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

EA 00-163 July 27, 2000

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 - NRC INSPECTION REPORT 50-259/00-03, 50-260/00-03, 50-296/00-03 AND NRC OFFICE OF INVESTIGATIONS REPORT NO. 2-1999-028.

Dear Mr. Scalice:

On June 24, 2000, the NRC completed an inspection at your Browns Ferry 1, 2, & 3 reactor facilities. The report, Enclosure 1, presents the results of that inspection which were discussed on June 30 and July 20, 2000, with Mr. J. Herron and other members of your staff. In addition, an investigation of activities at Browns Ferry was conducted by the NRC Office of Investigations (OI) between September 21, 1999 and June 15, 2000.

The inspection was an examination of activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and the conditions of your license. In addition to the routine baseline inspection program inspections, a supplemental inspection was conducted in accordance with Inspection Procedure 95001, Inspection for One or Two White Inputs in a Strategic Area. The purpose of the inspection was to assess your evaluation associated with a Unit 3 White performance indicator (Safety System Unavailability for the Heat Removal System, Reactor Core Isolation Cooling), determined to be White in the 4th quarter of calendar year 1998. Within the inspection areas, the inspection consisted of a selective examination of procedures and representative records, observations of activities, and interviews with personnel. The purpose of the investigation was to determine whether an individual employed by Browns Ferry deliberately failed to perform measuring and test equipment (M&TE) nonconformance evaluations as required by site procedures. A summary of the results of the OI investigation is provided as Enclosure 2.

Based on the results of the inspection and OI investigation, an apparent violation was identified and is being considered for escalated enforcement action in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions - May 1, 2000" (Enforcement Policy), NUREG-1600. The apparent violation involves the failure to adhere to the requirements of Technical Specification 5.4.1 and Tennessee Valley Authority's (TVA's) procedures related to out of tolerance measuring and test equipment (M&TE). As background, NRC Inspection Report 50-259, 260, 296/00-01 documented an unresolved item involving the potential failure to conduct M&TE nonconformance evaluations. During a self-assessment in June 1999, your staff identified that procedurally required actions had not been taken for numerous pieces of M&TE when out-of-tolerance reports were generated following testing by TVA's Central Laboratory Field Testing Services (CLFTS). Your review determined that, during the period from June 1997 to June 1999, from the population of M&TE identified by CLFTS to be out-of-tolerance, approximately 500 nonconformance evaluations were not properly issued and/or dispositioned for components tested or inspected using the out-of-tolerance M&TE. Your reevaluation of the nonconformance evaluations revealed that component operability was not affected. The OI investigation determined that there were deliberate aspects to the failure of a TVA employee to initiate and/or disposition nonconformance evaluations on the out-of-tolerance M&TE. Section 1R19 of the enclosed inspection report and Enclosure 2 describe the results of our review of both the technical and deliberate aspects of this issue.

The circumstances surrounding this apparent violation, the significance of the issue, and the need for lasting and effective corrective action were discussed with members of your staff at the inspection exit meeting on July 20, 2000. However, before the NRC makes its enforcement decision, we are providing you an opportunity to either (1) respond to the apparent violation addressed in this inspection report within 30 days of the date of this letter or (2) request a predecisional enforcement conference. If a conference is held, it will be closed to public observation and will be transcribed. Please contact Paul Fredrickson at 404/562-4530 within seven days of the date of this letter to notify the NRC of your intended response.

The enclosed summary of the OI report indicates that this situation occurred over an extended period. Although TVA had indications of personnel performance problems in this area, management oversight of this process failed to detect this situation earlier. As such, we request that you address any management oversight aspects of this issue as well as any associated corrective action, should you chose to respond to the apparent violation in writing or during a conference.

Your response, if you choose to provide one, should be clearly marked as a "Response to An Apparent Violation in Inspection Report No. 50-259/00-03, 50-260/00-03, 50-296/00-03" and should include: (1) the reason for the apparent violation, or, if contested, the basis for disputing the apparent violation, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Your response should be submitted under oath or affirmation and may reference or include previously docketed correspondence, if the correspondence adequately addresses the required response. If an adequate response is not received within the time specified or an extension of time has not been granted by the NRC, the NRC will proceed with its enforcement decision.

Please be advised that the characterization of the apparent violation described in the enclosed inspection report may change as a result of further NRC review. You will be advised by separate correspondence of the results of our deliberations on this matter.

In addition to the identified apparent violation, three issues of very low safety significance were identified. These issues were determined to involve violations of NRC requirements. However, the violations were not cited due to their low safety significance and because they have been entered into your corrective action program. If you contest these non-cited violations, 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 the Browns Ferry facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures, with Enclosure 2 redacted, and your response (if you choose to provide one), will be available electronically for public inspection in the NRC Public Document Room (PDR) or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). To the extent possible, your response should not include any personal privacy, proprietary, or safeguards information so that it can be placed in the PDR and PARS without redaction. ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Loren R. Plisco, Director
Division of Reactor Projects

Docket Nos. 50-259, 50-260, 50-296
License Nos. DPR-33, DPR-52, DPR-68

Enclosures: 1. NRC Inspection Report w/attachment
2. Summary of OI Report No. 2-1999-028

cc w/encls:
Karl W. Singer
Senior Vice President
Nuclear Operations
Tennessee Valley Authority
Electronic Mail Distribution

Jack A. Bailey, Vice President
Engineering and Technical Services
Tennessee Valley Authority
Electronic Mail Distribution

John T. Herron
Site Vice President
Browns Ferry Nuclear Plant
Tennessee Valley Authority
Electronic Mail Distribution

R. J. Adney, General Manager
Nuclear Assurance
Tennessee Valley Authority
Electronic Mail Distribution

General Counsel
Tennessee Valley Authority
Electronic Mail Distribution

Robert G. Jones, 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

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

Distribution w/encls:
W. Long, NRR
H. N. Berkow, NRR
B. Summers, OE
R. Borchardt, OE
PUBLIC

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NAME	RCarrion alt	WSmith	JStarefos	EDiPaolo	DJones	JBlake	JKreh
DATE	07/26/2000	07/21/2000	07/26/2000	07/26/2000	07/26/2000	07/26/2000	07/26/2000
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NAME	PFredrickson	ABoland	CEvans				
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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos:	50-259, 50-260, 50-296
License Nos:	DPR-33, DPR-52, DPR-68
Report No:	50-259/00-03, 50-260/00-03, 50-296/00-03
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Browns Ferry Nuclear Plant, Units 1, 2, & 3
Location:	Corner of Shaw and Nuclear Plant Roads Athens, AL 35611
Dates:	April 2 through June 24, 2000
Inspectors:	W. Smith, Senior Resident Inspector J. Starefos, Resident Inspector E. DiPaolo, Resident Inspector D. Jones, Senior Radiation Specialist J. Blake, Senior Project Manager J. Kreh, Emergency Preparedness Inspector
Approved by:	P. E. Fredrickson, Chief Reactor Projects Branch 6 Division of Reactor Projects

Enclosure 1

SUMMARY OF FINDINGS

Browns Ferry Plant, Units 1, 2 and 3
NRC Inspection Report 50-259/00-03, 50-260/00-03, 50-296/00-03

The report covers a twelve-week period of resident inspection. In addition, it includes the results of region-based inspectors associated with Unit 3 inservice inspection activities, emergency preparedness, and radiation safety.

