UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
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ATLANTA, GEORGIA 30303-8931

August 15, 2003

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

SUBJECT: BROWNS FERRY NUCLEAR PLANT UNIT 1 RECOVERY - NRC INTEGRATED INSPECTION REPORT 05000259/2003009

Dear Mr. Scalice:

On July 19, 2003, the US 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 29, 2003, with Mr. John Rupert and other members of your staff.

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

In accordance with 10 CFR 2.790 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 05000259/2003009
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No:	50-259
License No:	DPR-33
Report No:	05000259/2003009
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Browns Ferry Nuclear Plant, Unit 1
Location:	Corner of Shaw and Nuclear Plant Roads Athens, AL 35611
Dates:	April 6, 2003 - July 19, 2003
Inspectors:	W. Bearden, Senior Resident Inspector, Unit 1 R. Chou, Reactor Inspector, (Sections E1.4, E1.5) B. Crowley, Senior Reactor Inspector, (Section E1.6) J. Lenahan, Senior Reactor Inspector, (Section E1.7)
Accompanying Personnel:	S. Belcher, Nuclear Safety Intern K. Harper, Summer Safety Intern
Approved by:	Stephen J. Cahill, Chief Reactor Project Branch 6 Division of Reactor Projects

EXECUTIVE SUMMARY

Browns Ferry Nuclear Plant, Units 1
NRC Inspection Report 05000259/2003-009

This integrated inspection included aspects of licensee engineering and modification activities associated with the Unit 1 restart project. The inspection program for the Unit 1 Restart Program is described in NRC Inspection Manual Chapter 2509. The report covers a 3 month period of resident inspection. In addition, the NRC staff conducted inspections of Unit 1 Special Programs in the areas of drywell steel platforms, large bore pipe supports, torus integrity program, and intergranular stress corrosion cracking (IGSCC).

<u>Engineering</u>

- During reviews of system piping to determine the effects of long term lay-up, no flaw like indications or wall thickness values below acceptable were identified. Several examples of flow accelerated corrosion (FAC) and localized corrosion were found during ultrasonic examinations. The licensee determined that the condition of these Unit 1 components and piping was acceptable and comparable to the condition previously found on Unit 2 and Unit 3 (Section E1.1).

- During reviews of heat exchangers to determine the effects of long term lay-up, the licensee determined that condition of those heat exchangers was acceptable and comparable to the condition previously found on Unit 2 and Unit 3. Several minor examples of tube degradation were found during the eddy current examinations and repair activities were planned prior to restart of Unit 1. (Section E1.2).

- No violations or deviations were identified during the review of two Unit 1 modifications (Section E1.3).

- No violations or deviations were identified during the preliminary review of the licensee's Long Term Torus Integrity Special Program for Unit 1 (Section E1.4).

- An Unresolved Item (URI) was identified for an apparent inconsistency in measuring weld sizes to establish piping and support as-built drawings (Section E1.5).

- No violations or deviations were identified during the preliminary review of the licensee's Intergranular Stress Corrosion Cracking Special Program for Unit 1 (Section E1.6).

- No violations or deviations were identified during the preliminary review of the licensee's Drywell Steel Platforms Special Program for Unit 1 (Section E1.7).

- Initial reviews of licensee procedures relative to QA oversight of Unit 1 modifications/recovery activities and sampled QA assessments and observation reports indicated that adequate oversight was being applied. No violations or deviations were identified (Section E7.1).

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. Demolition and removal of selected portions of plant systems is continuing. Engineering and procurement activities to support replacement of plant components is ongoing. Some reinstallation of plant equipment and structures has occurred. Ongoing recovery activities include design walkdowns; replacement of drywell structural steel; removal and replacement of reactor coolant system piping safe ends; torus draindown and decontamination; and installation of new electrical penetrations, cable trays, and cable tray supports.

III. Engineering

E1 Conduct of Engineering

E1.1 Piping UT Examination Activities (Inspection Procedures 49001, 57080, 71111.08G, 73051)

a. Inspection Scope

The inspectors reviewed selected records for Ultrasonic Examination (UT) activities performed on the Unit 1 Drywell liner, Core Spray injection valves, Torus ECCS Ring Header along with piping from the Reactor Core Isolation Cooling (RCIC), High Pressure Injection Cooling (HPIC), and Residual Heat Removal (RHRSW) systems.

b. Findings and Observations

The licensee had performed extensive UT exams to determine the effects of long term lay-up, to verify that piping wall thickness measurements satisfied design requirements, and to verify their assumptions about the condition of the piping for those systems to remain in use. Based on the results of the licensee’s reviews, these components and piping systems remained in good condition and did not require replacement. With the exception of the stainless steel CS injection valves, the piping and components in these systems were manufactured from low alloy steel and not susceptible to Stress Corrosion Cracking. The reviewed records documenting manual UT examination of various components and piping segments are listed in the attachment.

