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

April 22, 2002

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 INTEGRATED INSPECTION REPORT 50-259/01-05, 50-260/01-05, 50-296/01-05

Dear Mr. Scalice:

On March 23, 2002, 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 March 22, 2002, with Mr. Ashok Bhatnagar 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.

No findings of significance were identified.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

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: (See page 2)

Enclosure: NRC Integrated Inspection Report 50-259/01-05,
w/Attachment - Supplemental Information

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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 Nos:	50-259/01-05, 50-260/01-05, 50-296/01-05
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:	December 23, 2001 - March 23, 2002
Inspectors:	W. Smith, Senior Resident Inspector J. Starefos, Resident Inspector S. Vias, Senior Reactor Inspector (Sections 1R02, 1R17.1) M. Scott, Senior Reactor Inspector (Sections 1R02, 1R17.1) W. Bearden, Reactor Inspector (Section 1R02, 1R17.1) R. Carrion, Project Engineer (Section 1R06)
Approved by:	P. E. Fredrickson, Chief Reactor Projects Branch 6 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000259-01-05, IR 05000260-01-05, IR 05000296-01-05, on 12/23/2001-03/23/2002, Tennessee Valley Authority, Browns Ferry Nuclear Plant, Units 1, 2 and 3.

The inspection was conducted by the resident inspectors, three regional engineering branch inspectors, and a project engineer. The inspection did not identify any findings of significance. The NRC's program for overseeing the safe operation of commercial nuclear reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/OVERSIGHT/index.html.

A. NRC Identified Findings

None

B. Licensee Identified Violations

Two non-cited violations (NCVs) of very low safety significance, which were identified by the licensee, have been reviewed by the inspectors. Corrective actions taken or planned by the licensee appeared reasonable. The violations are listed in Section 4OA7 of the report.

Report Details

Summary of Plant Status

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

Unit 2 operated at or near full power, except as noted, with brief reductions in power scheduled to adjust control rods and other routine testing. On December 27, 2001, Unit 2 power was reduced for three days to do power suppression testing for a leak in the fuel cladding. As a result of the tests, three control rods were fully inserted to suppress the effects of the leaks. Power was restored to 100% on December 30. On January 5 and 6, 2002, additional power suppression testing was performed, and two additional control rods were fully inserted. A brief mid-cycle outage has been scheduled for late April 2002 to find and replace the leaking fuel rods. On January 12, Unit 2 power was reduced to 65% for approximately 12 hours to facilitate replacement of the direct current (DC) solenoid coil on outboard main steam isolation valve (MSIV) “D”.

Unit 3 operated at or near full power, except as noted, with brief reductions in power scheduled to adjust control rods and perform routine testing. On January 13, 2002, Unit 3 power was reduced to 85% for approximately four hours to facilitate repairs on feedwater heater 3B3. On February 7, reactor power started coasting down with fuel depletion as the March 26 refueling outage approached. As of the end of this inspection period, Unit 3 was operating at 87%.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope

During the season of susceptibility, the inspectors examined the licensee’s equipment and procedures which were established to cope with tornadoes. The high winds preceding tornados were considered as well as tornados themselves, because they can lead to a loss of redundant trains of safety-related equipment needed to mitigate the consequences of an accident. The inspectors performed a detailed review of Abnormal Operating Instruction 0-AOI-100-7, Tornado, and Business Practice BP-305, Site Tornado Procedure, and evaluated the actions required.

b. Findings

No findings of significance were identified.

1R02 Evaluations of Changes, Tests or Experiments

a. Inspection Scope

The inspectors reviewed selected samples of safety evaluations to verify that the licensee had appropriately considered the conditions under which changes to the facility or procedures may be made, and tests conducted, without prior NRC approval. The inspectors reviewed safety evaluations for nine design and procedure changes. The inspectors verified, through review of additional information, such as calculations, supporting analyses and drawings, that the licensee had appropriately concluded that the changes could be accomplished without obtaining a license amendment. The nine safety evaluations reviewed are listed in the Attachment.

The inspectors also reviewed samples of design/engineering packages and procedure changes for which the licensee had determined that evaluations were not required, and verified that the licensee's conclusions to "screen out" these changes were correct and consistent with 10 CFR 50.59. The 13 "screened out" changes reviewed are listed in the Attachment.

