UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
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ATLANTA, GEORGIA 30303-8931

October 23, 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 50-259/00-04, 50-260/00-04, 50-296/00-04

Dear Mr. Scalice:

On September 23, 2000, the NRC completed an inspection at your Browns Ferry 1, 2, & 3 reactor facilities. The enclosed report presents the results of that inspection which were discussed on September 28, 2000, with Mr. J. Herron and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, the inspectors identified two issues of very low safety significance (green) were identified. One of these issues was determined to involve a violation of NRC requirements. Because of its very low safety significance and because it has been entered into your corrective action program, the NRC is treating this issue as a non-cited violation, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny this non-cited violation, you should provide a response, with the basis for your denial, within 30 days of the date of this inspection report, 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 enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's Document system

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Sincerely,

/RA/

Paul E. Fredrickson, Chief
Reactor Projects Branch 6
Division of Reactor Projects

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

Enclosure: NRC Inspection Report w/attachment

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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:	50-259/00-04, 50-260/00-04, 50-296/00-04
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:	June 25 - September 23, 2000
Inspectors:	W. Smith, Senior Resident Inspector J. Starefos, Resident Inspector E. DiPaolo, Resident Inspector P. Taylor, Sr. Project Engineer, Division of Reactor Projects T. Easlick, Senior Resident Inspector, Brunswick Nuclear Plant J. Blake, Sr. Project Manager, Division of Reactor Safety D. Jones, Sr. Radiation Specialist, Division of Reactor Safety
Approved by:	P. E. Fredrickson, Chief Reactor Projects Branch 6 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000259-00-04, IR 05000260-00-04, IR 05000296-00-04, on 06/25-09/23/2000, Tennessee Valley Authority, Browns Ferry Plant, Units 1, 2 and 3. non-routine evolutions/events, other activities.

The significance of issues is indicated by their color (green, white, yellow, red) and was determined by the Significance Determination Process (SDP) as found in NRC Inspection Manual Chapter 0609 and 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) 3.0.4 was identified for operators changing Unit 2 modes, from Mode 3 (Hot Shutdown) to Mode 2 (Startup). The mode change was made without all the required TS channel check surveillances being met within their specified frequency for instrumentation required to be operable in Mode 2.

  The risk was determined to be of very low safety significance because the required channel checks were promptly performed after the identification and no loss of function occurred (Section 1R14).

Other Activities:

- Green. A Unit 3 automatic reactor scram, that was caused by a pressure perturbation on the variable leg of the reactor vessel level instrumentation, revealed an inadequate procedure, that did not contain sufficient detail to assure that a level instrument was returned to service without perturbing the reactor instrument sensing lines.

  The risk was determined to be of very low safety significance because all mitigation systems remained operable and barrier integrity was not challenged (Section 4OA3.1).

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, and a power reduction to approximately 66 percent for 2 days, on August 25, 2000, to repair the seal on the 2A condensate pump and to conduct scram time 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, and a power reduction to approximately 57 percent and single loop operation, on September 5, 2000, for approximately 2 days to repair the 3A reactor recirculation pump motor-generator. The motor-generator failed to operate properly because of loose and broken fasteners on the internal speed control linkage.

1. REACTOR SAFETY

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

1R02 Evaluations of Changes, Tests or Experiments

a. Inspection Scope

The inspectors reviewed selected samples of design change evaluations to verify that the licensee had appropriately considered the conditions under which the licensee may make changes to the facility or procedures without prior NRC approval. The inspectors verified, through review of additional information, such as calculations and supporting analyses, that the licensee had appropriately concluded that the change, test or experiment could be accomplished without obtaining a license amendment. The design changes reviewed were as follows:

Design Change Notices (DCNs) with Safety Evaluations:

50111	Implement mitigation monitoring system associated with noble metal injection
T40978	Replace radwaste primary containment isolation valves for floor and equipment drain sump discharge lines
50083	Replace sudden pressure relay on 500-kilovolt (Kv) Main Transformer 3A, 3B, 3C and Unit Station Service Transformer 3A and 3B, add additional relay in series for 2 out of 2 logic
50426	Replace Diesel Generator (DG) A, B, and C battery exhaust fans with similar units
50097	Addition of 3" block valve with 3/4" test connection to facilitate Appendix J testing of Core Spray System Valve 2/3-FCV-75-57 (primary containment isolation system valve) pressure suppression head tank (keep-fill) suction line
50441	Modify hydrogen injection system to support implementation of noble metals (lower hydrogen and oxygen injection rates)
50180	Install air filter on Unit 1, Unit 2, and Unit 3 offgas dilution fans

T41301	Modify disc of 2-FCV-71-39 by drilling to prevent pressure locking
50340	Replace Rosemont internals to improve process noise
T40220	Residual heat removal service water pump impeller replacement
W40283	Control bay chiller modifications

The inspectors also reviewed the following samples of design and procedure changes for which the licensee had determined that safety evaluations were “screened out” (not required), and verified that the licensee’s conclusions were correct and consistent with 10 CFR 50.59:

DCNs with Safety Evaluations Screened Out:

50252	Change in calibration frequency to 24 months (Radiation Monitoring)
50036	Increase setpoint for control rod drive hydraulic control unit accumulator control room alarm for low nitrogen pressure
50195	Add redundant contactor for reactor zone and refueling zone exhaust fan circuits to assure trip for isolation signal
50335	Replace existing check valves, reactor building floor sump pump discharge check valves, with soft seat valves
50158	Reactor water level III setpoint change
50451	Replace reactor water cleanup regenerative heat exchanger vent line, leaking 3/4" line, replace with stainless steel schedule 80 pipe
50316	Replace Rosemont transmitter and sensing line transducers for main turbine electrohydraulic (EHC) pressure transmitter 3-PT-047-0170 and 2-PT-47-0170 with Omega pressure transmitters
50099	Removal of snubber on standby liquid control system
50187	Revise splice type for non-Raychem, on low pressure coolant injection motor-generator
T41160	High pressure coolant injection (HPCI) pipe support - excessive gaps

Procedure Changes with Safety Evaluations Screened Out:

2-GOI-100-1A, Revision 87, Unit Startup from Cold Shutdown to Power Operation
0-SR-3.8.1.7(D), Revision 6, Diesel Generator D 24-Hour Run
2-OI-68, Revision 79, Reactor Recirculation System
RCI-17, Revision 38, High Radiation Area Door Control
2-SR-3.3.3.2.3(8), Revision 7, Reactor Core Isolation Cooling Backup Control Panel Turbine Speed Indicator Calibration
½-SIMI-86B, Revision 13, DG Air Start Procedure
2-SR-3.3.1.1.16(APRM-2), Revision 7, APRM Functional Test
2-AOI-78-1, Revision 15, Fuel Pool Cleanup

The inspectors also verified that problems identified with 10 CFR 50.59 evaluations had been entered in the licensee’s corrective action program. For the following selected sample of problems associated with 10 CFR 50.59 evaluations, the inspectors verified that the licensee had appropriately resolved the technical concerns and regulatory requirements.

Reviewed Problem Evaluation Reports (PERs):

99-001611-000	Temporary Alteration Control Forms disabled SVLL Logic of EHC did not reference Final Safety Analysis Report section
99-002311-000	HPCI Test return valve de-energized, defeated auto closure function, no safety evaluation
99-003762-000	Calculation Revision 7 prior to Safety Assessment (Screening Review)/Safety Evaluation (SA/SE) revision
99-004116-000	SA/SE incorrect/incomplete entries
99-004117-000	Administrative errors in SA/SE
99-004118-000	SA failed to evaluate Updated Final Safety Analysis Report (UFSAR), Sections 10.5.5 and 10.5.6
99-004198-000	Nuclear Assurance assessment SA with “No” contradicted statement
99-004200-000	List of qualified 50.59 individuals did not contain 3 GP employees
99-013192-000	Several inappropriate SA or no SA
00-003464-000	Door Seal Belzona Repair
00-000816-000	Drywell Equipment Drain Leakage System, SA did not address cooling
00-004764-000	Nuclear Safety Review Board review of “50.59s” had minor comments

b. Findings

No findings of significance were identified.