The significance of an issue is indicated by its color (green, white, yellow, red) and was determined by the NRC's Significance Determination Process, as discussed in the attached summary of the NRC's Reactor Oversight Process.

Cornerstone: Initiating Events

- Green. A non-cited violation of Technical Specification (TS) Limiting Condition for Operation (LCO) 3.9.4 was identified for operators' failure to comply with the action statement requiring insertion and disarming of a control rod with a malfunctioning "full in" position indicating light during control rod testing on Unit 3.

  The finding had very low safety significance because administrative controls were in place to prevent more than one control rod from being withdrawn at any given time during the test (Section 1R14).

- Green. A non-cited violation of TS LCO 3.3.1.2 was identified for operators placing the Unit 3 reactor mode switch out of the shutdown position to perform reactor mode switch testing with less than the required number of operable source range monitors (SRMs).

  The finding had very low safety significance because the requirements of LCO 3.10.2 (i.e., no core alterations and all control rods inserted) were maintained at all times during reactor mode switch testing. In addition, the required TS surveillance required for SRM operability was subsequently completed satisfactorily on the A, B, and D SRMs (Section 1R20).

Cornerstone: Barrier Integrity

- Green. A non-cited violation of TS 5.4.1 was identified for an inadequate procedure utilized for the compensatory measures taken upon loss of both Unit 2 shutdown board room coolers which required actions that would cause a loss of function of the control room emergency ventilation (CREV) system and could degrade the radiation barrier designed to protect the control room operators during a design basis accident.

  The finding had very low safety significance because it represented a degradation of the radiological barrier function provided for the control room only. Modified compensatory actions to close certain dampers would result in the CREV system remaining operable (Section 1R15).

Other Activities

- Apparent Violation. An apparent violation of TS 5.4.1 was identified for apparent deliberate failure to implement measuring and test equipment (M&TE) control procedures which resulted in approximately 500 nonconformance evaluations either not being issued or completed for M&TE which had been identified as out-of-tolerance or otherwise meeting the criteria for evaluation (Section 1R19).

- A supplemental inspection was conducted in accordance with Inspection Procedure 95001, Inspection for One or Two White Inputs in a Strategic Area. The purpose of the inspection was to assess the licensee's evaluation associated with a Unit 3 White PI [Safety System Unavailability for the Heat Removal System, Reactor Core Isolation Cooling (RCIC)]. On December 30, 1998, during a manual start of the Unit 3 RCIC to perform a TS surveillance, there was no turbine speed indication in the control room, although there was indication of pump flow and pressure. A broken connector was found on the wiring to the turbine speed sensor. The licensee considered the sensor cable connector failure to have been an isolated, random failure, with possible damage due to personnel working in the area with the connector disconnected and hanging loose (the connector was not as vulnerable when assembled). The licensee stated that it could not determine the exact cause of the failure because the internal parts of the connector could have been broken for an extended period and the connector still could perform its function, as long as the pins were making contact. Although Unit 2 was not inspected at the time of the identification on Unit 3, subsequent disassembly and inspection on January 14, 2000, not related to the Unit 3 failure, did not identify any degradation of the connector on the Unit 2 RCIC. The licensee's corrective actions were appropriate for the circumstances (Section 40A5).

Report Details

Unit 1 has been shut down since March 19, 1985, and remained in a long-term lay-up condition with the reactor defueled.

Unit 2 operated at or near full power with the exception of scheduled brief reductions in power to adjust control rods and perform routine testing.

Unit 3 operated at or near full power with the exception of scheduled brief reductions in power to adjust control rods and perform routine testing until shortly before the scheduled refueling outage. On April 15, 2000, while the unit was coasting down prior to a refueling outage, the reactor scrammed in response to low reactor vessel water level when the 3C feedwater pump experienced a flow reduction due to a clogged turbine control oil filter (Section 4OA3.2). Due to this event's occurrence so close to the scheduled outage, the licensee commenced Refueling Outage (RFO) U3C9 approximately 21 hours early. The outage duration was 18 days, after which power was restored to 100%. On May 24, 2000, Unit 3 experienced an automatic scram caused by a pressure perturbation on the variable leg of the reactor vessel level instrumentation. The unit was restarted and restored to full power operation on May 26, 2000, where the unit remained through the end of this inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity and Emergency Preparedness

1R04 Equipment Alignment

a. Inspection Scope

The inspectors performed a partial walkdown of the below-listed systems to verify operability of the redundant train when one train was out of service.

- Unit 2 residual heat removal system (Operating)
- Emergency equipment cooling water (EECW) and residual heat removal service water (RHRSW) pump alignment during maintenance of A3 EECW pump
- Batteries 2 and 3 due to battery 1 being out of service for load testing

b. Issues and Findings

No findings were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors toured the below-listed plant areas to evaluate, as appropriate, conditions related to: (1) licensee control of transient combustibles and ignition sources; (2) the material condition and operational status of selected fire protection systems, equipment and features; and (3) the fire barriers used to prevent fire damage or fire propagation.

- Unit 2 reactor building, elevation 593
- Unit 2 reactor building, elevation 621
- Unit 3 reactor building, elevation 593
- Unit 3 reactor building, elevation 621
- Fire Area 8, 4 kilovolt (kV) Shutdown Board Room D
- Fire Area 13, Shutdown Board Room E
- Fire Area 18, Unit 2 Battery and Battery Board Room
- Fire Area 21, Unit 3 Diesel Generator Building
- Fire Area 22, 4kV Shutdown Board Room 3EA and 3EB
- Fire Area 23, 4kV Shutdown Board Room 3EC and 3ED
- Fire Area 24, 4kV Bus Tie Board Room

b. Issues and Findings

No findings were identified.

1R08 Inservice Inspection Unit 3

a. Inspection Scope

The inspectors evaluated inservice inspection (ISI) and repair and replacement activities during the ongoing Unit 3 refueling outage to determine the effectiveness of the licensee's American Society of Mechanical Engineers (ASME) Section XI ISI Program. The inspectors reviewed the videotapes of the in-vessel visual inspection (IVVI) of the core spray headers; jet pump beams, JP-11 through JP-20; and selected portions of the reactor pressure vessel (RPV) annulus. During the review of the RPV annulus tapes, photographs of the foreign materials (the nut and the Chicago fitting clip) found during the IVVI were examined. Records of the containment visual and ultrasonic thickness inspections were also reviewed.