The inspectors reviewed the licensee's corrective action program and self-assessments of the 10 CFR 50.59 process to confirm that the licensee was identifying 10 CFR 50.59 issues, entering issues into the corrective action program and resolving concerns.

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 redundant or diverse train operability while one train was out-of-service. Consideration was given to the operable equipment configuration as required by the applicable operating procedures or selected drawings:

- Core spray (CS) Loop II on Unit 3 to verify operability while CS Loop I was tagged out for preventive maintenance (PM)
- Residual heat removal (RHR) Loop I on Unit 3 to verify operability while RHR Loop II was out of service for pre-outage maintenance
- RHR Loop II to verify operability while 3A emergency diesel generator (EDG) was out of service for the two-year inspection and PM
- EDG 3A, 3B, and 3C to verify operability while EDG 3D was out of service for the two-year inspection and PM

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 licensee control of transient combustibles and ignition sources, and the material condition and operational status of selected fire protection systems, equipment and features. The inspectors used area maps and pre-plan descriptions from the licensee's Fire Protection Report to verify that the equipment described was available. In addition, for some areas, surveillance instructions were referenced to verify the location of detection equipment:

- Fire Area (FA)-17, Unit 1 Battery and Battery Board Rooms
- FA-21, Unit 3 EDG Building
- FA-25, Cable Tunnel
- FA-25, Radwaste Building
- FA-25, Intake Pumping Station
- FA-25, Residual Heat Removal Service Water (RHRSW) Pump Rooms

On March 7, 2002, the inspectors performed the annual inspection of fire drills. The simulated fire was located in the Unit 2, 621 elevation of the reactor building near the reactor recirculation pump motor-generators. The readiness of licensee personnel to fight and prevent the spread of fire in a vital area was evaluated in terms of proper utilization of equipment needed to combat the fire, utilization of pre-planned strategies, effectiveness of communications, and meeting drill objectives. The inspectors reviewed the post-drill critique with the licensee representative controlling the drill to confirm a satisfactory level of self-critical discussion.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope

The inspectors reviewed plant design features and licensee procedures intended to protect the plant and its safety-related equipment from internal and external flooding events. The inspectors reviewed flood analysis documents including: Updated Final Safety Analysis Report (UFSAR) Amendment 19, Section 2.4, which included Appendix 2.4A, Maximum Possible Flood; the Browns Ferry Individual Plant Examination, Appendix B, dated March 20, 2000; and Emergency Operating Instruction 3, Secondary Containment Control, Revision 9, for licensee commitments. The inspectors also interviewed cognizant licensee personnel knowledgeable about site flood protection measures and plant drainage plans. The inspectors performed a walkdown of risk-significant areas, susceptible systems and equipment including the intake pumping station, the Unit 3 reactor building, and the Unit 3 turbine building for flood-significant features such as level switches, room sumps, and door seals. The plant procedures for coping with flooding events were also reviewed to verify that the actions were consistent with the plant's design basis assumptions. The procedures are listed in the Attachment.

The inspectors also reviewed the licensee's corrective action documents with respect to flood-related items identified in problem evaluation reports (PERs) written in 2001 to verify the adequacy of the corrective actions. The inspectors reviewed selected completed preventive maintenance procedures and work orders for identified level switches and pumps for completeness and frequency.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance

a. Inspection Scope

The inspectors observed the condition of the cleaned emergency equipment cooling water (EECW) heat exchangers on the 3C EDG, the end bell removal and as-found condition of the EECW heat exchangers on the 3A EDG, and portions of the eddy current testing and tube plugging on one of the two EECW heat exchangers on the 3D EDG. The inspectors reviewed the eddy current inspection results for diesel generator coolers 3C1, 3C2, 3A1, 3A2, 3D1, and 3D2, and verified that the total number of tubes plugged were within the acceptable range per TVA Nuclear calculations MD-Q0067-880201 and MD-Q0082-000016.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

On February 6, 2002, the inspectors observed reactor operator and senior reactor operator requalification training activities on the plant simulator. One scenario was observed which included three events and four demonstrations. The inspectors verified expected actions described in the licensee's simulator event guides. The inspectors discussed the performance of the crews with the instructor at the end of the inspection.

