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

October 22, 2001

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 50-259/01-03, 50-260/01-03, 50-296/01-03

Dear Mr. Scalice:

On September 22, 2001, 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 24, and October 12 and 22, 2001, with Mr. R. G. Jones and Mr. A. Bhatnager, respectively, 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 (one was Green and one had No Color). These issues were determined to involve violations of NRC requirements. However, because the violations were of very low safety significance and because the problems were entered into your corrective action program, the NRC is treating these issues as non-cited violations, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny any non-cited violation in the enclosed report, 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.

Since September 11, 2001, your staff has assumed a heightened level of security based on a series of threat advisories issued by the NRC. Although the NRC is not aware of any specific threat against nuclear facilities, the heightened level of security was recommended for all nuclear power plants and is being maintained due to the uncertainty about the possibility of additional terrorist attacks. The steps recommended by the NRC include increased patrols, augmented security forces and capabilities, additional security posts, heightened coordination with local law enforcement and military authorities, and limited access of personnel and vehicles to the site.

The NRC continues to interact with the Intelligence Community and to communicate information to you and your staff. In addition, the NRC has monitored maintenance and other activities which could relate to the site's security posture.

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: NRC Inspection Report 50-259/01-03, 50-260/01-03, 50-296/01-03 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:	50-259/01-03, 50-260/01-03, 50-296/01-03
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:	June 24 - September 22, 2001
Inspectors:	W. Smith, Senior Resident Inspector J. Starefos, Resident Inspector E. DiPaolo, Resident Inspector W. Bearden, Reactor Inspector D. Jones, Senior Health Physicist E. Testa, Senior Health Physicist R. Carrion, Project Engineer P. Fillion, Reactor Inspector
Approved by:	P. E. Fredrickson, Chief Reactor Projects Branch 6 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000259-01-03, IR 05000260-01-03, IR 05000296-01-03, on 06/24-09/22/2001, Tennessee Valley Authority, Browns Ferry Plant, Units 1, 2 and 3, refueling and outage activities, and other.

The inspection was conducted by the resident inspectors and regional maintenance, engineering, and radiation protection specialists. The inspection identified one No Color and one Green finding, which were non-cited violations (NCVs). The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply are indicated by "No Color" or by the severity level of the applicable violation. 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. Inspector Identified Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified an NCV for failure to meet 10 CFR 50, Appendix R, Criterion III.L.2.b requirements for alternative shutdown involving loss of the residual heat removal (RHR) function following certain postulated fires. The RHR function would have been lost due to inadvertent closure of the RHR pump minimum flow control valves due to fire damage to control cables, because the cables were not protected as required by 10 CFR 50, Appendix R.

  This finding was of very low safety significance because the initiating event was of relatively low frequency, fire suppression systems and diverse systems for core heat removal remained available (Section 4OA5).

- No Color. The inspectors identified a Severity Level IV NCV for failure to meet 10 CFR 50.59 requirements, in that the safety evaluation conducted as required by 10 CFR 50.59 did not adequately provide the basis that a procedure change would not result in more than a minimal increase in the likelihood of occurrence of a malfunction of equipment important to safety previously evaluated in the Updated Final Safety Analysis Report.

  The finding's underlying technical issue was evaluated and determined to be of very low safety significance because in the worst case scenarios only a single train of equipment would malfunction because of, for example, a severe pipe failure, and the Technical Specifications would govern (Section 1R20).

B. Licensee Identified Violation

A violation of very low safety significance which was identified by the licensee has been reviewed by the inspectors. Corrective actions taken or planned by the licensee appeared reasonable. This violation is 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 perform routine testing. On July 25, 2001, Unit 2 reactor scrammed on a turbine trip caused by an invalid mismatch between turbine power and generator output. The electro hydraulic control system was not properly adjusted to reflect actual mismatches. The circuits were subsequently adjusted and on July 28, Unit 2 was restored to full power operation.