The results of the UT examinations were compared to the applicable drawings and system design requirements to verify acceptable piping wall thickness readings. Qualification and certification records for examiners, equipment and procedures for the UT examination activities were also reviewed.

c. Conclusions

During reviews of system piping to determine the effects of long term lay-up, no flaw like indications or wall thickness values below acceptable were identified. Several examples of flow accelerated corrosion (FAC) and localized corrosion were found during ultrasonic examinations. The licensee determined that the condition of these Unit 1 components and piping was acceptable and comparable to the condition previously found on Unit 2 and Unit 3.

No violations or deviations were identified.

E1.2 Eddy Current Examination of Heat Exchangers (49001, 73051)

a. Inspection Scope

The inspectors reviewed the licensee’s action plan for Unit 1 safety-related heat exchangers, completed tubing degradation evaluations and selected records for Eddy Current Examination (ET) activities performed on service water tubing from heat exchangers from the Residual Heat Removal (RHR) and Reactor Building Closed Cooling Water (RBCCW) systems. The licensee’s ET examinations and evaluations were performed to determine the effects of long term lay-up and to identify any degraded tubes that might be present in these heat exchangers. Based on the reviews conducted, these heat exchangers remained in good condition and did not require replacement. The evaluations and records documenting ET examination of these heat exchangers are listed in the attachment.

The results of the ET examinations were compared to the applicable drawings and system design requirements to verify acceptable piping wall thickness readings. Qualification and certification records for examiners, equipment and procedures for the ET examination activities were reviewed.

b. Findings and Observations

Although some examples of tube degradation were found during the ET examinations repair activities are planned prior to restart of Unit 1. The licensee concluded that the condition of heat exchangers was acceptable and comparable to the condition previously found on Unit 2 and Unit 3.

c. Conclusions

During reviews of heat exchangers to determine the effects of long term lay-up, the licensee determined that condition of those heat exchangers was acceptable and comparable to the condition previously found on Unit 2 and Unit 3. Several minor examples of tube degradation were found during the eddy current examinations and repair activities were planned prior to restart of Unit 1.

No violations or deviations were identified.

E1.3 Permanent Plant Modifications (71111.17A)

a. Inspection Scope

The inspectors reviewed the following permanent plant modifications to install new electrical penetrations, cable trays and supports in the control bay and the new equipment hatch sub-doors. 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 the risk significant plant modifications, were developed, reviewed, and approved per the procedure requirements. The inspectors observed field work and reviewed the post-modification work and test package following the DCN implementation to verify the design basis, licensing bases, and TS required performance for the system had not been degraded as a result of the modification.

b. Observations and Findings

b.1 Design Change Notice (DCN) 51088 - Control Bay Cable Tray Installation

The inspectors reviewed the permanent plant modification to install new electrical penetrations, cable trays and raceway components in the Cable Spreading Room, Auxiliary Instrument Room, and Control Room for Unit 1. The intent of this DCN was to minimize the use of conduit in these areas and to provide for future planned cable installation during Unit 1 recovery. The inspectors reviewed criteria in licensee procedures and modification instructions to verify the risk significant plant modification was developed, reviewed, and approved per the procedure requirements. The inspectors observed field work to perform core drilling, installation of cable trays and supports, and installation of concrete anchors. Additionally, the inspectors reviewed the licensee’s program for determining allowable size for breaches in the control room habitability zone and secondary containment during the ongoing work activities.

b.2 DCN 51268 - Equipment Access Lock Sub-door

The inspectors reviewed the permanent plant modification to retrofit the Reactor Building Equipment Access Lock (EAL) doors with sub-doors. This modification was performed to improve access and to support the increased need for ingress and egress of equipment and construction materials during the Unit 1 recovery effort. Routine opening and closing of the larger EAL doors while secondary containment is required increased potential for seal failure. The smaller sub-doors were intended to allow for routine opening without frequent seal failure. The inspectors observed field work associated with installation of the sub-doors and reviewed the post modification testing results. Additionally, the inspectors reviewed the licensees efforts to ensure that secondary containment integrity was maintained throughout the installation of the sub-doors.

c. Conclusions

No violations or deviations were identified.

