # MS-1-WO 0317394021-001 & -002, rigid restraints, support # RS13, Sketch 28, R4, main steam vent valve piping
Weld # PCI-1-WO03017394-018-003, -004, box beam, azimuth 33° - 45 ‘, ring girder 2 Sketch # 36-1, R0
Weld # PCI-1-WO03017394-018-009, box beam, azimuth 101° - 15 ‘, ring girder 5, Sketch # 36-3, R0
Weld # PCI-1-WO03017394-018-005, box beam, azimuth 78° - 45 ‘, ring girder 4, Sketch # 36-2, R0
Weld # PCI-1-WO03017394-019-001, box beam, azimuth 146° - 15 ‘, ring girder 7, Sketch # 35-2, R0
Weld # PCI-1-002-034, reinforcing pad on downcomer, Bay 5, Sketch # 44 (WO # -018)
Weld # PCI-1-002-024, reinforcing pad on downcomer, Bay 3, Sketch # 44 (WO # -018)
Weld # PCI-1-WO03017394-020-003, fishmouth on downcomer, Bay 12, Sketch # 44-Bay 12, R1, (WO # -020)
Weld # PCI-1-WO03017394-020-030, -031, -032, & -033, fishmouth on downcomer, Bay 11, Sketch # 44 - Bay 11, R1 (WO # -020)
Weld # PCI-1-WO03017394-018-034, fishmouth on downcomer, Bay 5, Sketch # 44-Bay 5, RR@. (WO # -020)
Weld # PCI-1-WO03017394-018-002, reinforcing plate on downcomer, Bay 1, Sketch # 44-Bay 1, R2 (WO # -020)

**Section M1.1 Conduct of Maintenance**

Procedures and Standards

Offsite Dose Calculation Manual (ODCM), Rev. 16
MAI-5.3, Protective Coatings for Service Level I, II, III and Corrosive Environments
Technical Instruction (TI) 1-TI-65, Control of Unit 1 Torus Coating Activities when Release of VOCs is Expected to Exceed 20 Pounds in a 24 Hour Period.

Problem Evaluation Reports (PERs)

04-003389-000, Raw cooling monitor, 1-RM-090-0132D removed from service for replacement by DCN T37520 without a detailed schedule of all activities to return to service

**Section R1.1 Follow-up of Unit 1 Reactor Building Contamination Event**

Procedures and Standards

Radiological Control Instruction, RCI-24, Control of Vacuum Cleaners and Portable HEPA Units within the RCA, Rev 20
Radiation Protection Instrument Program Implementing Procedure INST-IP-17, DOP Testing of HEPA Filters in Vacuum Cleaners and Portable HEPA Filters, Rev 58

Problem Evaluation Reports (PERs)

PER 62944, Personnel contamination event Unit 1 RWCU heat exchanger support decon, 6/904

**Section R1.2 Follow-up of Unit 1 Drywell Contamination Event**

Procedures and Standards

RCI-9.1, Radiation Work Permit Preparation and Administration

Radiation Work Permits (RWPs)

RWP 04130751, Unit 1 drywell maintenance/modifications on Core Spray System (various dress)
RWP 04130754, Unit 1 drywell maintenance/modifications on Core Spray System (dose control)

Problem Evaluation Reports (PERs)

PER 63286, Drywell core spray contamination event, 6/15/04