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 HPCI system alignment during the reactor core isolation cooling (RCIC) outage of July 19-20, 2000
- Unit 3 RHR system loop II alignment during the RHR system loop I outage of August 16, 2000
- Unit 3 RHR system loop II alignment during inspection and post-maintenance test of RHR heat exchanger 3A on September 15, 2000

One complete risk-important system walkdown was performed during this quarter. The Unit 2 HPCI system was selected as the risk-important mitigating system. Portions of Operating Instruction 2-OI-73, High Pressure Coolant Injection System, Revision 59, and Procedure 2-OI-73, Revision 043, Attachments 1, 2, and 4, were reviewed. The inspector verified the position of all main control room components listed on Attachment 2. A sampling of components and instruments were inspected using Procedure 2-OI-73, Attachments 1 and 4. The inspector reviewed outstanding design issues through review of the plant equipment action list, the operator workaround list, and the temporary alteration control form list which is further described in Section 1R23. The inspector also reviewed the outstanding maintenance work requests for the system.

b. Findings

No findings of significance 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. The inspectors referred to the licensee's Fire Protection Report, Volume 1 (Revision 15) and Volume 2 (Revision 27) while preparing for the inspections.

- Fire Zone 2-6, Unit 2 reactor building, elevation 639, south of column line R
- Fire Area 4, 4 KV shutdown board room B, Unit1 reactor building, elevation 593
- Fire Area 12, shutdown board room F, Unit 3 reactor building, elevation 593
- Fire Area 5, 4 KV shutdown board room A, Unit 1 reactor building, elevation 621
- Fire Zone 2-1, Unit 2 reactor building, elevation 519-565, west of column line R11
- Fire Zone 2-3, Unit 2 reactor building, elevation 593, north of column line R

On September 6, 2000, the inspectors conducted the annual observation of a fire brigade drill. The readiness of licensee personnel to fight and prevent the spread of fires was evaluated in terms of proper utilization of equipment needed to combat the fire, utilization of pre-plan strategies, communications, and meeting drill objectives. The inspectors attended the post-drill critique to confirm a satisfactory level of self-critical discussion.

b. Findings

No findings of significance were identified.

1R07 Annual Review of Heat Sink Performance

a. Inspection Scope

On August 30-31, 2000, the inspectors observed portions of the licensee's inspection of Unit 3 RHR heat exchanger 3B, to verify the following:

- Any potential heat exchanger deficiencies which could mask degraded performance were identified
- Inspection results were appropriately categorized against pre-established engineered acceptance criteria and were acceptable
- Ensure that the frequency of inspection was sufficient, given the site-specific potential for fouling

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program

a. Inspection Scope

On September 19, 2000, the inspector observed operator performance in the plant simulator and the subsequent evaluator's critique during licensed operator requalification training. The inspection focused on high-risk operator actions, emergency plant implementation and lessons learned from previous plant experiences. In addition, the inspector confirmed that the simulator board configurations reflected recently implemented plant temporary modifications.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

For the equipment issues described below, 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 3 control rod position indication failures following the reactor scram occurring on April 15, 2000
- Unit 3 C reactor feed pump functional failure causing reactor scram on April 15, 2000
- Unit 2 high pressure coolant injection system movement from 10 CFR 50.65 (a)(1) to (a)(2) status
- Unit 2 high pressure coolant injection system test return condensate pipe break event on April 12, 2000
- High pressure diesel fire pump functional failures due to engine coolant system problems occurring on July 7, 1999, and September 29, 1999
- Unit 2 condensate system failures resulting in unplanned capability loss events (UCLEs) exceeding the performance criterion (Expert Panel meeting minutes dated October 7, 1999)

b. Findings

No findings of significance 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 the following maintenance activities:

- Unit 2 core spray loop I outage of July 13-14, 2000 (planned)
- Unit 2 standby liquid control train B outage of August 1-2, 2000 (planned)
- Unit 2 inboard MSIV line A DC solenoid valve, 2-FSV-1-14B, failure (emergent)
- Unit 2 Temporary Alteration to provide backup power to the AC solenoid valve on 2-FCV-1-14, inboard line A MSIV, on August 26, 2000 (planned)
- Unit 3 RHR Loop II outage of August 29-31, 2000 (planned)
- Unit 3 failure of the 3A reactor recirculating pump motor-generator of September 5, 2000 (emergent)

The inspectors also verified that, upon identification of the emergent (unforseen) equipment problems, the licensee had taken the necessary steps to plan and control the resulting emergent work activities. In addition, for the Unit 2 inboard main steam isolation valve (MSIV) line A solenoid valve failure, the inspector verified that the licensee considered plant risk in rescheduling plant work and tests, planning interim actions to minimize plant risk, and had established additional controls to minimize potential personnel errors from causing an initiating event until a temporary alteration could be implemented to restore solenoid valve redundancy.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Nonroutine Plant Evolutions and Events

a. Inspection Scope

The inspectors observed and/or reviewed personnel performance during the following planned and unplanned non-routine plant evolutions: The review was performed to ascertain whether operator response was in accordance with the Technical Specifications (TS).

- Portions of Unit 2 reactor shutdown commenced on June 29, 2000
- Portions of Unit 2 reactor startup commenced on July 1, 2000
- Failure of Unit 3 control rod 50-51 to stop insertion upon demand on August 28, 2000, during control rod exercising

b. Findings

A non-cited violation of TS Surveillance Requirement (SR) 3.0.4 was identified for operators changing Unit 2 modes, from Mode 3 (Hot Shutdown) to Mode 2 (Startup), on July 1, 2000. The mode change was made without all the required TS channel check surveillances being met within their specified frequency for instrumentation required to be operable in Mode 2.

On July 1, 2000, operators placed Unit 2 in Mode 2 (Startup) following a maintenance outage which commenced on June 29. The inspector found that the required channel check surveillance requirements on the intermediate range monitors (neutron flux) and the average power range monitors (2-out-of-4 voter), required by TS SR 3.3.1.1, had not been performed prior to entering Mode 2. These instruments were required to be operable in Mode 2 in accordance with TS LCO 3.3.1.1, Reactor Protection System Instrumentation. The surveillances were not performed in the previous plant mode because the instruments were not required to be operable while the plant was in Mode 3. The inspector found that the last performance of these surveillances was outside the TS specified frequency. The inspector noted that TS SR 3.0.4 requires that entry into a mode shall not be made unless the LCO's surveillances have been met within their specified frequency. The inspector questioned the licensee why the channel checks were not performed prior to entering Mode 2.

The licensee concluded that the channel checks on the instruments should have been performed prior to entering Mode 2 because they had exceeded the TS allowed frequencies. The licensee found that required channel checks on several other instruments required to be operable in Mode 2 also had not been performed and were outside their TS required frequencies. The licensee promptly performed the required checks satisfactorily prior to proceeding with the reactor startup.

The licensee determined that the cause of the event was due to a failure to adequately implement SR 3.0.4 requirements in training and procedures. Corrective actions included revising plant procedures to provide checks and verifications that all applicable SRs are completed prior to a mode change. In addition, the licensee planned to perform operator training on the requirements of SR 3.0.4.

Not performing the required channel checks had a credible impact on safety. In addition, the failure to perform the TS required surveillances could affect the operability of a mitigating system. However, because the required checks were subsequently performed satisfactorily and no loss of function occurred, this finding is considered to be of very low safety significance (Green).