E1.4 Long Term Torus Integrity Program (62002)

a. Inspection Scope

The inspectors reviewed the Browns Ferry Regulatory Framework document dated December 13, 2002 and previous commitments stated in a letter sent to the NRC dated April 29, 1991 for the restart of Unit 1. TVA Nuclear Performance Plan Section 3.1.2 for Browns Ferry Plant, Rev. 2, dated October 1986 stated that modification and configuration problems were found for torus internal structural components and attached pipe supports such as undersized welds, excessive restraint gaps, and installation/fabrication configurations differing from design drawing requirements. The inspectors reviewed documents and discussed the program status with licensee engineers. The inspectors discussed the Unit 1 Restart Nuclear Assurance (NA) program review of the Long Term Torus Integrity Program with the Unit 1 NA Manager. NUREG 1232 Vol. 3 and Supplements 1 and 2, Safety Evaluation Report (SER) on Tennessee Valley Authority: Browns Ferry Nuclear Performance Plan Browns Ferry Unit 2 Restart were reviewed for acceptance criteria.

b. Observations and Findings

The Browns Ferry Regulatory Framework document, Table 3 stated that the Long Term Torus Integrity Program will include inspections of safety-related torus and torus related structures, catwalk inspections, which would be limited to welds and bolted connections, and resolution of torus attached piping discrepancies. The torus attached piping will be inspected and evaluated within the scope of the large bore piping and supports program.

The licensee has completed the structure walkdown inspections and has issued Work Orders for the outside area of the torus to fix missing stiffener plates and undersized and missing welds in the torus external ring girders. The licensee was in the process of draining the torus water during the inspection to prepare for the walkdown inspections inside of the torus.

a. Conclusions

Initial reviews of the Long Term Torus Integrity Program identified no violations or deviations.

E1.5 Large Bore Piping and Supports (62002)

a. Inspection Scope

The inspectors reviewed the Browns Ferry Regulatory Framework document and previous commitments for the restart of Unit 1. The inspectors reviewed the large bore piping and supports program and discussed the status of the program with licensee engineers. TVA Nuclear Performance Plan Sections 3.2 for Browns Ferry Plant, Rev. 2, dated October 1986, stated that TVA did not complete commitments made from NRC Bulletins 79-02, Pipe Support Base Plate Designs Using Concrete Expansion Anchors and 79-14, Seismic Analysis for As-Built Safety-Related Piping and would perform inspections, evaluations, and modifications for safety-related large bore piping systems including torus attached piping to meet requirements. The inspectors walked down and inspected two pipe supports and a short length of piping with the licensee's engineer and walkdown personnel to verify the licensee walkdown effectiveness. The inspectors discussed NA activities with NA manager. NUREG 1232 Vol. 3 and Supplements 1 and 2 were used for acceptance criteria. The inspectors also reviewed the Nuclear Assurance (NA) Verification and Oversight Plan, Assessment Reports, and Daily Observation Reports associated with the area of large bore piping and supports.

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

TVA Browns Ferry Regulatory Framework document, Table 3 references TVA's program for seismic qualification of large bore piping and supports, which was completed for Unit 3 using the Unit 2 precedent. Essentially, the Large Bore Piping and Supports Program for Units 2 and 3 was to complete the commitments and requirements for NRC Bulletins 79-02 and 79-14. The torus attached piping will be inspected and evaluated within the scope of the large bore piping and supports program.

The licensee completed the walkdown inspection, evaluation, and issuance of DCNs for the large bore piping inside the drywell. The licensee completed most of the walkdown, evaluation, and issuance of DCNs for the Reactor Building. The licensee produced new drawings of piping isometrics and supports based on the data gathered during the walkdown inspections.

The inspectors reviewed two supports and a portion of the piping with the licensee engineers and walkdown personnel and compared results to the new drawings contained in walkdown packages. The inspectors also observed licensee personnel obtain the data. Inspection elements included measurements for dimensions, member and component sizes, weld sizes and symbols, and pipe diameter. Licensee personnel reported numerous weld sizes which were below the values documented on the new walkdown drawings. Pipe support WDP-BFN-1-CEB-003-01-02-SK-1 (Formerly Support FW-H4), Rev. 1 was found to have 1/16 inch smaller welds in 15 locations and 1/8 inch smaller welds in two locations. Pipe support WDP-BFN-1-CEB-003-01-02-SK-3 (formerly Support FW-H2), Rev. 0 was found to have 1/16 inch smaller welds in five locations. The inspectors raised a concern with an apparent inconsistency in measuring weld sizes to establish the piping and support as-built drawings which would be used for evaluation to determine if modifications to the piping and supports were necessary. The licensee issued Problem Evaluation Report (PER) No. 03-010789-000 for the inconsistently measured welds found on these two pipe supports for resolution and initiated a reinspection for these two supports. This item is identified as Unresolved Item (URI) 50-259/2003-009-01, Inconsistency for Measurement of Welds in Pipe Supports WDP-BFN-1-CEB-003-01-02-SK-1 and 3, pending review of the identified deficiencies and extent of condition reviews.

c. <u>Conclusions</u>

An Unresolved Item was identified regarding an inconsistency for measurement of welds in pipe supports when compared to the new drawings based on the walkdown inspection.