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):

- Repeated plugging of filters and failures of sample pumps on the Unit 2 and 3 Hydrogen/Oxygen Analyzers
- Reverse power trip of EDG D during surveillance testing due to dirty contacts on a type HFA relay
- Standby gas treatment Unit A outlet damper failed closed while the unit was running, causing unplanned availability of equipment without a functional failure
- Failure of the 3C condensate booster pump seal water tubing, resulting in the pump being removed from service while operating Unit 3 at full power. Consequently, there was an unplanned capability loss
- Functional failure of the 3A and 3B electric board room air conditioning units

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

The objectives of this inspection were to verify that risk assessments were being performed when and as required by 10 CFR 50.65(a)(4). The inspectors evaluated the adequacy of the licensee's risk assessments and the implementation of compensatory measures for the planned maintenance activities listed below. The inspectors also verified that, upon identification of the emergent equipment issue listed below, the licensee had taken the necessary steps to plan and control the resulting emergent work activities to effectively manage and thus minimize that risk. For some emergent work, the inspectors verified that timely reassessment of the resultant plant risk was performed.

- Unit 2: Replacement of DC solenoid coil on main steam line C outboard MSIV 2-FSV-001-0038B while at power per Work Order (WO) 01-013007-000 (Emergent)
- Unit 3: Sequencing of the planned conduct of the Unit 3 core spray system logic functional test in response to emerging problems associated with the two-year maintenance performed on EDG 3EA (Planned/Emergent)
- Unit 3: Inspection and cleaning of RHR heat exchanger 3A and preventive maintenance on RHR pump 3A concurrent with a clearance that rendered RHR service water (RHRSW) pumps B1 and B2 inoperable, while RHRSW pump D1 was out of service awaiting installation of a repaired motor (Planned)

- Unit 3: Replacement of a 4160-volt breaker (BFN-3-BKR-211-03ED/007) resulting in inoperability of standby gas treatment Train C, per WO 98-014890-082 (Planned)
- Unit 3: Take thermography readings on the electrohydraulic control (EHC) master trip solenoid coils, and take voltage readings to determine the condition of the master trip solenoid indication circuits to resolve extinguished lamp on switch 3-HS-47-67C (Emergent)

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-routine Plant Evolutions

.1 Inadvertent Actuation of Two EDGs

a. Inspection Scope

The inspectors reviewed personnel performance related to a non-valid, inadvertent actuation of Unit 2 EDGs B and D. On February 28, 2002, while testing the Unit 2 common accident signal logic, the EDGs started and came up to speed, contrary to the intent of the test procedure. The EDGs did not assume any loads, because the respective boards were already energized. The unit was operating at full power, providing electrical power to the boards. The EDGs were properly placed back in standby service. The licensee reported the non-valid actuation of the EDGs with a verbal event report to the NRC per 10 CFR 50.73(a)(2)(iv)(A).

b. Findings

No findings of significance were identified.

.2 Replacement of Temperature Switches Resulting in Inoperable Reactor Core Isolation Cooling (RCIC)

a. Inspection Scope

The inspectors reviewed personnel performance related to an unintended replacement of the Unit 3 RCIC temperature switches with calibrated HPCI temperature switches. This mistake rendered the RCIC system inoperable

b. Findings

On January 9, 2002, supporting craft personnel inadvertently and incorrectly replaced the reactor RCIC temperature switches with calibrated HPCI temperature switches instead of replacing the HPCI switches with calibrated HPCI switches as required by Procedure 3-SR-3.3.6.1.3(3D). This caused related containment isolation instrumentation to become inoperable because the capability of the system to isolate the steam source in the event of a leak in the steam line space was lost. The operators were not informed of the problem until nearly six hours later, placing the unit in a condition prohibited by TS 3.3.6.1, in that the steam path was required to be manually isolated within one hour after the automatic isolation feature was rendered inoperable. After being informed of the error, the operators promptly (within 45 minutes) isolated the steam path, which, in turn, rendered RCIC inoperable.

The licensee initiated PER 02-233-000 to enter the problem(s) into the corrective action program. The inspectors reviewed the PER Summary Report later and noted that the root causes of the personnel error were related to improper verification methodologies and poor self-checking. The corrective actions planned and executed were appropriate for the circumstances. This issue was determined to be of very low safety significance because the as-installed HPCI switches were set at 170 degrees, F whereas the required setpoint for RCIC was 147 degrees. The difference may translate to a slightly greater loss of steam in the event of a leak or break. This problem was entered into the licensee's corrective action program as PER 02-233-000. The enforcement aspects of this licensee identified finding are described in Section 4OA7.

