UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W., SUITE 2900
ATLANTA, GEORGIA 30323-0199

Report Nos.: 50-259/95-44, 50-260/95-44, and 50-296/95-44

Licensee: Tennessee Valley Authority
6N 38A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Docket Nos.: 50-259, 50-260 and 50-296

License Nos.: DPR-33, DPR-52, and DPR-68

Facility Name: Browns Ferry Nuclear Power Station Units 1, 2, and 3

Inspection Conducted: July 31, 1995 - August 4, 1995

Inspector: J. L. Coley, Jr. — 8-11-95 Date Signed
J. L. Coley, Jr.

Approved by: JJ Blake — 8/15/95 Date Signed
J. J. Blake, Chief
Materials and Processes Section
Engineering Branch
Division of Reactor Safety

SUMMARY

Scope:

This routine, announced inspection was conducted in the area of followup inspection on previous violations, unresolved items, and other Unit 3 re-start items (Inspection Procedures No. 92701 & 92702).

Results:

In the areas inspected, violations or deviations were not identified. This report will close violation No. 50-296/95-03-01, unresolved item No. 50-259,260,296/93-11-01, Browns Ferry Nuclear Performance Plan Volume 3, Section III.7.0 Commitment Item No. 96, Generic Letter 88-01 and Generic Letter 94-03 for Unit 3.

Enclosure

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PDR ADOCK 05000259
Q PDR

## REPORT DETAILS

1. Persons Contacted

Licensee Employees

*C. Crane, Assistant Plant Manager
*J. Dollar, Outage
*E. Hartwig, Project Manager
*L. Madison, Unit 3, Civil Engineer
*D. Matherly, Operations
*G. Preston, Plant Manager
*P. Salas, Licensing Manager
*S. Wetzel, Licensing
*H. Williams, Engineering and Materials

Other licensee employees contacted during this inspection included engineers, technicians, and administrative personnel.

NRC Resident Inspectors

J. Munday, Resident Inspector
R. Musser, Resident Inspector
M. Morgan, Resident Inspector
*L. Wert, Senior Resident Inspector

*Attended exit interview

Acronyms and initialisms used throughout this report are listed in the last paragraph.

2. Followup on Previous Inspection Findings (Inspection Procedures No. 92701 and 92702)

(Closed) Unresolved Item No. 50-259,260,296/93-11-01, "Weld Differences Between the Welds Assumed in Support"

During an inspection in the spring of 1993 an inspector observed a reactor water clean-up (RWCU) pipe support being welded. During subsequent discussions with the welder, field engineer, welding foreman and design engineers, the inspector was concerned that there was a communication problem between the design engineers and the welders. In his inspection report the inspector stated that; "The licensee should therefore review, and revise as necessary Figure 5.1 of Bechtel's Computer Program Users Manual FAPPS ME 150; support drawing general notes; Welding Procedure G-29, and other pertinent documents to provide adequate communication so that welders will fabricate the weld connections that design engineers want and assume in the design calculations."

The weld configuration in question was the welding of the end of a wide flange section to an embedded plate or into the flange of another beam. The weld symbol showed a fillet weld with three arrows. One arrow went to the inside of the top flange, one to the web and the third went to the inside of the bottom flange. This created a "U" shaped configuration. The welder made three separate welds with no weld deposited at the radius of transition between the web and the flanges. The FAPPS ME 150 program considered the weld to be continuous at the radius resulting in one full length weld.

AWS A2.4 shows that when a continuous "U" shaped weld is desired, the weld symbol to be used is a fillet weld with three arrows pointing to the three straight segments of the weld. The inspector was concerned that this was called a desired weld and was not a mandatory requirement.

In order to insure that the weld symbol was not mis-interpreted again the licensee revised figure 1 of Process Specification 1.C.1.2 of TVA's General Welding Procedure G-29 to depict the extent of the weld as being entirely around the radius when three arrows are pointing to three straight segments of the weld. The licensee concluded that no other document needed to be revised since the clarification in G-29 will assure that the design intent of a continuous weld will be fabricated in the plant.

To determine the significance of this item on welding at Browns Ferry the inspector accompanied by a TVA civil engineer performed a random inspection of supports at various elevations in the Units 2 and 3 reactor buildings. Only four supports were questioned as a result of this walkdown inspection and after referring to the applicable drawings for these supports all were found to be satisfactory. Therefore, the inspector considers the mis-interpretation of the welding symbols on the support witnessed by the previous inspector to be an isolated instance. In addition, TVA's clarification of G-29 should prevent this problem from re-occurring since welding personnel are trained to the requirements of G-29. This issue is considered closed for all three Units at Browns Ferry.