E1.6 Intergranular Stress Corrosion Cracking (IGSCC) - Welding of Replacement of Reactor Vessel Safe Ends (55050)

a. Inspection Scope

As discussed in Section 3.6 of NUREG 1232 and in Section 7.0 of the Browns Ferry Nuclear Performance Improvement Plan, intergranular stress corrosion cracking (IGSCC) was identified in a number of stainless steel piping systems and reactor vessel (RV) safe ends during nondestructive examination (NDE) of these systems in response to NRC Generic Letter 88-01. As part of the IGSCC Special Program, TVA is replacing the RV N1 (RECIRC Outlet), N2 (RECIRC Inlet), N5 (Core Spray), and N8 (Jet Pump Instrumentation) nozzle safe ends. As detailed in TVA Browns Ferry Unit 1 Regulatory Framework Letters December 13, 2002 and February 28, 2003, and Letter of Response to Request for Supplemental Information on the Regulatory Framework for the Restart of Unit 1, dated June 11, 2003, the applicable Codes for the safe end replacements are: (1) ASME Section XI, 1995 Edition, 1996 Addenda, and (2) ASME Section III, Class 1, 1995 Edition, 1996 Addenda.

The inspectors reviewed the preparations, including procedures and personnel qualification records, for removal and welding of the new RV nozzle safe ends.

b. Observations and Findings

The licensee contracted Welding Services Inc. (WSI) for the nozzle safe end replacement project. Detailed work procedures were in place for each type of nozzle to perform the work in accordance with the WSI Quality Assurance (QA) Manual. In addition to review of these detailed procedures, the inspectors reviewed the following welding procedures and qualification records to verify compliance with the applicable Codes listed above:

Welding Procedure Specification (WPS) 08-08-TS-001, Revision 2, including applicable Procedure Qualification Records (PQRs)

Welder Qualification Records, including records of qualification maintenance, for welder Stamp Nos. RLA1203, MBH0896, TEK5505, DHL0445, DGP4964, and WCS1283

Certified Material Test Reports for the following heats/lots of welding material: .035" ER 316/316L Spooled Wire - Lot XF8056, 1/8" ER 316/316L Cut Length Wire - Heat 316041, 1/8"&5/32" ER 316/316L Consumable Inserts - Lot T6670, and 1/8"&5/32" ER 316/316L Consumable Inserts - Lot T6396

WSI Certificate of Qualification Records for 3 Welding Inspectors

Procedures and records reviewed were found to comply with the applicable Codes.

c. Conclusions

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

E1.7 Drywell Steel Platforms (62002)

a. Inspection Scope

The inspectors reviewed ongoing activities associated with removal and reinstallation of structural steel in the Unit 1 drywell to verify that the modified structural steel components will satisfy the current design criteria.

b. Observations and Findings

Background

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 fabricated from bent plates 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 (ie, failure to control addition of more loads to platforms). The Unit 1 drywell structural steel platforms have been redesigned to correct the deficiencies. The majority of the original structural steel members have been removed and will be replaced. 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 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, Subject: Design Criteria for Lower Drywell Steel Platforms and Miscellaneous Steel.

Structural Steel Verification Program

In order to perform the design verification for the as-built drywell structural steel platforms in Unit 1, walkdown inspections were performed by engineering personnel to document the condition, configuration, and existing loadings on the drywell platforms.

The inspectors reviewed TVA procedure WI-BFN-0-CEB-02, Walkdown Instructions for Seismic Issues (Civil), which specified the requirements for performance of the walkdown. The inspectors, accompanied by walkdown personnel, independently verified selected attributes on various structural steel members between azimuths 279° and 51° locations. The attributes included weld size and length, dimensions of members, connection details, and identification of type and condition of structural steel bolts. The inspectors reviewed walkdown package numbers WDP-BFN-1-CEB-303-01-01, -03, and -04 which document results of the walkdowns completed on the elevation 563 platforms from azimuth 351° to 81°, 171° to 261°, and 261° to 351°, respectively. The inspectors verified the existing as-built details identified during the walkdowns were evaluated in the revised and updated design calculations.

Procurement of New Steel for Modification of Drywell Structural Steel Platforms

The inspectors examined the licensee's program for procurement of new structural steel members for modifying the drywell structural steel. The inspectors reviewed procurement documents and discussed the licensee's vendor inspection and surveillance program with licensee engineers and nuclear assurance personnel. The inspectors reviewed the licensee's program for receipt and inspection of new structural steel members when they are received onsite, and the controls to maintain traceability of the materials to required quality standards. The inspectors reviewed quality records, including certified materials test reports, and reviewed the receipt inspection documents which demonstrate that fabricated steel members comply with design requirements.