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 as permitted by Generic Letter 91-18 (Information to Licensees Regarding NRC Inspection Manual Section on Resolution of Degraded and Nonconforming Conditions), and the SSC remained available or was addressed by plant TS or Technical Requirements Manual limiting conditions for operation such that no unrecognized increase in risk occurred:

- Engineering evaluation performed in response to degraded RHR snubber 2-SNUB-074-5005 found during 2-SI-4.6.H-1
- Limit switch problems with primary containment isolation valve 3-FCV-75-58 on January 4, 2001
- Pressure instrument 2-PIS-1-82 reading lower due to nearly closed root valve combined with packing leak
- Primary containment isolation valve 3-FCV-73-81 opened when carpenter inadvertently actuated local pushbutton
- Predictive Monitoring Test Report on Unit 3 RHR Pump 3B, after changing motor bearing oil and finding more particulates in the new oil after running the pump for over an hour, than in the oil that was removed.
- Engineering functional evaluation in response to degraded Unit 3 core spray room cooler 3A when a pinhole leak was found on the cooling coil, which was cooled by EECW

b. Findings

No findings of significance were identified.

1R16 Operator Work-Arounds

a. Inspection Scope

The inspectors reviewed the status of operator workarounds for Units 2 and 3 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 selected and reviewed in detail:

- Unit 3, Priority 1, reactor recirculation pump motor-generator 3A scoop tube (Bailey control) locked in position to mitigate power oscillations at certain speeds. Operator must reset the scoop tube before it will respond to a runback.
- Unit 3, Priority 1, low pressure heater strings may isolate during combined intercept valve testing. Operator should isolate the heater string isolation logic before commencing the tests

The inspectors also reviewed the cumulative effects of operator workarounds on both Units 2 and 3 that could increase an initiating event frequency or that could affect multiple mitigating systems. The review also considered the cumulative effects of operator workarounds on the ability of operators to respond in a correct and timely manner to plant transients and accidents.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

.1 Biennial Review of Modifications

a. Inspection Scope

The inspectors evaluated design change packages for 22 modifications, in all three cornerstone areas, to verify that the modifications did not adversely affect system availability, reliability, or functional capability. The inspectors verified inspection procedure attributes such as whether:

- Energy requirements can be supplied by supporting systems
- Materials and replacement components were compatible with physical interfaces
- Replacement components were seismically qualified for application
- Code and safety classification of replacement system, structures, and components were consistent with design bases
- Modification design assumptions were appropriate
- Post-modification testing would establish operability
- Failure modes introduced by the modification were bounded by existing analyses
- Appropriate procedures or procedure changes had been initiated.

For selected modification packages, the inspectors verified that the as-built configuration accurately reflected the design documentation.

Documents reviewed included procedures, engineering calculations, modifications, work orders, site drawings, corrective action documents, applicable sections of the living UFSAR, supporting analyses, TS, and design basis documentation. The inspectors also reviewed the results of the licensee's recent self-assessments of the design change process. The documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

.2 Periodic Reviews of On-Line Modifications

a. Inspection Scope

The inspectors reviewed Design Change Notice 50896, which terminated continuous operation of the Unit 2 and 3 hydrogen/oxygen analyzers and deactivated malfunction indicators that annunciated when the analyzers were placed in a standby mode. The modification was performed while the units were on-line. The inspectors verified that the modifications did not place the plant in an unsafe condition, that key safety functions and emergency/abnormal procedure actions were not impaired, and that the recent implementation of hydrogen water chemistry did not affect the decision to secure the analyzers. Evaluations pursuant to 10 CFR 50.59 were reviewed to ensure that NRC approval was obtained where required.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors evaluated the following activities by observing testing in progress and/or reviewing completed documentation to verify that the post maintenance test (PMT) was adequate to verify system operability and functional capability following completion of associated work:

- PMT of replacement DC solenoid coil on main steam line C outboard MSIV 2-FSV-001-0038B WO 01-013007-000
- PMT and rebaselining of EECW pump 3C after adjusting wear ring clearance per WO 02-000494-000
- PMT of EDG 3A after foreign material was removed from the generator cooling fan and generator windings by WOs 02-000837-000 and 02-000837-001
- PMT of replacement 4160-volt feeder breaker for the power supply feeding standby gas treatment (SBGT) train C per WO 98-014890-082

- PMTs of Unit 3 RCIC following system outage work in accordance with WOs 01-08312-000 and 01-07360-000T