(Closed) Violation No. 50-296/95-03-01, "Spring Can Installation"

This item dealt with an inspector's finding that an incorrect spring can was installed on Support No. 3-47B452-3035. The inspector's preliminary inspection indicated that the spring can in the field was a BP3100-19 type C, and was installed per Rev. 001 of DCA W17701-455. Rev. 002 to DCA W17701-455 was apparently issued October 4, 1993 via FDCN F26777A, and was incorporated in Work Package (WP) No. WP-3857-93 on March 3, 1994. Rev. 002 specified the installation of BP 3200-19 type C spring can. Work on this support was signed off as completed on March 31, 1994.

The licensee issued Problem Evaluation Report (PER) No. 950058 to investigate this finding and to take the appropriate corrective action. The investigation performed by the licensee revealed that the correct spring can was still in the warehouse. A requisition form to procure this spring can however, had been signed off by the Foreman/Designee. In addition, the foreman and QC inspector signed off the data sheet in the work package without verifying the installed component to be the component specified in the drawing.

The licensee's investigation revealed that there was no evidence that a new pre-job briefing was performed on WP-3857-93 (or the sub-package) when support 3-47B452-3035 was transferred to another shift for installation. Additionally the subtle but important change in the design output document, as documented in the Work Plan (WP) was not verbally conveyed between the shift that performed the work.

A turnover was performed by the foreman which consisted of providing the WP (for the support) and showing the crew assuming the support installation responsibilities the previously staged material in the lay-down storage area.

On March 23, 1994 through March 31, 1994 the support was installed using the pre-staged materials as pointed out to the new shift by the other shift supervisor. The subtle differences between DCA W17701-455 Rev. 001 and 002 went un-noticed by the shift performing the work.

The licensee's corrective actions consisted of the following:

- Reinspection of Support No. 3-47B452-3035 was performed

- The correct Spring Can (BP 3200-19 Type C) was installed

- The support inspector was interviewed/monitored by the TVA Corporate Level III Mechanical Inspector to determine whether he had the technical knowledge required to perform spring can inspections. The Level III concluded that the QC inspector was knowledgeable of his examination responsibilities.

- The support inspector received additional training in the requirements MAI 4.2A and the interpretations of 47B435 series of General Notes.

- A memorandum was issued to all craft supervisors, lead foreman, foreman and field engineers with the instructions to require field verification of unique items such as component standard support hardware, including serial Nos., prior to making late entries on Form SSP-80 unless existing documentation in the work plan confirms the correct 575 (requisition form).

- Training was given to 238 Unit 3 QC support and modification craft personnel concerning this violation, "paying attention to work details." Unit 2 personnel also reviewed the circumstances that led to this violation.

- A sample size of 60 pipe supports were randomly selected to ensure that the material identification problem was isolated. The sample size included 27 inspectors and the supports were reinspected per the applicable attributes contained in Modification and Addition

Instruction 4.2A. No material identification problems were identified during the reinspection effort which included various vendor supplied materials such as spring cans, sway struts and pipe clamps.

- A review of eighty-one open Unit 3 Problem Evaluation Reports (PERs) was performed. From the review, thirty-one PERs were categorized as being a potential negative trend in the area of pre-plant acceptance deficiencies. PER 950151 was initiated and TVA established an Incident Investigation Team to address the broader scope aspects of other events involving improper implementation of Unit 3 work plans/procedures. The teams responsibility was to identify any common cause factors, and determine appropriate corrective action needed. NRC's resident inspector issued Inspector Followup Item (No. 296/95-16-04) to track the licensee corrective actions on the items identified by the Incident Investigation Team.

To verify that the QC inspector was knowledgeable of the requirements in MAI 4.2A relating to final inspections of supports, the inspector re-examined eight supports which had been examined by the QC inspector prior to the violation. The supports included seven snubbers and one spring can. All of the supports examined were found to be satisfactory. The Unit 3 supports re-examined by the inspector were identified as RSSF-1, RSSF-2, RSSE-3, RSSE-2, RSSB-1, RSSB-2, 3-47B452-3038, and 3-47B452-3040. As a result of the 8 re-examinations performed by the inspector and the 60 re-examinations performed by the licensee the inspector concluded that this violation was an isolated failure caused by in-attention to details.

Within the areas examined, no violation or deviation was identified.

3. Followup on Miscellaneous Unit 3 Startup Issues (Inspection Procedure No. 92701)

a. Browns Ferry Nuclear Performance Plan Volume 3, Section III.7.0 Commitment Item No. 96 "Shroud Head Bolts IGSCC" Unit 3

General Electric (GE) Service Information Letter (SIL) No. 433, issued February 7, 1986, informed GE Boiling Water Reactor (BWR) owners that cracking had occurred in the Inconel 600 shaft in a crevice region of a shroud head bolt. Since then, several cracked shroud bolts of the original design have been found in pre-BWR/6 plants. A complete failure of the shaft of one of the original design shroud head bolts occurred in 1993 at a GE BWR/4. However, the most recent failure location was different from that described in SIL No. 433. Therefore, GE issued a supplement to SIL No. 433 on September 15, 1993 to inform GE BWR owners of the recent failure.