Inspection of Drywell Steel Modification Activities

The inspectors reviewed work orders issued to implement modifications to the drywell structural steel and verified that information from field walkdowns and design drawings were correctly translated into work instructions. Documents examined included work control instructions, including quality control (QC) holdpoints, weld maps, instructions and location sketches to control installation of new structural steel bolts, and weld travelers. The inspectors also examined selected modifications to the Unit 1 elevation 563 drywell structural steel frames and platforms between azimuth 351°and to verify that modifications were completed in accordance with Quality Assurance (QA) documentation, which included the work order packages, design drawings and documents contained in the Design Change Notices (DCN). During the walkdown inspection, the inspectors verified the following attributes complied with the requirements shown on the design drawings: member sizes, configuration, weld sizes, type, and length, connection details, and other requirements such as use of correct bolts and removal of existing steel members. The inspectors also witnessed visual inspections of welds performed by licensee QC inspectors.

Quality Assurance Activities Associated with Drywell Structural Steel

The inspectors reviewed ongoing activities associated with quality assurance oversight of the drywell structural steel design and construction program. These included work observations performed by Nuclear Assurance personnel. Work observations included the following activities: removal of existing structural steel members, walkdown activities, steel fabrication, and installation of new members. The inspectors reviewed the following Problem Evaluation Reports (PERs) which were initiated to document nonconforming conditions identified during the work observations: PER 02-005124, 03-004643, 03-010594, and 03-010595. The inspectors also reviewed a self-assessment, number RES-REN-02-001 conducted by TVA engineering personnel to assess the adequacy of the design methodology, calculations, and design output documents for the Unit 1 drywell structural steel modifications. Two PERs were initiated as a result of the assessment.

c. Conclusions

No violations or deviations were identified during the preliminary review of the licensee's Drywell Steel Platforms Special Program for Unit 1.

E7 Quality Assurance in Engineering Activities (71152)

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

a. Inspection Scope

The inspectors reviewed licensee procedures relative to QA oversight of modifications and Unit 1 recovery activities. The inspectors also reviewed a sample of QA assessments and observation reports to ensure that adequate oversight was being applied.

b. Observations and Findings

The inspectors reviewed completed Nuclear Assurance (NA) assessments of Unit 1 recovery activities performed by the licensee since May 2002. The inspectors also reviewed a sample of NA Observation Reports documenting QA observations and findings for Unit 1 recovery activities to ensure that adequate oversight was being applied. Specifically, the inspectors' review was to evaluate the effectiveness of the licensee's corrective actions for important safety issues and to assess whether issues were identified in a timely manner; documented accurately and completely; properly classified and prioritized; adequately considered for extent of condition, generic implications, common cause and previous occurrences; adequate to identify root causes; and identify appropriate corrective actions to prevent recurrence. Also, the inspectors' review was to assess whether the issues were processed in accordance with licensee Procedure SPP-3.1, Corrective Action Program.

c. Conclusions

Initial reviews of licensee procedures relative to QA oversight of Unit 1 modifications/recovery activities and sampled QA assessments and observation reports indicated that adequate oversight was being applied. No violations or deviations were identified.

E8 Miscellaneous Engineering Issues (92701)

E8.1 (Closed) Inspector Followup Item 50-259/89-20-02, CRD Seismic Analysis

This item dealt with the need to install CRD Housing lateral restraints due to a new seismic analysis performed by General Electric Company (GE). The CRD Housing lateral restraints were not required in the original stress analysis and were not installed in the field for Mark I containments like the Browns Ferry Plant. Later, GE performed a new stress analysis with the addition of lateral restraints at CRD Housing to upgrade Mark I containments. The licensee is required to add those lateral restraints for their Mark I Containment at Browns Ferry Plant and lateral restraints were added to Unit 2 and Unit 3 prior to restart of those units.

The licensee issued Design Change Notice (DCN) 50985 to implement the requirement of lateral restraints for Unit 1. DCN 50985 will install a CRDH lateral restraint beam structure and individual CRDH lateral restraint clamps similar to the design used on Unit 3. The CRDH lateral restraints on Unit 3 were installed under DCN W17383A. Inspection and closure of this item for Unit 3 had previously been documented in Inspection Report 50-259, 260, 296/95-03.

The inspector reviewed DCN 50985 Rev A. which included the design details of the lateral restraints and stress analysis for the restraint components. Therefore, since 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 performance deficiencies associated with the licensee's corrective actions would have only minor consequences, this item meets closure criteria established for Unit 1 recovery issues. This issue is closed for Unit 1.

E8.2 (CLOSED) LER 50-259/92-03, Failure Of The Reactor Zone Isolation Dampers To Close

This LER reported that the Unit 3 reactor zone inboard and outboard isolation dampers failed to close in response to a test isolation signal. In addition, the 3B refueling zone supply fan failed to trip. The failure of the inboard damper was attributed to a piece of debris lodged in the damper which prevented it from closing. The failure of the outboard damper was attributed to a failed air operated solenoid valve. The failure of the 3B supply fan to trip was determined to be due to a piece of lancing cord stuck in the relay plunger which prevented it from shifting from an energized position. Corrective actions for this event were reviewed and closure of this item for Unit 3 had previously been documented in Inspection Report 50-259, 260, 296/94-32. The inspector determined that no additional unique actions were required for Unit 1. This issue is closed for Unit 1.