The results of piping ultrasonic testing (UT) examinations conducted during this outage were inspected through a record review. The re-examination of Weld GR-3-63, was discussed in detail with the licensee's Level III UT examiner. This weld was found to have had an indication requiring Section XI resolution during the Cycle 8 refueling outage, thereby requiring augmented inspection. The Cycle 8 data was compared with the current Cycle 9 data.

In the area of repairs and replacements, the inspectors reviewed X-rays of two core spray, two main steam and two reactor core isolation cooling piping welds fabricated during this outage. The X-rays were reviewed to evaluate welder performance as well as the performance of the licensee personnel involved with the production and the evaluation of the X-rays.

b. Issues and Findings

No findings were identified.

1R11 Licensed Operator Requalification Program

a. Inspection Scope

On June 6 and 7, 2000, the inspector observed operator performance in the plant simulator and the subsequent evaluator's critique during licensed operator requalification training. In addition, the inspectors verified that the training program included high risk operator actions, emergency plan implementation, and lessons learned from previous plant experiences.

b. Issues and Findings

No findings were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

For the equipment issues described, the inspectors reviewed the licensee's implementation of the Maintenance Rule (10 CFR 50.65) to assess the effectiveness of the licensee's maintenance efforts that apply to scoped structures, systems and components (SSCs):

- Unit 2 and Unit 3 Recirculation System Transients
- Unit 3 Core Spray Pump B Failure on June 25, 1999
- Unit 3 High Pressure Coolant Injection (HPCI) loose wire connection in logic circuit on April 8, 1999
- Unit 3 HPCI oil leak on steam admission valve on April 14, 1999
- Unit 2 Control Rod Drive System Pump 2A Failure May 20, 2000
- Clearance error that resulted in a functional failure of EDG 1D on April 13, 2000

b. Issues and Findings

No findings were identified.

1R13 Maintenance Risk Assessment and Emergent Work Control

a. Inspection Scope

The inspectors evaluated the effectiveness of the licensee's risk assessments and the implementation of compensatory measures for several planned maintenance activities. In addition, the inspectors verified that, upon identification of two unforseen equipment problems, the licensee had taken the necessary steps to plan and control the resulting emergent work activities.

- Unit 3 500-kilovolt (kV) offsite power supply outage during refueling outage (planned)
- Unit 1 / 2 fire protection carbon dioxide storage tank piping repair (planned)
- Unit 2/3 containment atmospheric dilution tank B repair/modification (planned)
- Unit 2 calibration of low pressure ECCS permissive instrumentation (planned)

- Unit 2 combined intermediate valve (CIV) testing on May 11, 2000, CIV 4 would not close when the test demand signal was actuated (emergent)
- Unit 2 repair of shutdown board air cooling unit 2B on June 9, 2000 (emergent)

b. Issues and Findings

No findings were identified.

1R14 Personnel Performance During Nonroutine Plant Evolutions and Events

a. Inspection Scope

The inspectors reviewed personnel performance during planned and non-planned plant evolutions and selected licensee event reports focusing on those involving personnel response to non-routine conditions. The review was performed to ascertain whether operator response was in accordance with the required procedures.

b. Issues and Findings

On April 30, 2000, while Unit 3 was shut down for a refueling outage, operators performed control rod testing, which included a surveillance to verify control rod drive coupling integrity after refueling. When control rod 42-55 was withdrawn, the "full in" position indication light remained illuminated when it should have extinguished. In order to continue control rod testing, the operators were required by TS LCO 3.9.4 to immediately fully insert and disarm control rod 42-55. The operators performing the test failed to recognize this TS requirement and continued testing control rod 42-55. Following the test of control rod 42-55, the control rod was inserted but was not disarmed. The operators then continued testing other control rods.

The inspectors noted that when the Unit Supervisor became aware of the failed position indication light, approximately 2 hours later, he recognized that the requirements of TS LCO 3.9.4 were not fully met, and took action to achieve compliance. Approximately 4 hours after the position indication light had failed, control rod 42-55 was disarmed. After the Shift Manager appropriately debriefed the operators involved, control rod testing was resumed without further incident. The licensee entered this problem into the corrective action program under Problem Evaluation Report (PER) 00-004248-000 and took corrective actions to prevent a recurrence.

Although the administrative controls were in place which prevented more than one control rod from being withdrawn at any given time during the test, "full in" position indication was needed for the "all-rods-in" permissive to function as an interlock to prevent inadvertent withdrawal of more than one control rod. This problem had a credible impact on safety, and if left uncorrected, would have become a more significant safety concern. The safety significance of this operator failure was very low and; therefore the finding was determined to be Green. Administrative controls were in place to prevent more than one control rod from being withdrawn at any given time during the test.

Failure of the operators to immediately initiate action to insert and disarm control rod 42-55 upon recognizing the failed position indication light is a violation of TS LCO 3.9.4. This violation is being treated as a non-cited violation, consistent with Section VI.A.1 of the Enforcement Policy, issued on May 1, 2000 (65 FR 25368), and is identified as NCV 50-296/00-03-01: Failure to Meet TS LCO 3.9.4. This violation is in the licensee's corrective action program as PER 00-004248-000.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following operability evaluations affecting mitigating systems or barrier integrity to ensure that operability was properly justified and the component or system remained available such that no unrecognized increase in risk occurred.

- Technical Operability Evaluation (TOE) 2-00-073-146 Revision 0; 2-FCV 73-02 (HPCI Inboard Steam Isolation Valve) Sheared Torque Switch Roll Pin
- TOE 3-00-073-2107 Revision 0; 3-FCV-073-02 (HPCI Inboard Steam Isolation Valve) Valve Closing Seal-in Circuit Malfunctioning
- Calculation CD-Q2073-000023, Revision 0, Evaluation of HPCI System Over Pressurization Event (Pipe Analysis)
- Calculation MD-Q2073-000004, Revision 0, Evaluation of HPCI Pump Discharge Components Subjected to Overpressure Conditions
- PER 00-005465-000, Engineering Support of Operability Determination for Secondary Containment When Three SCIVs Were Discovered to Be Inoperable
- Engineering support of operability when both Unit 2 shutdown board room coolers failed

b. Issues and Findings

A non-cited violation of TS 5.4.1.a was identified for an inadequate procedure utilized for the compensatory measures taken upon loss of both Unit 2 shutdown board room coolers which required actions that would cause a loss of function of the control room emergency ventilation (CREV) system.

On May 27, 2000, both of the chiller units (100% capacity each) that are relied upon to cool shutdown board rooms 2A and 2B for Unit 2 failed. There is no TS LCO applicable to the air cooling units. However, the operability of the equipment in these rooms (e.g., 4-kV and 480-V shutdown boards) was dependent upon maintaining the board rooms below a pre-determined temperature. This ensured that the switchgear in the board rooms will be able to supply power to the Unit 2 engineered safety features during a design basis accident. Exceeding the pre-determined temperature in shutdown board rooms 2A and 2B has significant safety implications. The licensee had addressed this concern in Operating Instruction (OI) 0-OI-31, Control Bay and Off-Gas Treatment Building Air Conditioning System, which required operators to open the doors to shutdown board rooms in order to provide necessary cooling. This could result in the doors to shutdown board room 2A and 2B being opened simultaneously, thereby breaching the CREV boundary through ductwork connecting the rooms. The inspectors were concerned that this proceduralized compensatory measure would cause the CREV system to be incapable of fulfilling its intended safety function during the design basis accident.