The recent failed bolt separated approximately 68 inches above the bottom of the bolt at the weld connection between the lower portion of the NiCrFe Inconel alloy 600 rod and the stainless steel stud. The lower portion of the bolt, approximately 68 inches long, which includes the Inconel tee section, was found resting on a group of jet pump sensing lines in the annulus between the shroud and the vessel wall. The bolt tee section failed to disengage during unbolting after the upper segment of the bolt shaft had been rotated. This caused the lower bolt segment to separate when the shroud head was lifted. A visual inspection of the fractured surface indicated that essentially the entire cross section of the bolt was cracked before the plant personnel attempted to untorque the bolt.

The cause of the cracking described in SIL No. 433 and No. 433 Supplement 1 was confirmed to be Intergranular stress corrosion cracking (IGSCC). In response to these occurrences GE recommended that owners perform ultrasonic examination of the shroud head bolts the next time the bolts were accessible.

SIL No. 433 also informed GE BWR owners that GE had redesigned the shroud head bolt to eliminate the crevice condition in the Inconel collar. Other design changes reduced the probability of failure in other IGSCC-susceptible parts of the bolt. The new bolt design includes the following improvements:

- The bolt shafts are solution heat treated after welding, including the weld between the Inconel rod and the stainless steel stud. This significantly reduces weld residual stresses which contribute to IGSCC.

- The bolts no longer contains Inconel 182

- Quarter-inch holes are drilled in the outer sleeve to enhance water circulation between the bolt and the sleeve, thereby reducing the probability of IGSCC resulting from stagnated water.

- The stainless steel components, including the stud, are low carbon type 316 material.

- The nut is made of nitrated XM-19 material instead of the nitrated type 304 originally used.

TVA committed in their nuclear performance plan ( Volume III, page 57, Section 7.0 commitment No. 96) to replace Unit 3 shroud head bolts as needed prior to the restart of Unit 3. Ultrasonic testing found that thirty bolts were defective in Unit 3 and the licensee replaced these bolts using maintenance request forms A-762994 and A-762996 in 1987. In addition, an augmented program for periodic re-inspection of the "Old Design Head Bolts" was implemented into TVA's Inservice Inspection Plan (S.I. 4.6.G 1/3).

TVA however, subsequently decided to replace all of the shroud head bolts with the new design prior to restart of Unit 3. This work was performed in accordance with Design Change Notice (DCN) G-18595A and Work Order (WO) 89-07629-96. At the present time all 48 shroud head bolts are the new material/new design head bolts.

b. Review of TVA's Actions Regarding Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping", Generic Letter 94-03, "IGSCC of Shrouds in Boiling Water Reactors", and Other IGSCC Issues on Reactor Vessel Internals for Browns Ferry Unit-3

The inspector held discussions with TVA's Project Engineer for Piping and Pressure Vessels and reviewed TVA/NRC correspondence relating to the status of TVA's actions to comply with Generic Letter 88-01, Generic Letter 94-03, and other NRC/Industry initiatives for the mitigation of Intergranular Stress Corrosion Cracking (IGSCC). The following documents were reviewed and discussed with the licensee:

- NRC letter to TVA dated January 13, 1995, "Browns Ferry Nuclear Plant Units 1, 2, and 3 Safety Evaluation of Response to Generic Letter 94-03 (IGSCC of Core Shroud in BWRs"

- NRC letter to TVA dated August 27, 1994, "Summary of August 11, 1994 Meeting with TVA Regarding Core Shroud Inspection Plans for the Browns Ferry Nuclear Plant - Units 1, 2, and 3"

- TVA letter to NRC dated August 23, 1994, "Response to NRC Generic Letter 94-03 - IGSCC of Shrouds in Boiling Water Reactors - Units 1, 2, and 3"

- NRC letter to TVA dated December 3, 1993, "Browns Ferry Nuclear Plant Unit 3 - Safety Evaluation of Supplemental Response to Generic Letter 88-01"

- NRC letter to TVA dated February 18, 1993, "Alternative Inservice Inspection Methods for the Reactor Water Clean-up and Residual Heat Removal Systems"

- TVA letter to NRC dated January 19, 1993, "Implementation Plans for Hydrogen Water Chemistry Control (HWCC)"