V. Management Meetings

X1 Exit Meeting Summary

On July 29, 2003, the resident inspectors presented the inspection results to Mr. John Rupert and other members of his staff, who acknowledged the findings. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

R. Acree, Welding Services Inc. (WSI) QC Supervisor
T. Abney, Nuclear Site Licensing & Industry Affairs Manager
W. Crouch, Mechanical/Nuclear Codes Engineering Manager, Unit 1
R. Cutsinger, Civil/Structural Engineering Manager, Unit 1
B. Ditzler, WSI Site Services Manager
R. Drake, Maintenance and Modifications Manager, Unit 1
R. Jones, Plant Recovery Manager, Unit 1
C. Ottenfeld, Radiological and Chemistry Control Manager, Unit 1
J. Ownby, Project Support Manager, Unit 1
J. Rupert, Vice President, Unit 1 Restart
J. Schlessel, Maintenance Manager, Unit 1
J. Symonds, Modifications Manager, Unit 1
S. Tanner, Nuclear Assurance Manager, Unit 1
J. Valente, Engineering Manager, Unit 1
T. Wiggins, WSI Manager

INSPECTION PROCEDURES USED

IP 49001	Erosion/Corrosion Program
IP 55050	Nuclear Welding General Inspection Procedure
IP 57050	Visual Testing Examination
IP 57060	Liquid Penetrant Testing Examination
IP 57080	Ultrasonic Testing Examination
IP 62002	Inspection of Structures, Passive Components, and Civil Engineering Features at Nuclear Power Plants
IP 71111.08G	Inservice Inspection Activities
IP 71111.17A	Permanent Plant Modifications
IP 71152	Identification and Resolution of Problems
IP 73051	Inservice Inspection Program
IP 92701	Followup

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-259/2003-009-01	URI	Inconsistency for Measurement of Welds in Pipe Supports WDP-BFN-1-CEB-003-01-02-SK-1 and 3 (Section E1.5)

Closed

50-259/89-20-02	IFI	CRD Seismic Analysis (Section E8.1)
50-259/92-03	LER	Failure Of The Reactor Zone Isolation Dampers To Close (Section E8.2)

LIST OF DOCUMENTS REVIEWED

Section E1.1: Piping UT Examination Activities

Procedures, Instructions, and Guidance Documents

NDE Procedure, N-UT-26, Ultrasonic Examination for the Detection of ID Pitting, Erosion, and Corrosion, Rev 21

UT Examination Records

Piping Segments	Component
1-RCIC-X-1	Three inch RCIC steam supply piping
1-RCIC-X-2	Three inch RCIC steam supply piping
1-RCIC-X-3	Three inch RCIC steam supply piping
1-RCIC-X-4	Three inch RCIC steam supply piping
1-HPIC-X-1	Ten inch HPCI steam supply piping
1-HPIC-X-2	Ten inch HPCI steam supply piping
1-ECCS-X-1	30 inch ECCS ring header
1-ECCS-X-2	30 inch ECCS ring header
1-ECCS-X-3	30 inch ECCS ring header
1-ECCS-X-4	30 inch ECCS ring header
1-RHRSW-SA-2	16 inch RHRSW piping
1-RHRSW-SA-3	16 inch RHRSW piping
1-RHRSW-SC-3	16 inch RHRSW piping
Unit 1 Drywell liner	
1-HCV-075-0027	12 inch Core Spray Loop A injection valve
1-HCV-075-0055	12 inch Core Spray Loop B injection valve

Records, Worksheets, and Data

01-009826-000, UT examination of Unit 1 Drywell liner
01-011799-000, UT examination of Unit 1 RHRSW piping
02-011110-000, UT examination of Unit 1 ECCS ring header and HPCI, RCIC piping
98-013880-000, UT examination of Core Spray injection valve, 1-HCV-075-0027
98-013881-000, UT examination of Core Spray injection valve, 1-HCV-075-0055

Section E1.2: Eddy Current Examination of Heat Exchangers

Records

Unit 1 Restart Heat Exchanger Eddy Current Test Results Evaluation and Recommended Action Plan, January 17, 2003
Unit 1 RBCCW Heat Exchanger Report, January 17, 2003
Unit 1 RHR Heat Exchanger Report, January 17, 2003
Unit 1 RHR Heat Exchanger tube plugging history
Eddy Current Examination Report for Unit 1 RHR Heat Exchanger 1B, December 2002
Eddy Current Examination Report for Unit 1 RHR Heat Exchangers 1A and 1C, November 2002