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

Unit 3 Refueling Outage

a. Inspection Scope

The inspectors observed portions of the licensee's receipt inspection of new fuel for Unit 3 Refueling Outage U3C10 to confirm that the licensee handled and inspected the new fuel in accordance with Procedure 0-GOI-100-2, New Fuel Operations, and had appropriately resolved deficiencies.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors observed surveillance testing in progress and/or reviewed test data of selected risk-significant SSCs, listed below, to assess, as appropriate, whether the SSCs met TS, UFSAR, and licensee procedure requirements, and to verify that the testing effectively demonstrated that the SSCs were operationally ready and capable of performing their intended safety functions:

- Surveillance Instruction 3-SI-4.5.C.1(2), C3 EECW pump operational inservice test per ASME Code Section XI (All Units)
- Surveillance Requirement (SR) 2-SR-3.3.1.1.8(5), MSIV reactor protection system (RPS) trip channel functional test (Unit 2)
- SR 3-SR-3.8.1.1(3A), diesel generator 3A monthly operability test (Unit 3)
- SR 2-SR-3.5.1.7, HPCI main and booster pump developed head and flow rate quarterly inservice test per ASME Code Section XI (Unit 2)
- SR 3-SR-3.3.5.1.5 (CS II) , CS system Loop II logic time delay relay calibration test (Unit 3)
- SR 0-SR-3.8.1.6, common accident signal logic test (Unit 2)

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 and pending temporary plant modifications, documented on temporary alteration control forms (TACFs) provided by the licensee. TACF 0-2001-002-065, SBGT Train A/B Cross Tie Damper 0-DMP-065-0022 motor removal and blocking the damper shut to support repairs on SBGT Train A suction damper was 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 to verify that the alteration did not adversely affect the safety functions of important safety systems. Where practicable, the installed hardware was inspected to verify proper configuration and to ensure that there were no interferences with operable systems.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

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

Cornerstone: Mitigating Systems

Safety System Unavailability - Emergency AC Power

a. Inspection Scope

The inspectors verified the accuracy and completeness of the licensee’s PI data on Safety System Unavailability, Emergency AC Power, for the fourth quarter of 2001, for Units 2 and 3. Data reviewed included the operators’ logs, the licensee’s PI submittal to the NRC covering the fourth quarter 2000 through the fourth quarter 2001, the licensee’s monthly operating reports, and licensee event reports.

b. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution

a. Inspection Scope

The inspectors selected PER 02-00233-000, dated January 9, 2002, for detailed review. The PER was initiated to identify a personnel error, where maintenance technicians removed the RCIC steam line space temperature switches instead of the HPCI switches required by the work order. The PER was reviewed by the inspectors to ensure that the full extent of the issues involved were addressed, that an appropriate evaluation was performed, and that corrective actions were specified and prioritized commensurate with the cause(s).

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up

(Closed) Licensee Event Report (LER) 50-296/2002-001-000: Failure to Meet TS Requirements for Inoperable Primary Containment Isolation Instrumentation Due to Removal of Incorrect Temperature Switches. This event was previously discussed in Section 1R14.2 of this report. No new information was found during the review of this LER. There was a licensee-identified NCV (Green) for failure to comply with the applicable work instructions. described in Section 4OA7.

4OA6 Management Meetings

.1 Exit Meeting Summary

The inspectors presented the inspection results to Mr. Ashok Bhatnagar and other members of licensee management on March 22, 2002. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

.2 Annual Assessment Meeting Summary

On March 22, 2002, the Region II, Division of Reactor Projects Branch Chief and the Senior Resident Inspector assigned to Browns Ferry met with Tennessee Valley Authority, to discuss the NRC's Reactor Oversight Process (ROP) and the Browns Ferry annual assessment of safety performance for the period of April 1, 2001 - December 31, 2001. The major topics addressed were: the NRC's assessment program, the results of the Browns Ferry assessment, and the NRC's Agency Action Matrix. Attendees included Browns Ferry site management, members of site staff, local officials, and members of the public.

This meeting was open to the public. Information used for the discussions of the ROP is available from the NRC's document system (ADAMS) as accession number ML020600179. ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

4OA7 Licensee-Identified Violations

The following findings of very low significance were identified by the licensee and are violations of NRC requirements and which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600 for being dispositioned as NCVs.