- TVA letter to NRC dated December 28, 1992, "Supplemental Response to Generic Letter 88-01, NRC Position On IGSCC in BWR Austenitic Stainless Steel Piping - Unit 3"

- TVA letter to NRC dated November 25, 1992, "Request for NRC Approval of Alternate Methods for the Reactor Water Cleanup

(RWCU) and Residual Heat Removal (RHR) Inservice Inspections Required by Generic Letter 88-01 - Units 1, 2, and 3"

- TVA letter to NRC dated July 13, 1990, "Supplemental Response to Generic Letter 88-01 - Units 1, 2, and 3"

- NRC letter to TVA dated December 21, 1989, "Evaluation of TVA's Supplemental Responses to Generic Letter 88-01 - Units 1, 2, and 3"

- TVA letter to NRC dated June 30, 1989, "Supplemental Response to Generic Letter 88-01"

- NRC letter to TVA dated May 19, 1989, " Technical Specification Changes in Accordance With Generic Letter 88-01 - Units 1, 2, and 3"

- NRC letter to TVA dated April 24, 1989, "Summary of Meeting With Tennessee Valley Authority Held on April 20, 1989, to Discuss Generic Letter 88-01 Response for Reactor Water Cleanup System Piping Outside the Drywell Penetrations"

- TVA letter to NRC dated December 9, 1988, "TVA Browns Ferry Nuclear Plant Technical Specification No. 262 - Intergranular Stress Corrosion Cracking"

- NRC letter to TVA dated December 8, 1988, "Generic Letters 84-11 (Inspection of BWR Stainless Steel Piping) and Generic Letter 88-01 (NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping) - Evaluation of Browns Ferry, Unit 2 Responses

- TVA letter to NRC dated August 1, 1988, "Response to Generic Letter 88-01, NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping, Dated January 25, 1988"

- Browns Ferry Nuclear Performance Plan, Volume III, Section 7.0 (Intergranular Stress Corrosion Cracking)

- TVA Inservice Inspection Program (Surveillance Instruction No. 4.6.G 1/3) " Augmented Inspections for Reactor Vessel Internals"

The inspector's review of documents and discussions with cognizant engineers revealed that TVA has an aggressive program to mitigate the long term effects of IGSCC. Actions required by Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping," and Generic Letter 94-03, "IGSCC of Core Shrouds in BWRs," have been performed by the licensee or alternative methods approved. NRC issued a satisfactory Safety Evaluation Report on the licensee's response to Generic Letter 88-01 for Unit 3 on December 3, 1993. The licensee's response to Generic Letter 94-03 was also found to be satisfactory and NRC issued a Safety Evaluation Report on January 13, 1995.

The inspector's review of TVA's actions for long term mitigation of IGSCC of Unit 3 reactor vessel internals were also found to be aggressive and satisfactory. These issues are identified primarily by General Electric in Surveillance Instruction Letters to BWR owners.

No IGSCC issue was found as a result of the above reviews, that would negatively impact Unit 3 Re-start.

Note: Generic Letter 88-01 superceded Generic Letter 84-11 on BWR pipe cracking. Safety Issues Management System (SIMS) item 86, Long Range Plan Dealing with Stress Corrosion Cracking in BWR Piping, established TI 2515/089 to follow-up on GL 84-11, however the TI was cancelled with the issue of GL 88-01. For administrative tracking purposes, this inspection report closes SIMS item 86 for Unit 3.

Within the areas examined, no violation or deviation was identified.

4. Exit Interview

The Inspection scope and results were summarized on August 4, 1995, with those persons indicated in paragraph 1. The inspector described the areas inspected and discussed in detail the inspection results listed below. Proprietary information is not contained in this report. Dissenting comments were not received from the licensee.

(Closed) Violation No. 50-296/95-03-01, "Spring Can Installation", paragraph 2

(Closed) Unresolved Item No. 50-259,260,296/93-11-01, "Weld Differences Between the Welds Assumed in Support," paragraph 2

5. Acronyms and Initialisms

| | | |
|---|---|---|
| AWS | - | American Welding Society |
| BWR | - | Boiling Water Reactor |
| DCA | - | Design Change Authorization |
| DCN | - | Design Change Notice |
| GE | - | General Electric |
| IGSCC | - | Intergranular Stress Corrosion Cracking |
| MAI | - | Modification and Addition Instruction |
| No. | - | Number |
| NRC | - | Nuclear Regulatory Commission |
| PER | - | Problem Evaluation Report |
| QC | - | Quality Control |
| Rev. | - | Revision |
| RWCU | - | Reactor Water Cleanup System |
| SI | - | Surveillance Instruction |
| SSP | - | Site Standard Procedure |
| TVA | - | Tennessee Valley Authority |
| WO | - | Work Order |
| WP | - | Work Plan |