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

Report Nos.: 50-259/95-39, 50-260/95-39, and 50-296/95-39

Licensee: Tennessee Valley Authority
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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 24-28, 1995

Inspector: S. Rudisail — 8/24/95
S. Rudisail — Date Signed

Approved by: for — 8/25/95
M. Shymlock, Chief — Date Signed
Plant Systems Section
Engineering Branch
Division of Reactor Safety

SUMMARY

Scope:

This routine, announced inspection was conducted in the areas of Post Accident Monitoring (PAM) instrumentation for implementation of the licensee's commitments for Regulatory Guide (RG) 1.97, Instrumentation for Light-Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.

Results:

In the areas inspected, violations or deviations were not identified.

The licensee actions for implementation of the RG 1.97 program for PAM were found to be adequate. The inspector reviewed drawings, calculations, the licensee's design criteria, and the Unit 3 RG 1.97 Compliance Report. Walkdowns were performed of selected instrumentation to ensure adequate implementation of the RG 1.97 program. The inspector concluded from these reviews that the Unit 3 RG 1.97 program was being adequately implemented.

Enclosure

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The inspector also reviewed Corrective Action Tracking Documents (CATDs) associated with cable installation issues. The corrective actions documented for CATD 23900-BFN-04 and CATD 23900-BFN-09 were reviewed and found to address the issues adequately.

REPORT DETAILS

1. Persons Contacted

Licensee Employees

*T. Abney, Unit 3 Recovery Manager
*T. Chandler, Electrical Engineer
*C. Crane, Assistant Plant Manager
*H. Crisler, Site Engineer
*J. Davenport, Licensing Engineer
*B. Endsley, Maintenance Engineer
*C. Galuska, Site Engineer
*D. Gurber, Maintenance Training Engineer
*S. Hilmen, Technical Support I&C and Electrical Manager
*J. Johnson, Site Quality Manager
*T. Langley, I&C Principal Engineer
*J. Maddox, Maintenance/Modifications Manager
*R. Maintenance Planning & Technical Manager
*J. McCarthy, Lead, Mechanical Nuclear Engineer
*G. Pierce, Technical Support Manager
*B. Pratt, Corporate Maintenance
*G. Preston, Browns Ferry Nuclear Plant Manager
*P. Salas Licensing Manager
*T. Shriver, Manager NA&L
*L. Turner, Technical Support System Engineer
*S. Wetzel, Acting Compliance Manager
*H. Williams, Engineering and Materials

Other licensee employees contacted during this inspection included craftsmen, engineers, operators, mechanics, security force members, technicians, and administrative personnel.

Other Organizations

*W. Peabody, Vice President Atwater

NRC Employees

*R. Musser, Resident Inspector
J. Munday, Resident Inspector
*M. Shymlock, Section Chief, Plant Systems Section

*Attended exit interview

2. Review of RG 1.97 Implementation for Unit 3 (IP 37550)

The inspector reviewed the licensee's implementation of the RG 1.97 program for Unit 3 PAM instrumentation. RG 1.97 describes an acceptable method for complying with the Nuclear Regulatory Commission (NRC) regulations to provide instrumentation to monitor plant variables and systems during and following an accident. The NRC Safety Evaluation Reports (SERs) for RG 1.97 were issued January 23, 1985, June 23, 1988, and February 8, 1990. The staff conclusion of the SERs was that the licensee's design with respect to conformance to RG 1.97 was acceptable.

The licensee documented the implementation of the RG 1.97 instrumentation program for Unit 3 in the Regulatory Guide 1.97 Compliance Report for Unit 3 Restart. This report documented the licensee's review of engineering documentation for each RG 1.97 component. Each component identified as RG 1.97 instrumentation was reviewed to ensure the attributes identified with RG 1.97 for PAM instrumentation were satisfied. A reference document for the acceptability of the technical basis for each component was also identified within this report. This report documented the licensee's review for each loop component from the sensor to the indicator/recorder. Loop power supplies were reviewed back to the distribution boards.

For this inspection a sample of instrumentation was selected for review. The sample was selected based on the instrumentation identified in the SER requiring additional action for Unit 3, instrumentation for systems identified in the Browns Ferry Multi-Unit PRA Main Report as important systems, and a random sample of other instrumentation variables. The inspector reviewed each loop for the following items as identified in the Temporary Instruction (TI) 2515/087, Inspection of Licensee's Implementation of Multiplant Action A-17: Instrumentation for Nuclear Power Plant to Assess Plant and Environs Conditions During and Following an Accident. The attributes identified and reviewed were:

(1) Equipment Qualification - The Equipment Management System database was reviewed to ensure EQ designations were appropriate for the specified instrumentation.

(2) Redundancy - Walkdowns and a review of drawings was performed to ensure that separation and redundancy requirement were met for Category 1 variables.

(3) Power Source - Drawings and the RG 1.97 report was reviewed to ensure that the source of power for instrumentation was safety-related as required.

(4) Quality Assurance - The RG 1.97 report was reviewed to determine that EQ designation were applied as required.