Section E1.3: Permanent Plant Modifications

Procedures, Instructions, and Guidance Documents

MAI-3.9, Installation of Cable Trays, Cable tray Supports, and Cable Tray Covers, Rev. 8
MAI-4.10, Piping Clearance Instruction, Rev 4B
MAI-5.1A, Expansion Shell Anchors Installation, Rev. 19
MAI-5.1B, Wedge Bolt (WB) Anchor Installation, Rev. 24
0-TI-237, Secondary Containment Penetration Breach Analysis, Rev 9
0-TI-272, Control Bay Habitability Zone Penetration Breach Analysis, Rev 7

Modification Packages
DCN 51088A, install new cable trays and raceway components in cable spreading room, auxiliary instrument room, and control room
DCN 51268A, equipment air lock sub-hatch doors

Engineering Calculations
MD-Q0031-920163, Habitability Zone, CREVS
MD-Q0031-920154, Evaluation of Habitability Zone Un-filtered In-leakage

Records and Data
Post modification testing results package,1-PMTI-BF-303.02, Functional testing of equipment access lock doors/subdoors

Work Orders
WO 02-010194-006, Install cable tray for DCN 51088A in auxiliary instrument room, elevation 593
WO 02-010194-012, Install cable tray supports for DCN 51088A in cable spreading room, elevation 606
WO 02-010212-003, Install new equipment airlock sub hatch door 229C for DCN 51268A
WO 02-010212-012, Visual verification and smoke testing for equipment airlock door 226C
WO 02-010212-013, Visual verification and smoke testing for equipment airlock door 229C

Problem Evaluation Reports (PERs)

02-006693-000, Partially cut rebar while core drilling to install electrical penetration
02-016199-000, Rebar damaged while core drilling to install support anchors
03-005391-000, Functional drill stop failure while drilling
03-005426-000, Rebar damaged during drilling when functional drill stop failed to trip
03-005991-000, Craft performed core drills at wrong elevation
03-006312-000, Embedded conduit encountered while drilling in control room floor
03-007434-000, Craft performing core drills drilled into conduit

Section E1.4: Long Term Torus Integrity Program

Procedures and Standards

TVA Browns Ferry Unit 1 Regulatory Framework Letters dated December 13, 2002 and February 28, 2003
NUREG 1232 Vol. 3, SER, Browns Ferry Unit 2 restart, April 1989
NUREG 1232 Vol. 3, Supplement 1, SER, Browns Ferry Unit 2 Restart, October 1989
NUREG 1232 Vol. 3, Supplement 2, SER, Browns Ferry Unit 2 Restart, January 1991
Browns Ferry Unit 1 Restart Nuclear Assurance Verification and Oversight Plan, Rev. 0

Other Documents

Torus External Ring Girders - Walkdown Package Review Status and Forecast
Large Bore Pipe Support Status (Including Torus Attaching Piping Supports) for the Week Ending June 8. 2003

Section E1.5: Large Bore Piping and Supports Program

Procedures and Standards

TVA Browns Ferry Unit 1 Regulatory Framework Letters dated December 13, 2002 and February 28, 2003
NUREG 1232 Vol. 3, SER, Browns Ferry Unit 2 restart, April 1989
NUREG 1232 Vol. 3, Supplement 1, SER, Browns Ferry Unit 2 Restart, October 1989
NUREG 1232 Vol. 3, Supplement 2, SER, Browns Ferry Unit 2 Restart, January 1991
IE Bulletin 79-02, Pipe Support Base Plate Designs Using Concrete Expansion Anchors
IE Bulletin 79-14, Seismic Analysis for As-Built Safety-Related Piping
Browns Ferry Unit 1 Restart Nuclear Assurance Verification and Oversight Plan, Rev. 0
Procedure No. WI-BFN-0-CEB-01, Rev. 1, Browns Ferry Nuclear Plant Walkdown Instructions

Other Documents

Problem Evaluation Report (PER) Nos. 02-007170-000, 02-007299-000, 02-007300-000, and 02-007508-000
Nuclear Assurance Assessment Report No. NA-BF-02-007, Unit 1 Walkdown Program, Dated October 28, 2002
Daily Observation Reports ID 27604, Dated February 4, 2003
Nuclear Assurance Assessment Report No. NA-BF-03-002, Unit 1Residual Heat Removal System (074) Design and Analysis, Dated March 13, 2003

Walkdown Package (WDP) No. WDP-BFN-1-CEB-003-01-02, Rev. 1,
WDP No. WDP-BFN-1-CEB-003-01-01-SK1, Problem No. NI-103-IRA
WDP No. WDP-BFN-1-CEB-003-01-02-SK1(Formerly Support FW-H4), Problem No. NI-103-IRA
WDP No. WDP-BFN-1-CEB-003-01-02-SK3 (Formerly Support FW-H2), Problem No. NI-103-IRA

Section E1.6 Intergranular Stress Corrosion Cracking (IGSCC) - Welding of Replacement of Reactor Vessel Safe Ends