Changing Unit 2 modes, from Mode 3 (Hot Shutdown) to Mode 2 (Startup), on July 1, 2000, without all the required TS channel check surveillances being met within their specified frequency for instrumentation required to be operable in Mode 2, was a violation of TS SR 3.0.4, which required that entry into a mode shall not be made unless the LCO's surveillances have been met within their specified frequency. This violation is being treated as a non-cited violation, consistent with the Section VI.A.1 of the NRC Enforcement Policy, and is identified as NCV 50-260/00-04-01, Failure to Meet TS SR 3.0.4 for Instrument Channel Checks. In addition, the licensee submitted Licensee Event Report (LER) 50-260/2000-001. This violation is in the licensee's corrective action program as PER 00-006762-000.

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) 0-00-082-9000, Revision 01, 3C diesel generator exhaust piping to exhaust muffler weld failure and resultant leak, January 11, 2000
- TOE 0-00-031-9004, Revision 00, control room emergency ventilation (CREV) A post filter differential pressure indicating switch failed calibration at high point, May 6, 2000
- PER 00-007409-000, Unit 1&2 diesel generator mounting bolt washers have medium duty instead of heavy duty washers installed, July 20, 2000
- TOE 0-00-073-7535, Revision 0, calculated loads on Unit 2 and Unit 3 high pressure coolant injection system turbine inlet nozzles exceed the allowable load limit specified in Updated Final Safety Analysis Report, July 26, 2000
- PER 00-007901-000, Unit 2 D core spray pump upper motor bearing high temperature indications, August 5, 2000
- Associated plant equipment operability (main control room air conditioning, control room emergency ventilation, and shutdown board rooms) during the Unit 2 A and B shutdown board room chiller outage on September 19-20, 2000

b. Findings

No findings of significance were identified.

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:

- Priority 2 operator workaround - emergency equipment cooling water north header low pressure alarm not functioning with C3 pump in service
- Priority 1 operator workaround - Unit 2, 2C reactor feed pump miniflow valve failed to fully shut, and is therefore isolated

b. Findings

No findings of significance 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 G intermediate range monitor detector replacement PMT per work order (WO) 99-005645-000, performed on July 1, 2000
- Unit 2 refueling zone air supply outboard secondary isolation valve PMT following maintenance per WO 00-000678-000, performed on July 18, 2000
- Unit 3 core spray loop II motor operated valve PMTs per WO 00-000353-000, WO 00-000354-000, and WO 00-000974-000, performed on August 4, 2000
- Unit 2 inboard line A MSIV temporary alteration PMT per WO 00-006886-003/AR, performed on August 26, 2000
- Unit 2 DN low pressure coolant injection motor-generator PMT per Procedure EPI-0-268-MEZ003, Maintenance for LPCI Motor-Generator Sets, Revision 17, performed on September 8, 2000
- Unit 3 3A RHR heat exchanger RHRSW side pressure test PMT per WO 99-007602-000, performed September 15, 2000

b. Findings

No findings of significance were identified.

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, 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, and code requirements, (5) range and accuracy of test instruments, and (6) required corrective actions:

- Surveillance Procedure (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 July 7, 2000
- SP 0-SR-3.8.1.1(B), Diesel Generator B Monthly Operability Test, Revision 12, performed August 8, 2000
- SP 0-SR-3.3.8.1.1(C), 4 KV Shutdown Board C Degraded Voltage Relay Calibration and Functional Test, Revision 0, performed August 22, 2000
- SP 2-SR-3.1.4.1, Scram Insertion Times, Revision 9, performed August 26, 2000
- SP 2-SR-3.3.5.1.3 (ADS B/CS), Core Spray System Pump Discharge Pressure ADS Permissive Calibration 2-PS-75-35 and 2-PS-75-44, Revision 4, performed August 31, 2000
- SP 2-SR-3.5.1.6(RHR I), Quarterly RHR System Rated Flow Test Loop I, Revision 7, performed September 8, 2000

b. Findings

No findings of significance 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 modifications were selected because the system was determined to be a key system from a probabilistic safety assessment perspective. The 10 CFR 50.59 screening, and selected sections of the UFSAR and TSs were reviewed:

- Temporary Alteration Control Form (TACF) 2-00-009-073, Revision 0, temporary packing leak repair on HPCI inlet steam line drain pot level switch isolation valve (2-RTV-073-201A)
- TACFs 2-00-012-074 and 3-00-008-074, Revisions 0, remove power from residual heat removal minimum flow valves to support Appendix R

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation

a. Inspection Scope

The inspectors observed an emergency preparedness training evolution performed on August 2, 2000. 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 observed with a focus on the classification and notification activities by control room personnel and did not include a PAR activity. The inspectors verified the adequacy of the classification and notification activities. The results of the licensee's drill critique were also reviewed.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY
Cornerstone: Occupational Radiation Safety

2OS2 ALARA Planning and Controls

a. Inspection Scope

The inspectors reviewed the plant collective exposure history and the exposures incurred during the recently completed Unit 3 Cycle 9 (U3C9) refueling outage (RFO) to assess the licensee's performance in maintaining radiation exposures ALARA. The inspectors utilized the U3C9 ALARA Planning Report (APR) Summary to select the five APRs for the work activities which incurred the most dose during the U3C9 RFO. The inspectors verified that the ALARA controls established for those selected APRs (Nos. 00-0030, 00-0037, 00-0046, 00-0057, and 00-0063) were integrated into selected radiation work permits correlated with those work activities. Implementation of ALARA controls and radiation worker performance for work in radiation areas were observed during the inspection. Exposure tracking and records of exposures to declared pregnant workers during calendar year 2000 were also reviewed. Plant source term monitoring records were reviewed to assess the licensee's source-term reduction program. The inspectors reviewed a plot of the averages for the contact dose rates at the suction and discharge sides of the recirculation pumps which depicted the effects of chemical decontamination, depleted zinc oxide injection, and hydrogen water chemistry. The differences in the magnitude of the dose rates between Unit 2 and Unit 3 were reviewed with respect to the licensee's Source Term Reduction Plan for the chemical decontamination of Unit 2 recirculation piping during the next Unit 2 RFO scheduled for the spring of 2001. The effectiveness of problem identification and resolution for selected ALARA related issues identified during calendar year 2000 (YTD) was also evaluated by the inspectors.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

Licensee records were reviewed to determine whether the submitted PI statistics were calculated in accordance with the guidance contained in NEI 99-02, Regulatory Assessment Performance Indicator Guideline, Revision 0.

Initiating Events Cornerstone

.1 Unplanned Scrams per 7000 Critical Hours

a. Inspection Scope

The inspectors performed a review of the Units 2 and 3 PI data pertaining to unplanned scrams per 7000 critical hours for the second quarter of 2000 to determine its accuracy and completeness. Documentation reviewed included the control room operator logs, licensee review and verification reports, licensee event reports, and the PI data provided at the NRC web site.

b. Findings

No findings of significance were identified.

Mitigating Systems Cornerstone

.2 Safety System Functional Failures

a. Inspection Scope

The inspectors performed a review of the Units 2 and 3 PI data pertaining to safety system functional failures for the prior four quarters to determine its accuracy and completeness. Documentation reviewed included licensee event reports and the PI data provided at the NRC web site.

b. Findings

No findings of significance were identified.

Barrier Integrity Cornerstone

.3 Reactor Coolant System Leakage

a. Inspection Scope

The inspectors performed a review of the Units 2 and 3 PI data pertaining to reactor coolant system total leakage for the second quarter of 2000 to determine its accuracy and completeness. Documentation reviewed included the control room operator leakage calculations and data entries, licensee review and verification reports, and the PI data provided at the NRC web site.