(5) Display and Recording - Walkdowns were performed to verify the method and number of indicators per instrumentation variable.

(6) Range - The range of instrumentation was verified by walkdown to verify this was as stated in the licensee RG 1.97 report.

(7) Equipment Identification - Equipment identification was reviewed from walkdowns to ensure the labels agreed with the design requirements.

(8) Equipment Interfaces - Drawings and the RG 1.97 Report were reviewed to ensure that equipment was isolated as required.

(10) Direct Measurement - The selected variables were reviewed to determine direct measurement by the designated sensors.

These are the attributes identified for instrumentation designated as Category 1. Other categories have similar inspection attributes. Browns Ferry identifies their instrumentation as Categories 1, 2, and 3. The requirements for each of these categories was identified in Design Criteria No. BFN-50-7307, Browns Ferry Nuclear Plant Post Accident Monitoring. This design criteria identifies each instrumentation variable needed for PAM and assigned a category and a variable type as determined by the RG 1.97 guidance.

The inspector selected the following variables for review.

(1) HPCI Flow, Loop and Instrument 3-FIC-73-33

(2) LPCI Flow, Loop and Instruments 3-FI-74-50, 3-FI-74-64, and 3-FR-74-64

(3) Reactor Coolant Level, Loop and Instruments 3-LI-03-52, 3-LI-03-62A, 3-LI-03-58A and 3-LI-03-58B

(4) Residual Heat Removal Service Water Flow, Loop and Instruments 3-FI-23-36, 3-FI-23-42, 3-FI-23-48, and 3-FI-23-54

(5) Core Spray System Flow, Loop and Instruments, 3-FI-75-21 and 3-FI-75-49

(6) Reactor Coolant System Pressure, Loop and Instruments, 3-PI-3-74A and 3-PI-3-73B

(7) Primary Containment Isolation, Loop and Instruments, 3-FCV-73-2, 3-FCV-73-3, 3-FCV-73-26, 3-FCV-73-27, 3-FCV-73-30, and 3-FCV-73-81

(8) Emergency Ventilation Damper Position, Loop and Instruments, 3-FCO-64-5, 3-FCO-64-6, 3-FCO-64-9, 3-FCO-64-10, 3-FCO-64-13, 3-FCO-64-14, 3-FCO-64-40, 3-FCO-64-41, 3-FCO-64-42, and 3-FCO-64-43

(9) Primary Containment Isolation Valve Position Indication, Loop and Instruments, 3-FCV-73-53, 3-FCV-73-57, 3-FCV-73-58, 3-FCV-73-60, 3-FCV-73-61, 3-FCV-73-67, 3-FCV-73-71, 3-FCV-73-72, 3-FCV-73-74, and 3-FCV-73-75.

In addition to the sample selected for Unit 3 restart review, the inspector verified the actions being taken to address the RG 1.97 issues identified in Safety Evaluation Report (SER) NUREG-1232, Volume 3, Browns Ferry Nuclear Performance Plan for the Browns Ferry Unit 2 Restart issued on January 23, 1991. Section 3.2.2.1 of this SER documented the Nuclear Regulatory Commission (NRC) evaluation and acceptance of the adequacy of the licensee providing upgraded instrumentation for core spray flow, low pressure coolant injection flow, residual heat removal system flow, and emergency ventilation damper position variables prior to restart of Unit 3. Additionally, in the SER dated February 8, 1990, the staff identified the following variables requiring additional action by the licensee. The licensee committed to provide EQ instrumentation for the RHRSW instrument loops and to provide upgraded cable for the cable passing through the Unit 3 Reactor Building for the cooling water temperature for ESF components instrumentation. The inspector verified that these actions were complete during the review of the Unit 3 RG 1.97 instrumentation.

During walkdowns and other inspection reviews the inspector did not identify any concerns. A minor problem was identified with the labeling of two valves on the Control Room panels. The labels for valves 3-FCV-84-8C and 3-FCV-84-8D on panel 3-9-55 were the wrong color. Drawing 0-47B601-0-2 established the requirements for RG 1.97 identification in the control room. Design Change Authorization T32086-057 clarified these labeling requirements to require black labels with white lettering for these RG 1.97 indicators. The labels installed were white labels with black lettering. The licensee was aware of this labeling error and had informally documented this discrepancy. The inspector asked how this would be tracked to ensure it was completed. The licensee initiated a Problem Evaluation Report BFPPER950936 to ensure this error was corrected.

During the review of the RG 1.97 Report for Unit 3 restart, the inspector noted that the licensee had identified open item discrepancies which would be required to be addressed prior to restart. The following items were identified:

(1) No surveillance instructions existed for Unit 3.

(2) Flow indicating switch 3-FS-071-0035 is in a Class 1E circuit and is located in a harsh environment but has not been environmentally qualified.

(3) Valve 3-FCV-73-81 has a MOV heater cable, 3ES4456-II still connected to the terminal block at JB6128, but has been disconnected at the valve. The cable needs to be completely disconnected.