The licensee was informed that, If either NCV is to be denied, they should provide a response with the basis for denial, within 30 days of the date of this inspection report, to the U. S. Nuclear Regulatory Commission, Attn: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Browns Ferry facility.

NCV Tracking Number	Requirement Licensee Failed to Meet
NCV 50-296/01-05-01	(Green) TS 5.4.1.a requires, in part, that written surveillance procedures as recommended in Regulatory Guide 1.33, Revision 2, Appendix A, shall be implemented. Surveillance Procedure 3-SR-3.3.6.1.3(3D) required HPCI steam line space temperature switches 3-TS-73-2N, -2P, -2R, and -2S to be replaced. Contrary to this, RCIC steam line space temperature switches 3-TS-71-2N,-2P, -2R, and -2S were replaced. Consequently, the unit was placed in a condition prohibited by TS, unknown by the control room operators until about six hours had transpired. This issue was determined to be of very low safety significance because the as-installed HPCI switches were set at 170 degrees, F whereas the required setpoint for RCIC was 147 degrees. The difference may translate to a slightly greater loss of steam in the event of a leak or break. The violation was entered into the licensee's corrective action program as PER 02-233-000.
NCV 50-260/01-05-02	(Green) TS 5.4.1.a requires, in part, that written equipment control procedures as recommended in Regulatory Guide 1.33, Revision 2, Appendix A, shall be implemented. SPP 10.1, System Status Control, requires the responsible individual to ensure that work documents restore systems and equipment to the correct configuration. Contrary to the above, root valve 2-RTV-001-0220A, main steam line pressure source for pressure transmitter PT-1-82, was found throttled nearly closed during plant operations after the valve was repaired, when it should have been open. The licensee concluded the valve allowed pressure to be transmitted until the packing began to leak, thereby bleeding off part of the pressure. The operators became aware of a problem as the redundant instruments mismatched.

This issue was determined to be of very low safety significance because the instrument senses main steam line low pressure (steam leak/break detection), and sensing slightly lower pressure would activate the isolation conservatively. This violation was entered into the licensee's corrective action program as PER 02-135-000.

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

T. Abney, Licensing Manager
A. Bhatnagar, Site Vice President
L. Clardy, Site Quality Assurance Manager
R. Coleman, Radiological Control Superintendent
J. Corey, Radiation Protection and Chemistry Manager
T. Cornelius, Emergency Preparedness Supervisor
R. Jones, Plant Manager
G. Little, Operations Manager
T. Niessen, Jr., Site Support Manager
J. Rupert, Unit 1 Project Manager
D. Sanchez, Maintenance and Modifications Manager
M. Scaggs, Assistant Plant Manager
T. Trask, Acting Site Engineering Manager
J. Valente, Unit 1 Project Engineering Manager

NRC

R. Bernhard, Region II Senior Reactor Analyst

ITEMS OPENED AND CLOSED

Open and Closed

50-296/01-05-01	NCV	Failure to Follow SI Instruction to Replace HPCI Steam Line Space Temperature Switches. RCIC switches were Inadvertently replaced (Section 4OA7).
50-296/01-05-01	NCV	Failure to Follow System Status Control, procedure to Restore Systems Configuration. Root Valve Partially Closed for Main Steam Pressure Transmitter. (Section 4OA7).

Closed

50-296/2002-001-000	LER	Failure to Meet TS Requirements for Inoperable Primary Containment Isolation Instrumentation Due to Removal of Incorrect Temperature Switches (Section 4OA3).

LIST OF DOCUMENTS REVIEWED

Sections 1R02

Safety Evaluations

- DCN 41356 Removal of Core Spray Testable Check Valves Panel Indication
- DCN 40713 Replace RCIC Steam Isolation Valves
- DCN 40283 Replace Existing Unit 1 & 2 Control Bay Chillers
- EEC 50095 Replace Existing Float Level Switches
- DCN 39881 Enable U3 Oscillation Power Range Scram Trip and Rod Block
- DCN 50364 Modify Control Rod Scram Timing Circuit Input into Integrated Computer
- DCN 50884 Modify U3 SJAE Control Circuits for Steam Isolation Valve Position Interlocks
- TACF-2-2001-09-92-0 Reinstall U2 Oscillation Power Range Monitors Scram Trip Function Jumpers
- TACF-2-2001-01-79-0 Disable U2 Refuel Interlocks to Allow RPIS Maintenance