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 July 26, 2001, power was decreased to 68% by inserting control rods in response to decreasing vacuum in the condenser, caused by steam jet air ejector (SJAE) failures. One SJAE was restarted after resolving the cause of the failure and subsequently, the unit was restored to full power.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R04 Equipment Alignment

a. Inspection Scope

The inspectors performed a partial walkdown of the below-listed systems to verify redundant train operability while one train was out of service. Consideration was given to the operable trains' configuration as required by the applicable operating procedures. The inspectors questioned any existing danger or caution tags applied to operating trains, and verified there was no work in progress that could affect operability.

- While the 250V DC Battery 3EB Test was being performed, the inspectors verified accessible critical portions of the alternate power supplies for two safety related loads. The two safety related loads were determined from licensee Drawing 3-45E709-2 to be 4160V shutdown board 3EB control power and Unit 3 ATWS power distribution channel B.
- Unit 3 core spray loop II alignment while loop I was out-of-service for preventive maintenance.
- AC and DC power source availability while 4160V shutdown board C control power battery was inoperable for a modified performance test.

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. Acceptance standards for the above conditions are delineated in the licensee's Fire Protection Plan.

- Fire Zone 2-5, Unit 2 Reactor Building EL 621 and EL 639 North of Line R
- Fire Zone 2-6, Unit 2 Reactor Building EL 639 South of Line R
- Fire Area 6, Unit 1A 480V Shutdown Board Room
- Fire Area 7, Unit 1B 480V Shutdown Board Room
- Fire Zone 3-3, Unit 3 Reactor Building EL 593 and residual heat removal (RHR) Heat Exchanger Rooms
- Fire Zone 3-4, Unit 3 Reactor Building EL 621 and EL 639 North of Line R

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

On August 22 and 23, 2001, the inspector observed reactor operator and senior reactor operator requalification training activities in the plant simulator. The subsequent evaluators' discussions and feedback to the crew were observed on August 22. The inspection was performed to evaluate licensee compliance with 10 CFR 55.59. In addition, the inspector compared a selected sample of the simulator boards with the actual control room board configuration and discussed discrepancies found with training personnel.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

.1 Periodic Evaluation

a. Inspection Scope

The inspector reviewed a maintenance rule (MR) periodic assessment (April 1, 1998, through March 31, 2000), quarterly system status reports, and MR self-assessment reports issued since completion of the periodic assessment to determine that the periodic assessment report met the requirements of 10 CFR 50.65(a)(3), 0-TI-346, Maintenance Rule Performance Indicator Monitoring, Trending, and Reporting, and SPP-6.6, Maintenance Rule Performance Indicator Monitoring, Trending, and Reporting. The inspector verified that the periodic assessment was issued in accordance with the time requirements of the MR and included an evaluation of balancing reliability, unavailability, 10 CFR 50.65(a)(1) and (a)(2) activities, and use of industry operating experience. The inspector reviewed selected MR activities covered by the assessment period for the following risk significant systems to verify compliance with 10 CFR 50.65: main steam safety relief valves, main turbine control, high pressure coolant injection (HPCI), emergency diesel generator (EDG), and diesel starting air. The inspector also reviewed selected MR activities associated with containment hydrogen analyzers and radiation monitoring. The inspector also reviewed licensee documentation associated with corrective actions and reclassification of HPCI and containment hydrogen analyzer systems which had previously been classified as (a)(1). The procedures and documents reviewed during the inspection are listed below:

- Design Criteria Document, BFN-50-7082, Standby Diesel Generator System
- MR system quarterly reports for the 3rd and 4th quarter fiscal years 2000, 1st and 2nd quarter fiscal year 2001
- MR performance improvement plan for main steam safety relief valves, June 2001
- MR performance improvement plan for main turbine electrical hydraulic control system, February 2000
- Raw water system quarterly evaluation for 2nd quarter fiscal year 2001.
- Problem Evaluation Report (PER) 99-011997-000, EDG 3B air start system low pressure
- Self Assessment, BFN-ENG-00-013, Maintenance Rule Program dated 3/21/00
- MR Second Periodic Assessment Report 10 CFR 50.65(a)(3), dated 7/16/00

b. Findings

No findings of significance were identified.