The inspector requested the licensee's justification for operability of the shutdown boards when both chiller units were out-of-service on May 27, 2000. The licensee responded that they were taking actions to monitor shutdown board room temperatures in accordance with OI 0-OI-31 and that there was sufficient time to repair one air cooling unit before the shutdown boards became overheated. The licensee could not produce a documented basis for the statement that there was sufficient time to repair the cooler, and the statement was based on ambient conditions rather than accident conditions. It was not determined if maintenance personnel could safely get to the cooling units and repair them during accident conditions. The inspectors considered the licensee's basis for considering the shutdown boards operable with no room coolers to be unacceptable because the compensatory measures placed in effect were controlled by a procedure that required actions that would have caused the CREV system to be inoperable. This would have placed both operating units in a condition prohibited by TS. Subsequent to the inspection period, the licensee conducted testing on June 24, 2000, and determined that when certain dampers are closed, the CREV system would remain operable. The licensee has indicated that they would close the dampers in the future and has initiated actions to revise the operating instructions.

This issue was determined to have a credible impact on safety and it involved a degradation of one radiation barrier that could have resulted in a currently indeterminate dose to the control room staff during certain accidents. The safety significance of proceduralized compensatory measures which would cause the CREV system to be incapable of fulfilling its intended safety function during the design basis accident was very low and; therefore, the finding was determined to be Green. The issue only represented a degradation of the radiological barrier function provided for the control room. Because the licensee implemented compensatory measures to prevent a loss of function of the shutdown boards (not-withstanding the loss of the CREV function), it was not necessary to evaluate the potential loss of the shutdown boards.

CREV system operating procedure 0-OI-31, Revision 73, was inadequate in that it caused the CREV system to be inoperable when the shutdown board room doors were opened. This procedure deficiency is a violation of TS 5.4.1, which requires procedures to be established and maintained that are recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Procedures for operating safety related systems are recommended in Appendix A. This violation is being treated as a non-cited violation, consistent with the Section VI.A.1 of the NRC Enforcement Policy, issued on May 1, 2000 (65 FR25368), and is identified as NCV 50-260/00-03-02: Inadequate Procedure Renders CREVS Inoperable. This violation is in the licensee's corrective action program as PERs 00-006662-000 and 00-000227-000.

1R16 Operator Workarounds

a. Inspection Scope

The inspectors reviewed the status of selected operator workarounds to determine if the functional capability of the system or operator reliability in responding to an initiating event was affected. This included evaluating the effect of the operator workaround on the operator's ability to implement abnormal or emergency operating procedures. The following operator workarounds were reviewed:

- Unit 3 priority 1 workaround - Reactor Feed Pump 3C minimum flow valve manually isolated
- Unit 2 priority 2 workaround - Operation with control rod 14-51 "full-in" indication locked-in independent of rod position
- Unit 0 Diesel Generator C Battery Exhaust Fan has bad motor bearing

The inspectors also reviewed the cumulative effects of operator workarounds on the ability of operators to respond in a correct and timely manner to plant transients and accidents. Where applicable, the cumulative effects on the reliability and availability of a system were reviewed.

b. Issues and Findings

No findings were identified.

1R19 Post-Maintenance Testing (PMT)

a. Inspection Scope

The inspectors reviewed the performance of the following activities to verify that the PMT was adequate to verify system operability and functional capability:

- Unit 2 HPCI system PMT following discharge piping overpressurization, pump no flow operation, and repair of the test return line, performed on April 16, 2000
- Unit 3 Core Spray Pump 3B mechanical seal assembly was replaced by WO 99-012986-000, post-maintenance testing was performed on April 25, 2000
- Unit 2 shutdown board room air cooling unit 2A PMT after replacing the refrigeration suction pressure controller under WO 00-006079-000, performed on June 15, 2000
- Unit 2 shutdown board room air cooling unit 2A PMT after replacing the compressor head reed valve, completed on June 21, 2000
- RHRSW pump discharge check valve (0-CKV-23-502) replacement PMT performed in accordance with WO 99-008459-000. This test was performed on June 16, 2000
- Unit 3 Low Pressure Coolant Injection System Motor Generator Set 3DN PMT following several maintenance activities performed on June 19, 2000.

b. Issues and Findings

(Closed) URI 50-260,296/00-01-01: M&TE Out-of-Tolerance Investigations Not Performed. The licensee had identified that procedurally required actions to issue and/or disposition nonconformance evaluations had not been taken for numerous pieces of measuring and test equipment (M&TE) when out-of-tolerance reports were received at the site following testing by TVA Central Laboratory Field Testing Services (CLFTS). The licensee identified the problem during a self-assessment of the M&TE program in June 1999. During the early part of the assessment, the site M&TE Program Administrator resigned. This issue was placed in the corrective action program (PER 99-007248-000) on June 24, 1999, before the extent of the issue was recognized. On August 19, 1999, the licensee completed a final comparison of the CLFTS M&TE out-of tolerance listing and the site M&TE database. There were approximately 500 nonconformance evaluations that were not properly issued and/or dispositioned.

The licensee determined that 103 of the nonconformance evaluations not properly issued and/or dispositioned involved surveillance testing, among other work documents. To assess the number of safety-related applications, the licensee evaluated a 10% sample of the equipment related to the nonconformance evaluations not properly issued and/or dispositioned, and determined that approximately 78% of the uses were associated with safety-related equipment. The licensee applied additional resources to the surveillance test dispositions, and completed the review of the surveillance testing line items in the 103 nonconformance evaluation packages on September 2, 1999, with no operability issues. In October 1999, the licensee subsequently completed reevaluation of the remaining nonconformance evaluations with none affecting component operability.

An investigation conducted by the NRC Office of Investigations (OI) after the URI was identified, was completed on June 15, 2000. The evidence indicated that the M&TE Program Administrator deliberately failed to initiate and/or disposition nonconformance evaluations on test equipment that was out of tolerance, as required by TVA site procedures. Based on the deliberate aspects of this issue, the finding was not evaluated by the NRC Significance Determination Process.

Technical Specification 5.4.1 requires written procedures be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Appendix A, Section 8, specifically addresses procedures for control of M&TE. It further states that procedures of a type appropriate to the circumstances should be provided to ensure that tools, gauges, instruments, controls, and other measuring and test devices are properly controlled, calibrated, and adjusted at specified periods to maintain accuracy.

Procedure SSP-6.7, Control of Measuring and Test Equipment, Revision 8A, Effective May 27, 1997 through June 1, 1998, Step 3.14.A states that nonconformance evaluations shall be issued for the following conditions: lost M&TE or standards, out-of-tolerance M&TE or plant standards, damaged or otherwise defective M&TE or plant standards, and disassembled M&TE or plant standards. Step 3.14.E states that all nonconformance evaluations should be completed within 30 calendar days of the site receipt of the initiating document. An extension of up to ten calendar days may be approved by the Plant Manager, or designee, using Appendix H.