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up

.1 (Closed) LER 50-296/2000-005-000: Scram During Level Transmitter Calibration. On May 24, 2000, Unit 3 experienced an automatic reactor scram that was caused by a pressure perturbation on the variable leg of the reactor vessel level instrumentation when a level instrument was being returned to service. This perturbation caused both channels of the reactor protection system level instrumentation to sense a low level and scram the reactor. All safety systems functioned as required.

Following the scram, reactor level momentarily dropped below the low level setpoint (-45 inches) which caused the high pressure coolant injection (HPCI) and the RCIC systems to start on a valid low level indication. The licensee determined that the lower than normal reactor water level immediately following the scram was caused by a higher pre-scram power level. This higher power level (power uprate) was approved by the NRC and implemented by the licensee on Unit 3 in Fall 1998. The licensee's review determined that the feedwater control system (FCS) responded as designed, however, the licensee indicated in the LER that they plan to evaluate the FCS to determine if initiation of HPCI and RCIC can be avoided for this type of transient.

The licensee's event investigation report, scram report, licensee event report, and PER were reviewed. The instrument procedure utilized to perform the calibration did not contain the specific valving sequence necessary to restore the instrument without perturbation of the reactor instrument sensing lines. The licensee found that the instrument maintenance personnel actions associated with the valve manipulation to restore the instrument to service were in accordance with management expectations and training; however, the procedure covering this activity did not adequately address the special precautions necessary while the unit was at power. If this procedure problem was left uncorrected, it would become a more significant safety concern. Based on the resultant scram caused by the level instrumentation pressure perturbation, this problem could increase the frequency of an initiating event (transient). However, because all mitigation systems remained operable and barrier integrity was not challenged, this finding is considered to be of very low safety significance (Green). The issue was placed in the licensee's corrective action program as PER 00-005345-000.

Since this level instrument is not a safety-related component, the inadequate procedure is not a violation of regulatory requirements.

.2 (Closed) LER 50-260/2000-001-000: Mode Change Not Allowed by Technical Specifications SR 3.0.4 Made During Reactor Startup. This event is discussed in Section 1R14 and resulted in an NCV. No new issues were revealed by the LER.

.3 (Closed) LER 50-296/2000-006-000: Main Steam Safety/Relief Valves Exceeded the Technical Specification Setpoint Tolerance Due to Pilot Valve Disc/Seat Bonding. The licensee identified that 8 of the 13 Unit 3 main steam safety/relief valves (SRVs) exhibited lift settings outside the TS setpoint tolerance during testing at Wyle Laboratory. The cause was attributed to corrosion bonding at the pilot valve disc/seat interface. The licensee found that Unit 3 was within the reload specific analysis for the operating cycle based on the as-found data and that the SRVs would not have exceeded TS safety limits during an abnormal operating transient.

This issue continues to be an industry problem and is being evaluated by the Boiling Water Reactor Owner's Group (of which the licensee is a member) SRV Drift Fix Development Committee and the valve manufacturers. This issue is a known industry problem, and not related to any deficient licensee performance. Because there was no firm evidence to establish the time of SRV inoperability, the licensee appropriately assumed the discovery time during surveillance testing and the inoperability time were synonymous. Based on these conditions, the issue was not evaluated under the SDP and no violation of regulatory requirements was identified. The LER was reviewed as being satisfactory and closed.

4OA5 Other

.1 (Closed) Temporary Instruction (TI) 2515-144: Performance Indicator (PI) Data Collecting and Reporting Process Review. The inspectors reviewed the licensee's PI data collecting and reporting process for the second quarter to determine if the process was consistent with the NRC supported, industry guidance contained in NEI 99-02, Revision 0, March 2000, Regulatory Assessment Performance Indicator Guidelines. TVA had issued Business Practice Procedure BP-243, Performance Indicator Information to NRC, applicable to all TVA nuclear facilities for managing data collecting and reporting methods. The inspector found that Procedure BP-243 consistently applied NEI 99-02 guidance in the areas of indicator definitions, data reporting elements, calculation methods, and clarifying notes for the following six PIs:

- Unplanned Power Changes per 7000 Critical Hours
- Safety System Unavailability
- Safety System Failures
- Emergency Response Organization Drill Participation
- Occupational Exposure Control Effectiveness
- Protected Areas Security Equipment Performance Index

.2 Unit 1 Lay-up and Equipment Preservation Program Inspection (92050)

a. Inspection Scope

The purpose of this inspection was to verify that the licensee was following the prescribed program established to preserve Unit 1 safety-related equipment, which is in long term lay-up in accordance with Procedure 0-TI-373, Plant Lay-up and Equipment Preservation, Revision 0. Although not currently involving safety-significant activities, the review of Unit 1 equipment preservation provides a periodic quality status of Unit 1 equipment. The inspectors reviewed the Unit 1 lay-up process, the preventive maintenance program for equipment preservation, and the water chemistry history of the suppression pool and reactor coolant. The Unit 1 TS and Technical Requirements Manual were reviewed to ensure compliance with those requirements applicable to the current plant condition (defueled).

The inspectors also reviewed the results of quality assurance observations of Unit 1 conducted during the week of August 7, 2000. These observations identified several deficiencies relative to the lay-up program, and were documented in PERs 00-008059, -60, and -62. In addition the inspectors reviewed a Chemistry Department follow-up Self-assessment BFN-CEM-00-006 conducted August 28 through September 1. These reviews identified a significant lack of dedicated management oversight to prioritize and provide the necessary resources to maintain Unit 1 lay-up as required by the program. The deficiencies identified were placed into the licensee's corrective action program under PER 00-008107-000. The inspectors also reviewed the corrective action plan for these issues, which focused on management oversight, clarification of procedures, establishment of equipment configuration status of each lay-up system, and setting up of a team composed of site and/or corporate engineering, chemistry, maintenance, and operations representatives to review current lay-up flow paths and systematically evaluate which equipment required continued dry lay-up.

To determine the actual lay-up condition of Unit 1, the inspectors conducted a walkdown inspection with emphasis on a sampling of two systems, standby liquid control and RCIC. In addition, the inspectors accompanied a Unit 1 auxiliary unit operator (AUO)to evaluate the effectiveness and thoroughness of the AUO rounds.

b. Findings

No findings of significance were identified.

.3 Review of World Association of Nuclear Operations (WANO) Report

On August 24, 2000, the inspectors reviewed the results of a WANO peer review of Browns Ferry performance conducted during the weeks of June 12 and 19, 2000. The report was dated August 2, 2000, and did not identify any significant issues that had not been previously addressed and/or reviewed by the NRC.

4OA6 Management Meetings

Exit Meeting Summary

The inspectors presented the inspection results to Mr. John Herron, Site Vice President, and other members of licensee management on September 28, 2000. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any 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, Plant Manager
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, Site Support Manager
R. Rogers, Maintenance Superintendent
G. Little, Operations Manager
R. Moll, System Engineering Manager
C. Ottenfeld, Chemistry Superintendent
D. Olive, Operations Superintendent
D. Sanchez, Training Manager
J. Schlessel, Project Manager
M. Scaggs, Maintenance and Modifications Manager
J. Wright, Design Engineering Manager
R. Wiggall, Site Engineering Manager

NRC
R. Bernhard, Region II Senior Reactor Analyst

LIST OF ITEMS OPENED AND CLOSED

Opened and Closed

50-260/00-04-01	NCV	Failure to Meet TS SR 3.0.4 for Instrument Channel Checks (Section 1R14).

Closed

50-296/2000-005-000	LER	Scram During Level Transmitter Calibration (Section 4OA3.1).
50-260/2000-001-000	LER	Mode Change Not Allowed by Technical Specifications SR 3.0.4 Made During Reactor Startup (Section 4OA3.2).
50-296/2000-006-000	LER	Main Steam Safety/Relief Valves Exceeded the Technical Specification Setpoint Tolerance Due to Pilot Valve Disc/Seat Bonding (Section 4OA3.3).

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

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

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.