.2 Maintenance Effectiveness Routine Evaluation

a. Inspection Scope

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

- Functional failure of Unit 2 area radiation monitors 2-RM-90-13, 2-RM-90-14, and 2-RM-90-20 due to a failed power supply
- Failure of Unit 2 high pressure coolant injection (HPCI) suction transfer Switch 2-LS-073-0057A during surveillance testing and while channel B was out of service as required by the test procedure
- Failure of Unit 3 emergency diesel generator C lubricating oil circulating (soak-back) pump
- Failure of Unit 2 RHR pump 2B discharge check valve 2-CHK-074-599B to hold pressure

- Unit 2 RHR Loop II valve logic relay fingers found bent during surveillance testing

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 maintenance 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.

- Replacement of Unit 3 reactor core isolation cooling (RCIC) delay relay (3-RLY-071-13A-K42) that failed during surveillance testing per Work Order (WO) 01-006640-000 completed on June 29, 2001 (emergent)
- Failure of Unit 2 B outboard main steam isolation valve DC pilot solenoid coil (2-FSV-0001-27B) including protection of redundant plant equipment (emergent)
- Calibration and adjustment of steam pressure regulating valve 3-PC-1-152 after shifting the operating SJAE, per WO 001-006435-000 (emergent)
- Troubleshooting power supply failure to A and D emergency equipment cooling water (EECW) strainers and valve 0-FCV-067-0049 per WO 01-007891-000 and WO 01-007892-000 (emergent)
- Replacement of the EDG 3B jacket water heater because of the ground it caused on the 480V shutdown board 3A per WO 01-008281-001 (emergent)
- Replacement of Unit 2 electro hydraulic control (EHC) power supply involving risk of turbine trip (planned)

b. Findings

No findings of significance were identified.

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 that the evaluated SSC remained available such that no unrecognized increase in risk occurred:

- Compensatory measures implemented on Units 2 and 3 to detect and suppress thermal hydraulic instability oscillations due to the oscillation power range monitors being declared inoperable as a result of 10 CFR Part 21 Notification
- Operability of Unit 2 reactor pressure C anticipated transient without scram recirculation pump trip (ATWS-RPT) instrumentation with indicated failure of isolator module input to electro-hydraulic control system
- Operability of the Unit 2 to Unit 3 RHR crossover when motor operated valve 2-FCV-74-101 failed to meet the surveillance acceptance criterion of 95 seconds to open
- Operability of secondary containment during breach of 2A/2B shutdown boardroom coolers for chiller modification W/P 40287-113 & 114
- Technical evaluation of replacing RHR permissive for ADS pressure switch 2-PS-74-8B with a different pressure range switch (2-PS-31-7206D) through use of Design Change Notice 50877

b. Findings

No findings of significance were identified.

1R16 Operator Work-Arounds

a. Inspection Scope

The inspectors reviewed the status of operator workarounds for both units to determine if the functional capability of the system or operator reliability in responding to an initiating event was affected. A Priority 2 operator workaround, involving periodic manual rotation of the A and D EECW strainers due to power supply failures, was selected and reviewed in detail. The review included evaluating the effect of the operator workaround on the operator’s ability to implement abnormal or emergency operating procedures.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed the performance of the following activities to verify that the post maintenance test (PMT) addressed the nature of the work done and was adequate to verify system operability and functional capability:

- PMT of Unit 3 RCIC replacement time delay relay 3-RLY-071-13A-K42 per work order (WO) 01-006640-000
- PMT of D2 RHR service water (RHRSW) pump following replacement per WO 01-002255-000
- PMT of C2 RHRSW pump following impeller gap adjustment per WO 01-006513-000