Procedures and Standards

WSI QAP 2.1, Selection, Training, Qualification and Certification of Quality Control Inspection and Test Personnel to ANSI N45.2.6 and ANSI/ASME NQA-1, Revision 9
WPS 08-08-TS-001, Revision 2, including applicable PQRs
WSI Instruction WSI-BF-20.0, Procedure for RECIRC Inlet Safe End Replacement, Revision 1
WSI Instruction WSI-BF-21.0, Procedure for RECIRC Outlet Safe End Replacement, Revision 1
WSI QAP 8.0, Control and Issue of Weld Filler Metal, Revision 7
WSI QAP 9.1, Welding Procedure and Performance Qualification, Revision 8
WSI QAP 9.3, Workmanship and Visual Inspection Criteria for ASME Welding, Revision 13
WSI QAP 9.6, Liquid Penetrant Inspection Procedure

Other Documents

DCN 51045A, U1 Recovery Drywell mechanical Lead System C68
Drawing Change Authorization (DCA) 51045-125 for Drawing 47A1408-4
Work Order (WO) 02-010314-006, Recirculation Outlet Nozzles N1A and N1B
Sample of WSI Welder Qualification Records
Certified Material Test Reports for four heats/lots of welding material
WSI Certificate of Qualification Records for 3 Welding Inspectors

Section E1.7: Drywell Steel Platforms

Procedures and Standards

TVA General Engineering Specification G-89, Requirements for Structural and Miscellaneous Steel, Rev. 3, dated 4/26/94.
TVA General Engineering Specification 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. 12, dated 3/27/03.
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. 9, dated 9/24/96.
Walkdown Instruction WI-BFN-0-GEN-01, General requirements for BFN Unit 1 Walkdowns, Rev. 1, dated 2/22/02
Walkdown Instruction WI-BFN-0-CEB-02, Walkdown Instructions for Seismic Issues (Civil), Rev. 0, dated 3/19/02
General Design Criteria Document BFN-50-C-7100, Design of Civil Structures, Rev. 13, dated 12/20/00

Other Documents

Drawing 1-48E442-1 through 1-48E442-5, Reactor Building Floor Framing, Elev. 563, Rev. 5
Drawing 1-48E442-6, Reactor Building Floor Framing Details, Elev. 563, Sheet 1, Rev. 3
Drawing 1-48E442-7, Reactor Building Floor Framing Details, Elev. 563, Sheet 1, Rev. 5
Drawing 1-48E442-8, Reactor Building Floor Framing Details, Elev. 563, Sheet 3, Rev. 5
Drawing 1-48E442-9, Reactor Building Floor Framing Details, Elev. 563, Sheet 4, Rev. 3
Drawing 1-48E442-10, Reactor Building Floor Framing Details, Elev. 563, Sheet 5, Rev. 4
Drawing 1-48E442-11, Reactor Building Floor Framing , Elev. 563, Rev. 1
Drawing 1-48E442-12, Reactor Building Floor Framing Details, Elev. 563, Sheet 6, Rev. 3
Drawing 1-48E442-13, Reactor Building Mark and Assembly No. Listing, Elev. 563, Sheet 1, Rev. 2
Drawing 1-48E442-14, Reactor Building Mark and Assembly No. Listing, Elev. 563, Sheet 2, Rev. 2
Drawing 1-48E442-15, Reactor Building Floor Framing Details, Elev. 563, Sheet 7, Rev. 0
DCN 51020, Civil Drywell Lower Steel, EL 563
DCN 51019, Civil Drywell Lower Steel, EL 584
Calculation number CDQ1-303-2002-0245, Lower Drywell Floor Steel Framing Modifications at Elevation 563
Calculation number CDQ1-303-2002-0246, Connections for Lower Drywell Floor Steel Framing Modifications at Elevation 563
Work Order numbers 02-008180-005, 006, -007, and -008, for modification of Elevation 563 drywell structural steel between azimuths 81° and 15°

Section E7.1: Licensee QA Oversight Activities

Audits, NA Assessments, NA Observations, Self-Assessments, and Problem Evaluation Reports (PERs)

Assessment NA-BF-02-007, Unit 1 Restart Walkdown Program
Assessment NA-BF-03-001, Unit 1 Restart Power Service Shop
Assessment NA-BF-03-002, Unit 1 RHR System Design and Analysis
Observation 26314, Drywell steel DCNs 51019 and 51020
Observation 27330, Unit 1 modifications constructability
Observation 27404, Welding procedure qualification
Observation 27855, Drywell steel platforms and cable trays
Observation 28025, Drywell floor structural modification
Observation 28122, Walkdowns to support drywell steel design
Observation 28264, Lower drywell steel framing modifications
Observation 28281, Design and analysis of drywell steel
Self-Assessment Report RES-REN-02-001, Unit 1 drywell floor steel structural modification