Procedure SPP-6.4, Measuring and Test Equipment, Revision 0, Effective May 29, 1998, Step 3.15.1 states that nonconformance evaluations shall be issued to determine the validity and acceptability of previous work for the following conditions: lost M&TE or standards, out-of-tolerance M&TE or plant standards, damaged or otherwise defective

M&TE or plant standards, and disassembled M&TE or plant standards. Step 3.15.6 requires all nonconformance evaluations be completed within 30 calendar days of the site receipt of the initiating document.

During the period from June 2, 1997, to June 14, 1999, SSP-6.7 and SPP-6.4 were not implemented, in that approximately 500 nonconformance evaluations either were not issued or completed for measuring and test equipment which had been identified as out-of-tolerance or otherwise meeting the criteria for evaluation. An estimated 78 percent of the nonconformance evaluations involved safety-related equipment.

Based on the deliberate aspects of this issue, the failure to implement both SSP-6.7 and SPP-6.4, as required by TS 5.4.1, is identified as apparent violation (EEI) 50-260,296/00-03-04, Failure to Implement Measuring and Test Equipment Procedures. Based on the identification of the apparent violation, and the inspector's satisfactory review of the non-safety related issues identified in the unresolved item, URI 260,296/00-01-01 is closed.

1R20 Refueling and Outage

a. Inspection Scope

The inspectors reviewed the licensee's Outage Safety Plan and contingency plans for the Unit 3 RFO (U3C9) to confirm that the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth. During the refueling outage, the inspectors observed portions of the shutdown and cooldown processes and monitored licensee controls over the below-listed outage activities:

- Licensee configuration management, i.e., maintenance of defense-in-depth commensurate with the outage safety plan for key safety functions and compliance with the applicable TS when taking equipment out of service.
- Implementation of clearance activities and confirmation that tags were properly hung and equipment appropriately configured to safely support the work or testing.
- Installation and configuration of reactor coolant pressure, level, and temperature instruments to provide accurate indication and an accounting for instrument error.
- Controls over the status and configuration of electrical systems to ensure that TS and outage safety plan requirements were met, and controls over switchyard activities.
- Monitoring of decay heat removal processes.
- Controls to ensure that outage work was not impacting the ability of the operators to operate the spent fuel pool cooling system.
- Reactor water inventory controls including flow paths, configurations, and alternative means for inventory addition, and controls to prevent inventory loss.
- Controls over activities that could affect reactivity.
- Maintenance of secondary containment as required by TS.
- Refueling activities, including fuel handling and sipping to determine which fuel assemblies were leaking.

- Startup and ascension to full power operation, tracking of startup prerequisites, walkdown of the drywell (primary containment) to verify that debris had not been left which could block emergency core cooling system suction strainers, and reactor physics testing.
- Licensee identification and appropriate resolution of problems related to refueling outage activities.

b. Issues and Findings

A non-cited violation of TS LCO 3.3.1.2 was identified for operators placing the Unit 3 reactor mode switch out of the shutdown position to perform reactor mode switch testing with less than the required number of operable source range monitors (SRMs).

Following the Unit 3 reactor scram on April 15, 2000, the reactor mode switch was placed in the shutdown position and the plant entered Mode 3. During a review of licensee reactivity controls, the inspector noted that the A, B, and D SRMs were not declared operable until the evening of April 17 when all the required TS surveillances were completed (the C SRM had a bad detector and remained inoperable). The inspector noted that operators entered TS LCO 3.10.2, Reactor Mode Switch Interlock Testing, on the morning of April 16. The testing required that the mode switch be moved out of the shutdown position. This contradicted the Actions of TS LCO 3.3.1.2, Source Range Monitor Instrumentation, which required all insertable control rods to be fully inserted and the reactor mode switch to be placed in the shutdown position within 1 hour unless the required number of SRMs, in this case 2, were operable in Mode 3.

Operators believed that the SRMs were operable prior to commencing reactor mode switch testing on April 16 because all of the required surveillances specified by plant procedure were performed. However, TS Surveillance Requirement (SR) 3.3.1.2.4, which verifies that the SRM count rate is greater than or equal to 3.0 cps with a signal to noise ratio greater than or equal to 3:1, was not performed. This SR is applicable in Mode 3 and was required for SRM operability. The licensee determined the root cause of the failure to perform SR 3.3.1.2.4 to be inadequate plant procedures. Neither the reactor scram nor unit shutdown procedures directed operators to perform the required surveillance.

TS LCO 3.10.2 allowed moving the reactor mode switch position defined by TS for Mode 3 to include the run, startup/hot standby, and refuel position and operations considered not to be in Mode 1 or 2 for the purpose of performing the mode switch interlock testing. This was allowed provided that all control rods remain fully inserted in core cells containing one or more fuel assemblies and no core alterations were in progress. TS LCO 3.0.7 stated that the Special Operations LCOs in TS Section 3.10 allow TS requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Because the requirements of LCO 3.3.1.2.D were not changed by LCO 3.10.2, the Actions of TS LCO 3.3.1.2.D (mode switch in shutdown) remained in effect. Therefore, placing the reactor mode switch out of the shutdown position to perform reactor mode switch testing was contrary to TS requirements.

The safety significance of this failure to place the mode switch in the shutdown position was considered to be very low and; therefore the finding was determined to be Green. Although the actions of LCO 3.3.1.2 were not maintained, the requirements of LCO 3.10.2 (i.e., no core alterations and all control rods inserted) were maintained at all times during reactor mode switch testing. In addition, TS surveillance requirement (SR 3.3.1.2.4), which was required for SRM operability, was subsequently completed satisfactorily on the A, B, and D SRMs.

Placing the reactor mode switch out of the shutdown position on April 16 to perform reactor mode switch testing was a violation of TS LCO 3.3.1.2 which required the mode switch be placed in the shutdown position with one or more required SRMs inoperable with the plant in Mode 3. This violation is being treated as a non-cited violation, consistent with the Section VI.A.1 of the NRC Enforcement Policy, issued on May 1, 2000 (65 FR 25368), and is identified as NCV 50-296/00-03-03: Failure to Meet TS LCO 3.3.1.2. This violation is in the licensee's corrective action program as PER 00-003778-000.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors witnessed surveillance tests and/or reviewed test data of the selected risk-significant SSCs listed below, to assess whether the SSCs met TS, updated final safety analysis report (UFSAR), and licensee procedure requirements, and to determine if the testing effectively demonstrated that the SSCs were operationally ready and capable of performing their intended safety functions. For in-service testing of selected risk significant mitigating system pumps and valves listed below, the inspectors evaluated the effectiveness of the licensee's American Society of Mechanical Engineers (ASME) Section XI testing program to determine equipment availability and reliability. The inspectors evaluated selected portions of the following areas: (1) testing procedures, (2) acceptance criteria, (3) testing methods, (4) compliance with the licensee's in-service testing program, technical specifications (TSs), and code requirements, (5) range and accuracy of test instruments, and (6) required corrective actions. The following surveillance tests were inspected:

- Surveillance Procedure (SP) 2-SR-3.5.1.6 (RHR I), Quarterly RHR System Rated Flow Test Loop I, Revision 5, ASME Section XI double frequency in-service test on the Unit 2 residual heat removal pump A, performed on May 5, 2000
- SP 3-SI-4.7.lA.2.g-3/75c, Primary Containment Local Leak Rate Test PSC High Level Control: Penetration X-227A, Revision 8, performed April 20, 2000; and 0-TI-106, General Leak Rate Test Procedure as performed on 3-FCV-75-57, performed April 23, 2000
- SP 3-SR-3.6.1.3.10(A), Primary Containment Local Leak Rate Test Main Steam Line A: Penetration X-7A, Revision 3, performed April 20, 2000; and 3-SR-3.6.1.3.10(D), Primary Containment Local Leak Rate Test Main Steam Line D: Penetration X-7D, Revision 3, performed April 20, 2000
- SP 2/3-SR-3.4.6.1, Dose Equivalent Iodine 131 Concentration, Revisions 3/2, performed on June 8, 2000

- SP 2-SR-3.3.5.1.4(C) Core and Containment Cooling Systems Reactor Low Pressure Instrument Channel C Calibration 2-P-68-95, Revision 0, performed on June 15, 2000
- SP 3-SR-3.5.1.7, HPCI Main and Booster Pump Set Developed Head and Flow Rate Test at Rated Reactor Pressure, Revision 10, performed on May 4, 2000

b. Issues and Findings

No findings were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors conducted a review of the list of active temporary plant modifications provided by the licensee. The following temporary modification was selected because the system was determined to be a key system from a probabilistic risk assessment perspective. A walkdown of the temporary leak sealant box was performed. The 10CFR50.59 screening, and selected sections of the UFSAR and TSs were reviewed.

- Temporary Alteration Control Form (TACF) 2-00-003-074, temporary repair of leak on threaded connection of drain piping of 2-SEP-074-0016 on residual heat removal pump 2C, installed March 15, 2000

b. Issues and Findings

No findings were identified.

1EP2 Alert and Notification System Testing

a. Inspection Scope

The inspector reviewed the alert and notification system design and associated testing commitments, and evaluated the adequacy of the testing program. Reviews were also conducted of the siren testing results.

b. Observations and Findings

No findings were identified.

1EP3 Emergency Response Organization Augmentation

a. Inspection Scope

The inspector reviewed the design of the emergency response organization (ERO) augmentation system and the maintenance of the licensee's capability to staff emergency response facilities within stated timeliness goals. Records of ERO augmentation drills were reviewed. These were unannounced, off-hour drills involving actual travel to the plant by ERO personnel. Although not required by the Radiological Emergency Plan (REP), such drills had been conducted annually since at least 1994 as a "good practice." Follow-up activities for problems identified through augmentation testing were reviewed to determine whether appropriate corrective actions had been implemented.

b. Issues and Findings

No findings were identified.

1EP4 Emergency Action Level and Emergency Plan Changes

a. Inspection Scope

The inspector reviewed changes to the REP as contained in Revisions 47, 48, 53, and 55 to determine whether any of the changes decreased the effectiveness of the REP. Revisions 47 and 48 included changes to the generic portion of the REP; Revisions 47, 53, and 55 contained changes to REP Appendix A (site-specific for Browns Ferry). Minor changes to the emergency action levels were made in Revisions 47 and 55. The inspector reviewed the REP changes against the requirements of 10 CFR 50.54(q).

b. Issues and Findings

No findings were identified.

1EP5 Correction of Emergency Preparedness Weaknesses and Deficiencies

a. Inspection Scope

The inspector evaluated the efficacy of licensee programs that addressed weaknesses and deficiencies in emergency preparedness. Documents reviewed included exercise and drill critique reports, PERs, self-assessment reports, and Audit Report No. SSA9903, issued August 25, 1999. No emergency declarations had been made since the last NRC inspection of the emergency preparedness program (March 1999).

b. Issues and Findings

No findings were identified.

1EP6 Drill Evaluation

a. Inspection Scope

The inspectors reviewed the drill scenario narrative to identify the timing and location of classification, notification, and protective action requirement (PAR) development activities. The drill was performed on May 16, 2000, at 6:30 p.m., CDT. The drill was observed with a focus on the classification and notification activities; this drill did not include a PAR activity. The inspectors verified the adequacy of the classification and notification activities and attended the formal drill critique.

b. Issues and Findings

No findings were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS2 ALARA Planning and Controls

a. Inspection Scope

The inspector reviewed the plant collective exposure history and observed job site implementation of ALARA controls and radiation worker performance for work in high radiation areas during preparations for Unit 3 Cycle 9 RFO U3C9. The inspector discussed with licensee personnel and reviewed records associated with source-term reduction, radiological work planning records for the U2C9 RFO, exposure estimates and tracking for the U2C9 RFO, and exposures to declared pregnant workers during 1999 and year-to-date (YTD) 2000. The inspector also reviewed recent ALARA program self-assessments and problem identification and resolution during 1999 and 2000 YTD.

b. Issues and Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

Mitigating Systems, Initiating Events and Barrier Integrity Cornerstone

.1 Heat Removal System Unavailability PI

a. Inspection Scope

The Safety System Unavailability for the Heat Removal System, Reactor Core Isolation Cooling (RCIC) PI was verified for Unit 3. The inspector reviewed PI data from the first quarter 2000 back to the second quarter 1997, to determine its accuracy and completeness.

b. Issues and Findings

No findings were identified.

Barrier Integrity Cornerstone

.2 Reactor Coolant System Specific Activity PI

a. Inspection Scope

The Reactor Coolant System Activity PI was verified for Units 2 and 3. PI data from April 1999 to March 2000 was reviewed. The inspector reviewed reactor water chemistry logs for the month of December 1999, for Unit 2, and February 2000, for Unit 3. The inspector verified that the proper value for dose equivalent iodine-131 was reported for those months. In addition, the inspector observed the performance of the dose equivalent iodine-131 surveillance test on Units 2 and 3 (See Section 1R22).

b. Issues and Findings

No findings were identified.

Initiating Events Cornerstone

.3 Unplanned Scrams per 7000 Critical Hours PI

a. Inspection Scope

The Unplanned Scrams per 7000 Critical Hours PI was verified for Units 2 and 3. Performance indicator data from April 1999 to March 2000 was reviewed. The inspector reviewed monthly operating reports, reviewed operations logs for a sample of dates, and selected specific corrective action documents to clarify questions.

b. Issues and Findings

No findings were identified.