- PMT of EDG 3B jacket water heater after replacement per WO 01-008281-001
- PMT of shutdown board room air handling unit 2B conversion for new chillers per 0-PMT-031.067
- PMT of new cables installed to supply power to the Kinney EECW strainer panels in RHRSW bays A & D per WO 01-009870-000

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

a. Inspection Scope

The inspectors reviewed the licensee's actions in response to a primary coolant leak found in a Unit 2 Code Class I core spray (CS) system vent line valve, while the Unit was in Mode 3 (hot shutdown), preparing to begin a refueling outage. The inspectors verified that the applicable Technical Specifications (TS) and TRM requirements were being met.

b. Findings

One non-SDP finding was identified with a very low safety significant underlying technical issue (No Color). The finding was identified as a Severity Level IV non-cited violation of 10 CFR 50.59, for failure to provide an adequate basis in a 10 CFR 50.59 safety evaluation that a procedure change would not result in more than a minimal increase in the likelihood of occurrence of a malfunction of equipment important to safety previously evaluated in the Updated Final Safety Analysis Report (UFSAR).

While in Mode 3, in preparation for the Unit 2, Cycle 11 refueling outage, the licensee discovered a through-wall leak on the vent line from the core spray loop II injection manual shut-off Valve 2-HCV-75-55. The vent line piping was classified as ASME Code Class 1 piping. The licensee determined that the leak did not represent reactor coolant pressure boundary leakage, as defined by the TS, because the leak could be isolated by shutting valve 2-HCV-75-55, however, the leak was not isolated by the operators at that time. This was allowed by the TRM, under TR 3.4.3, Structural Integrity, Revision 22, which allowed a 48-hour completion time for determining that the structural integrity non-compliance did not adversely affect the operability of the affected component. As an alternative, the affected component was required to be isolated within the same 48 hours. Operation of the plant continued in Mode 3 until the plant was placed in Mode 4 (cold shutdown) over 2 days later. The evaluation performed to satisfy TR 3.4.3 showed that the structural integrity non-compliance did not adversely affect the operability of the CS system. The inspectors did not identify any issues with the evaluation.

During the review of this issue, the inspectors noted that Revision 22 of the TRM changed the required actions in TR 3.4.3 when the structural integrity of components were discovered to not meet the ASME Code (i.e., boundary leakage). Specifically, the previous version of TR 3.4.3 required the licensee to immediately restore the structural integrity of the affected ASME Code Class component to within its limit or isolate the affected component from all operable systems. Appendix B of the UFSAR states that the change control process for the TRM is provided in Section 5.0 of the TRM. TRM Section 5.1.2 states that the licensee may make changes to the TRM without prior NRC approval provided the changes have been determined not to require NRC approval pursuant to 10 CFR 50.59.

The inspectors reviewed the changes implemented by Revision 22, including the safety evaluation (10 CFR 50.59 review), and found that the evaluation did not adequately demonstrate that the Revision 22 change to TR 3.4.3 would not result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

The basis for the NRC determination that the evaluation provided an inadequate basis was that the evaluation did not consider that this change provided at least two more days of exposure time that the flaw could be subjected to the stresses of high energy fluids, and did not consider all possible piping systems and flaw sizes for which this change could be applied. The safety evaluation issue was evaluated as a non-SDP finding because 10 CFR 50.59 issues involve the potential for impacting the NRC's ability to perform its regulatory function. However, the finding's underlying technical issue was evaluated and determined to be of very low safety significance, because in the worst case scenarios only a single train of equipment would malfunction, due to the consequences of a piping failure, and the TS would directly address that failed system with a limiting condition for operation (LCO). The finding was determined to be a No Color finding, due to the non-SDP nature of the finding, and the very low safety significance of its underlying technical issue.