Screened Out Safety Evaluations

- DCN 50158 Reactor Water Level III Setpoint Changes
- TACF-1-2001-01-32-0
- TACF-0-2001-01-67-0
- TACF-3-2001-01-31-0
- WO 98-015555-000-0
- EDC 50751 Delete need for CA Dryers
- DCN 41301 Modify Disc of 2/3-FCV-71-39 by Drilling Down Stream Disc Face to Prevent Pressure Locking
- DCN 50782 Remove Recirculation Pump Motor HVAC Dampers DMP-70-1022 & 1023
- EDC 50657 Substitute Replacement RHRSW Thermal Relief Valves Calculation
- DCN 50195 Modify HVAC exhaust Fan Circuits to Provide Additional Trip Contactors
- DCN 25853 Replace EDG-125VDC Battery Chargers
- DCN 50868 Replace Damaged Portion of Cables for Emergency Equipment Cooing Water from Handhole 15 to Intake Pumping Structure
- EDC 50548 Update Calculations for Replacement DG Jacket Water Heat Exchangers

Procedures

- SPP-9.4, 10CFR50.59 Evaluation of Changes, Tests and Experiments, Rev 2, Rev 3, and Rev 4

- SPP-9.3, Plant Modifications and Engineering Change Control, Rev 6
- SPP-9.5, Temporary Alterations, Rev 4

Self Assessment Documents

- CRP-ENG-01-012, Self Assessment Report 50.59 Multi -Site Evaluation, dated 8/24/01
- NA-CH-00-002, Corrective Action Program, dated 10/20/00
- BFN-ENG-01-006, Self Assessment on Design Quality, dated 9/1/01
- SSA-0006, Engineering Functional Area Audit, dated 2/16/01
- NA-BF-01-001, Design Performance and Capability Assessment, dated 1/29/01

Section 1R06

Procedures

- 0-AOI-100-3, Flood Above Elevation 558'
- MPI-0-260-DRS001, Mechanical Preventive Instruction for Inspection and Maintenance of Doors
- MPI-0-000-INS001, Inspection of Flood Protection Devices
- EPI-0-077-SWZ002, Electrical Preventive Instruction for Functional Check of the Reactor Building Flood Level Switches
- MAI-3.4B, Modification and Addition Instruction for Installation of Flood and Moisture Intrusion Seals

Section 1R17

Document Change Notices (DCN)

- DCN 41356, Removal of Core Spray Testable Check Valves Panel Indication
- DCN 40713, Replace RCIC Steam Isolation Valves
- DCN 50158, Reactor Water Level III Setpoint Changes
- DCN 40283, Replace Existing Unit 1 & 2 Control Bay Chillers
- DCN 41301, Modify Disc of 2/3-FCV-71-39 by Drilling Down Stream Disc Face to prevent Pressure Locking
- DCN 50782, Remove Recirculation Pump Motor HVAC Dampers DMP-70-1022 & 1023
- DCN 50195, Modify HVAC exhaust Fan Circuits to Provide Additional Trip Conrtactors
- DCN 25853, Replace EDG-125VDC Battery Chargers
- DCN 39881, Enable U3 Oscillation Power Range Scram Trip and Rod Block
- DCN 50364, Modify Control Rod Scram Timing Circuit Input into Integrated Computer
- DCN 50868, Replace Damaged Portion of Cables for Emergency Equipment Cooing Water from Handhole 15 to Intake Pumping Structure
- DCN 50884, Modify U3 SJAE Control Circuits for Steam Isolation Valve Position Interlocks

Engineering Equivalancy Changes (EEC)

- EEC 50095, Replace Existing Float Level Switches

Temporary Alteration Forms (TACF)

- TACF-2-2001-09-92-0
- TACF-2-2001-01-79-0
- TACF-1-2001-01-32-0
- TACF-0-2001-01-67-0
- TACF-3-2001-01-31-0

Engineering Document Changes (EDC)

- EDC 50751, Delete need for CA Dryers
- EDC 50657, Substitute Replacement RHRSW Thermal Relief Valves Calculation
- EDC 50548, Update Calculations for Replacement DG Jacket Water Heat Exchangers

Procedures

- SPP-9.4, 10CFR50.59 Evaluation of Changes, Tests and Experiments, Rev 2, Rev 3, and Rev 4
- SPP-9.3, Plant Modifications and Engineering Change Control, Rev 6
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Self Assessment Documents

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