Emergency Preparedness Cornerstone

.4 ERO Drill/Exercise PI

a. Inspection Scope

The inspector assessed the accuracy of the ERO Drill and Exercise Performance (DEP) PI through a review of the licensee's documentation. This included a review of evaluated exercise scenarios and a sample of drill and training scenarios to verify opportunities for ERO personnel to classify events, prepare notifications, and to develop protective action notifications (PARs). The inspector also reviewed licensee critique records to assess failures to classify, notify, or develop PARs. Documentation resulting from drills and training was reviewed for accuracy.

b. Issues and Findings

No findings were identified.

.5 ERO Drill Participation PI

a. Inspection Scope

The inspector assessed the accuracy of the ERO Drill Participation PI through a review of source records for selected individuals. In addition, the inspector reviewed and discussed the licensee's methodology for calculating this PI.

b. Issues and Findings

No findings were identified.

.6 Alert and Notification System PI

a. Inspection Scope

The inspector assessed the accuracy of the ANS Reliability PI through a review of the licensee's records of monthly full-cycle tests, annual growl tests, and biweekly silent tests.

b. Issues and Findings

No findings were identified.

4OA3 Event Follow-up

.1 Unit 2 High Pressure Coolant Injection System Pipe Break During Testing

a. Inspection Scope

The inspectors observed plant conditions and reviewed the status of plant mitigating systems when the 20-inch test return line (non-safety related) to the Unit 2 condensate storage tank (CST) broke during surveillance testing of the Unit 2 HPCI system on April 12, 2000. During the performance of quarterly flow rate testing, the HPCI system was lined up to take suction from and discharge to the CST. System discharge pressure was controlled by throttling the test return valve on the pump discharge. Following initial startup of the Unit 2 HPCI system on April 12, 2000, the operator closed the test return valve in an attempt to attain proper system discharge pressure. Subsequent manipulations of the test return valve and system automatic responses produced system flow and pressure transients. These transients, in combination with a substandard non-safety related weld, resulted in the failure of a non-safety related portion of the test return line.

The inspectors reviewed the effect of the pipe break on the operation of plant equipment. Water which spilled from the test return line was contained and processed by the licensee's radiological waste facility. During the transient, pressure in the pump discharge piping and components reached 1851 psig, exceeding its design pressure of 1500 psig. In addition, the HPCI pump was subjected to zero flow conditions for brief periods during the event because the pump minimum flow valve was not designed to open automatically on low flow unless a system initiation signal was present. There were no effects on other plant systems.

During the time that repairs were made, the HPCI system was available to perform its safety function to inject to the reactor vessel, if called upon. The system was unavailable for providing a means of condensing reactor steam in the suppression chamber by circulating condensate in a closed loop to/from the CST. This mode of operation was specified by emergency operating instructions as an alternate method of reactor pressure control. However, this mode of operation was categorized as a non-risk significant function by the licensee's maintenance rule scoping document 0-TI-346, Maintenance Rule Performance Indicator Monitoring, Trending, and Reporting, Revision 13.

The inspectors reviewed the licensee's operability evaluations of the overpressurized HPCI injection piping and components (See Section 1R15). PMT associated with the discharge piping overpressurization, pump no flow operation, and repair of the test return line were also reviewed (See Section 1R19).

This event was reviewed with the assistance of an NRC Region II Senior Risk Analyst. Based on the fact that the HPCI system remained available to fulfill its safety function to inject to the reactor vessel and the system was returned to an operable status within the TS allowed outage time, this event was determined to be of very low safety significance. This event was documented in the licensee's corrective action program as PER 00-003572-000.

b. Issues and Findings

No findings were identified.

.2 Unit 3 Reactor Trip Due to Low Reactor Vessel Level

a. Inspection Scope

On April 15, 2000, while in coastdown prior to the Unit 3 Cycle 9 RFO, operators experienced speed/flow fluctuations on reactor feed pump (RFP) 3C. At the time, RFP 3B was secured following the completion of maintenance. Earlier that day, operators received indication that there was high differential pressure on the feed pump turbine control oil filters. Operators were in the process of reducing reactor power in preparation for placing RFP 3B in service when RFP 3C experienced a flow reduction. The flow reduction on RFP 3C was determined to be caused by a drop in control oil pressure due to a clogged control oil filter. The reactor scrammed properly on the resultant reactor vessel low level. The inspector responded to the control room and verified that the HPCI system, RCIC system, and all of the expected containment isolations operated properly. Operators initially received indications that 2 control rods did not insert in response to the reactor scram. Operators entered the proper emergency operating instructions and took appropriate actions until proper indications were received. The licensee later determined by reviewing integrated computer system data that the control rods actually inserted during the scram and that the problem was attributed to control rod indication problems. Based on the fact that all expected isolations and safety systems responded properly to the low reactor vessel water level, the event was determined to be of very low safety significance. This event is documented in the licensee's corrective action program as PER 00-003657-000.

b. Issues and Findings

No findings were identified.

.3 (Closed) LER 50-296/2000-003-000: Unplanned Diesel Generator Automatic Start. This LER described a minor issue and was closed.

.4 (Closed) LER 50-296/2000-001-00: Reactor Scram Due to Feedwater Pump Control Oil System Problem. This event is discussed in Section 4OA3. No new issues were revealed by the LER.

.5 (Closed) LER 50-296/2000-002-00: Failure to Meet the Requirements of Technical Specifications During Reactor Mode Switch Testing. This event is discussed in Section 1R20.1 and resulted in an NCV. No new issues were revealed by the LER.

.6 (Closed) LER 296/2000-004-00: Missed Control Rod Limiting Condition for Operation (LCO). This event is discussed in Section 1R14 and resulted in an NCV. No new issues were revealed by the LER.

4OA5 Other

.1 Supplemental Inspection of Performance Indicator

Cornerstone: Mitigating Systems

a. Inspection Scope

The inspector conducted a supplemental inspection in accordance with Inspection Procedure 95001, Inspection for One or Two White Inputs in a Strategic Area. The purpose of the inspection was to assess the licensee's evaluation associated with a Unit 3 White PI (Safety System Unavailability for the Heat Removal System, RCIC). The PI was determined to be White in the 4th quarter of calendar year 1998.

b. Issues and Findings

The inspector reviewed the licensee's records to determine who identified the performance issue (i.e., licensee, self revealing, or NRC), and under what conditions the issue was identified. On December 30, 1998, during a manual start of the Unit 3 RCIC to perform TS Surveillance Requirement 3.5.3.3, there was no turbine speed indication in the control room, although there was indication of pump flow and pressure. The system was shut down immediately and Work Order 98-015969-000 was initiated to troubleshoot and repair the system as required. A broken connector was found on the wiring to the turbine speed sensor. A work order was initiated, the connector was replaced and RCIC was retested on December 31, 1998, with no further problems. A PER was not initiated in that the licensee's engineers considered this to have been an isolated, random failure that simply needed a work order to be repaired. There were no prior opportunities for identification of the broken connector, because the broken part was not visible from the outside of the connector. However, on February 2, 1999, the licensee issued a PER (99-001767-000), not due to the broken connector, but for the purpose of attaching a maintenance rule cause determination evaluation (CDE). Subsequently, when the licensee developed the PIs, 905 fault exposure hours were entered for the fourth quarter 1998. This was based on ½ the time since RCIC was last successfully operated, in accordance with PI program guidelines. The inspector considered this to have been appropriate for the circumstances.