(4) FDCN F31344 has been generated against DCN W18742 to replace cables 3ES3176-II and 3ES3151-II for EQ concerns. This FDCN has not been issued.

(5) Non-divisional cables 3PL2263 and 3P12264 have no separation from the divisional PAM loops 3-FCV-75-57 and 3-FCV-75-58.

The inspector reviewed the status of these open items and how they are being addressed. The surveillance instructions for Unit 3 were in progress during this inspection. The remaining items were addressed by PERs BFPER940757 and BFPER940758. These PERs were closed and the actions required to address the above discrepancies have been completed. The inspector determined that the corrective actions of these PERs adequately addressed the discrepancies identified in the RG 1.97 Report.

3. Review of CATDs (IP 92701)

The inspector reviewed the licensee's corrective actions for Unit 3 applications of electrical issues that were identified as CATDs. The inspector reviewed CATD 23900-BFN-09 and CATD 23900-BFN-04. This inspection assessed the adequacy of the licensee's action in resolving the identified concern.

CATD-23900-BFN-09 identified a concern that the TVA program of incorporating cable lengths of installed cables per Policy Memo PM 87-26 was not completed.

PM 87-26 established a policy for choosing values of cable length to be used in short circuit and voltage drop calculations. PM 87-26 applied to those calculations which were completed using the ELMS-AC software. Browns Ferry does not use this software. Calculations at Browns Ferry which require cable length information were completed for Unit 3 using E-CALC, MAPS 108, and MAPS 109 software. This information and software was applicable to Unit 3 baseline calculations for short circuit and voltage drop determination. The inspector reviewed the method which incorporates cable length in the calculations. General Engineering Specification G-38 requires the actual installed cable length to be documented for all medium, low, and control voltage cables. This information is then required to be transmitted to engineering in accordance with site instructions. Modification and Addition Instruction (MAI) 3.3, Cable Terminating and Splicing for Cables Rated up to 15,000 Volts provides instruction for ensuring this information is provided to engineering. Site Engineering is then responsible for ensuring that this information is utilized properly for determination of voltage drop and short circuit information.

The inspector considered this response to CATD 23900-BFN-09 adequately addressed the concern. The inspector considered the implementation of the site procedure was adequate for ensuring the cable length information was adequately documented and transmitted. This was also determined from previous inspections for cable installation activities and calculation reviews. This was being tracked by the licensee as commitment Nuclear Central Office (NCO) 930185055 for Unit 3 restart.

CATD 23900-BFN-04 identified a concern that verified cable data for cable weight and cable diameter was not being entered into computerized cable routing program. Also, all the cable weight data for Browns Ferry has not been incorporated into the design standards DS-E12.1.13 and DS-E12.1.14.

The inspector reviewed the licensee corrective action identified to address these concerns. Significant Condition Reports (SCR) BFNECB8601 and BFNEEB8602 were issued to identify the concerns being tracked by CATD 23900-BFN-04. Browns Ferry utilizes the TVA on-line Mark Number Database (ON-MARK) to track cable weight information and cable outside diameter information. This database has superseded the design standards which previously contained this cable information. This information has been utilized in ampacity calculations used to determine derating factors for cable due to tray fill. Each time a new cable is added to a cable tray an ampacity calculation is required. This additional cable also results in the notification of the Lead Civil Engineer requiring seismic evaluation of the cable tray support affected by the new cable. The Lead Civil engineer is responsible for maintaining data bases for cable tray loading. These data bases are calculations that are revised each time a request for a new cable pull is initiated. The adequacy of existing tray and conduit supports was determined in calculation CD-Q000-931227, Qualification of Cable Tray and Conduit System.

The inspector also reviewed during previous inspections the methods used to track tray and conduit fill for new cable installation and the method for determining tray and conduit fill for previously installed cable. The inspector considers the corrective actions to be completed for this CATD 23900-BFN-04 adequate for resolution of the concern. This was being tracked as NCO 930185055 and NCO 910060002 for Unit 3.

4. Exit Meeting

The inspection scope and results were summarized on July 28, 1995, with those individuals indicated in paragraph 1. The inspector described the areas inspected and discussed in detail the inspection findings. There was no dissenting comments received from the licensee. Proprietary information is not contained in this report.

5. Acronyms and Abbreviations

CATD	Corrective Tracking Action Documents
DCA	Design Change Authorization
DCN	Design Change Notice
EQ	Equipment Qualification
FCV	Flow Control Valve
FCO	Flow Control Operator
FDCN	Field Design Change Notice
FI	Flow Indicator
FIC	Flow Indicating Controller
FR	Flow Recorder
HPCI	High Pressure Coolant Injection

IP	Inspection Procedure
LI	Level Indicator
LPCI	Low Pressure Coolant Injection
MAI	Modification and Addition Instruction
NRC	Nuclear Regulatory Commission
PI	Pressure Indicator
PAM	Post Accident Monitoring
PER	Problem Evaluation Report
PRA	Probabilistic Risk Assessment
RG	Regulatory Guide
QC	Quality Control
SCR	Significant Condition Report
SER	Safety Evaluation Report
TI	Temporary Instruction