The inspector evaluated the plant specific risk consequences. The CDE from PER 99-001767-000 stated that if RCIC injection had been called upon by an event, the turbine would have tripped on overspeed if the connector for the speed sensor cable was broken at the time. With automatic control lost, manual RCIC injection would have been possible by resetting the overspeed trip, and controlling manually with the throttle valve. With high pressure coolant injection and automatic depressurization systems available, the safety consequences of this failure were not significant.

The inspector questioned the licensee as to possible causes of the failure. The licensee considered the sensor cable connector failure to have been an isolated, random failure, with possible damage due to personnel working in the area with the connector disconnected and hanging loose (the connector was not as vulnerable when assembled). The licensee stated that it was not possible to determine the exact cause of the failure because the internal parts of the connector could have been broken for an extended period and the connector would perform its function, as long as the pins were making contact. The inspectors determined this explanation to be reasonable.

The inspector examined the PER CDE to determine if it included a review to see if similar problems had previously been reported with the RCIC system. This was the first known instance of a failure of this type, and a search of Browns Ferry and Institute of Nuclear Plant Operations (INPO) databases only identified one other related incident at another nuclear power plant. In addition the inspector reviewed the impact on Unit 2 from this condition adverse to quality. Although Unit 2 was not inspected at the time of the identification on Unit 3, subsequent disassembly and inspection on January 14, 2000, not related to the Unit 3 failure, did not identify any degradation of the connector on the Unit 2 RCIC.

The inspector concluded that the licensee’s actions in response to the RCIC connector failure were consistent with regulatory requirements. Based on the absence of data to the contrary, the probability of a repeat occurrence was very low.

4OA6 Management Meetings

The inspectors presented the inspection results to Mr. J. Herron, Site Vice President, and other members of licensee management at the conclusion of the inspection on June 30 and July 20, 2000. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any of the materials examined during the inspection should be considered proprietary. No proprietary information was identified.

LIST OF PERSONS CONTACTED

Licensee
T. Abney, Licensing Manager
A. Bhatnagar, Site Support Manager
J. Chenkus, Emergency Preparedness Systems Manager (corporate)
R. Coleman, Radiological Control Manager
J. Corey, Radiation Protection and Chemistry Manager
T. Cornelius, Emergency Preparedness Manager

J. Grafton, Site Quality Assurance Manager
J. Herron, Site Vice President
R. Jones, Plant Manager
R. LeCroy, Site Security Manager
R. Rogers, Maintenance Superintendent
G. Little, Operations Manager
B. Marks, Supervisor, Emergency Preparedness Programs and Implementation (corporate)
R. Moll, System Engineering Manager
W. Nurnberger, Chemistry Superintendent
D. Olive, Operations Superintendent
D. Sanchez, Training Manager
J. Schlessel, Maintenance Manager
J. Shaw, Design Engineering Manager
R. Wiggall, Site Engineering Manager

NRC
R. Bernhard, Region II Senior Reactor Analyst
W. Long, Project Manager, NRR
J. Colaccino, NRR-EMEB

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-260,296/00-03-04	EEI	Failure to Implement Measuring and Test Equipment Procedures (1R19).

Opened and Closed During this Inspection

50-296/00-03-01	NCV	Failure to Meet TS LCO 3.9.4 (1R14).
50-260/00-03-02	NCV	Inadequate Procedure Renders CREVS Inoperable (1R15).
50-296/00-03-03	NCV	Failure to Meet TS LCO 3.3.1.2 (1R20.1).

Previous Items Closed

50-296/2000-003-000	LER	Unplanned Diesel Generator Automatic Start (4OA3.4).
50-296/2000-001-000	LER	Reactor Scram Due to Feedwater Pump Control Oil System Problem (4OA3.5).
50-296/2000-002-000	LER	Failure to Meet the Requirements of Technical Specifications During Reactor Mode Switch Testing (4OA3.6).

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety	Radiation Safety	Safeguards
• Initiating Events	• Occupational	• Physical Protection
• Mitigating Systems	• Public	
• Barrier Integrity		
• Emergency Preparedness		

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.

SUMMARY OF OFFICE OF INVESTIGATIONS REPORT 2-1999-028

The Nuclear Regulatory Commission's (NRC) Office of Investigations (OI) Report No. 2-1999-028 involved an investigation to determine whether a Tennessee Valley Authority (TVA) employee deliberately failed to perform nonconformance evaluations as required by site procedures while employed as the Measuring and Test Equipment (M&TE) Program Administrator at the Browns Ferry Nuclear Plant (BFN).

Certain M&TE used at BFN are calibrated on a regular basis by TVA's Central Laboratory Field Testing Services (CLFTS). When CLFTS identifies an instrument that is out of tolerance, that information is forwarded to the BFN Maintenance Department, M&TE Group. The M&TE Program Administrator is responsible for issuing and ensuring disposition of each nonconformance evaluation. BFN Site Standard Practice Procedure (SSP)-6.7, Control of Measuring and Test Equipment, Revision 8A, effective May 27, 1997 through June 1, 1998, and TVA Standard Programs and Processes Procedure (SPP)-6.4, Measuring and Test Equipment, Revision 0, effective May 29, 1998, through August 15, 1999, require nonconformance evaluations to be issued and dispositioned for conditions such as lost M&TE or standards, out-of-tolerance M&TE or plant standards, damaged or otherwise defective M&TE or plant standards, and disassembled M&TE or plant standards.

In June 1999, a BFN self-assessment of the M&TE program revealed that several out-of-tolerance M&TE items did not have nonconformance evaluations initiated by BFN. BFN initiated Problem Evaluation Report (PER) 99-007248-000 on June 24, 1999, to review the extent of condition and develop corrective actions. TVA determined that, over a period from June 1997 to June 1999, from the population of M&TE identified by CLFTS to be out-of-tolerance, approximately 500 nonconformance evaluations were not properly issued and/or dispositioned for components tested or inspected using the out-of-tolerance M&TE.

The M&TE Program Administrator stated that he responded to every out-of-tolerance evaluation he received from CLFTS. However, he could not explain why the large number of nonconformance evaluations had not been issued and/or dispositioned. The day after the licensee identified the issue and questioned the M&TE Program Administrator, the M&TE Program Administrator resigned from TVA.

The following paragraph contains personal privacy information and is withheld from public disclosure in accordance with 10 CFR 2.790.

[Withheld per 10 CFR 2.790]

The evidence indicated that the M&TE Program Administrator deliberately failed to initiate and/or disposition nonconformance evaluations on test equipment that was out-of-tolerance, as required by TVA site procedures.