10 CFR 50.59, paragraph (c)(1) states, in part, that a licensee may make changes to procedures as described in the UFSAR without obtaining a license amendment pursuant to 10 CFR 50.90 only if the change does not result in more than a minimal increase in the likelihood of occurrence of a malfunction of equipment important to safety previously evaluated in the UFSAR. Paragraph d(1) states in part that the written evaluation of changes to procedures must provide the basis for the determination that the change does not require a license amendment. The safety evaluation for Revision 22 of TR 3.4.3 did not adequately provide the basis that this procedure change would not result in more than a minimal increase in the likelihood of occurrence of a malfunction of equipment important to safety previously evaluated in the UFSAR. This finding is a violation of 10 CFR 50.59; however, because the licensee has included this problem in the corrective action program (PER 01-010639-000), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy, and is identified as NCV 50-260,296/01-03-01, Failure to Meet 10 CFR 50.59 Requirements.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors witnessed surveillance tests 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 determine if the testing effectively demonstrated that the SSCs were operationally ready and capable of performing their intended safety functions:

- Surveillance Procedure (SP) 3-SR-3.3.6.1.6(4), RCIC Time Delay Relay Calibration, completed satisfactorily for all relays except the K42 relay that failed to actuate, was replaced and post-maintenance tested
- SP 2-SR-3.5.1.6(CS II), CS Flow Rate Loop II
- SP 0-SR-3.8.1.1(C), EDG C Monthly Operability Test
- SP 2/3-SR-2, Drywell Unidentified and Identified Leakage Calculations, Revision 26 and 27, respectively, performed
- SP 2-SR-3.6.1.3.5(RHR I), RHR System Motor Operated Valve Operability Test
- SP 2-SR-3.5.1.6(RHR I), RHR System Rated Flow Test - Loop I, inservice test

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 0-2001-001-067, Power Supply for the A and D EECW Strainer Kinney Panels, was selected for detailed review, because the EECW 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 there were no interferences with operable systems.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation

a. Inspection Scope

The inspector observed simulator activities to assess the licensee's classification and notification of an Alert during the licensee's emergency preparedness training exercise on September 5, 2001.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY
Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas

a. Inspection Scope

The inspector reviewed licensee Radiological Control Instruction (RCI)-17, Control of High and Very High Radiation Areas, and performed plant walkdowns to verify that postings, barricades (including 31 locked doors to high radiation areas) and other controls of access to radiologically-controlled areas, including high radiation areas and extra high radiation areas, were being implemented in accordance with the procedure. In addition, the inspector reviewed the licensee's program with respect to control of keys to locked high radiation and very high radiation areas, including the key sign out log, against the requirements of 10 CFR 20.1601 and 20.1602. Also, the inspector independently measured dose rates in three posted high radiation areas to verify licensee surveys.

The inspector observed work conducted in posted high radiation areas, including support by health physics personnel who monitored radiation fields in which the work was done. Associated with that work, the inspector observed and evaluated pre-job briefings with the personnel who were scheduled to perform the tasks, which reviewed the work to be performed and included radiation work permit/as low as reasonably achievable (RWP/ALARA) discussions to review expected radiological conditions of the work area and actions to be taken in the event that those conditions changed.

The inspector reviewed licensee control and storage of highly activated materials (e.g., fuel channels and low power range monitor sources) underwater in the spent fuel storage pool (SFSP), which could be raised inadvertently to the pool surface thereby creating a high radiation area or extra high radiation area, as specified in RCI-1.1, Field Operations Program Implementation, and RCI-9.1, Radiation Work Permit Preparation and Administration. The inspector also reviewed a recent inventory of these items, and verified the presence of selected listed items in the SFSP. The inspector also reviewed selected calendar year (CY) 2001 PERs related to access control issues in the licensee's corrective action program for assignment, effectiveness of characterization, resolution/closeout timeliness, and trending. In addition, a self-assessment report, related to Assessment Number BFN-RP-01-006, conducted in May, 2001, with respect to radcon personnel knowledge and ability, was reviewed and the findings evaluated for significance and timely correction.

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety

2PS1 Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems

a. Inspection Scope

The inspectors reviewed the licensee's most recent radioactive effluent release report which delineated the quantities of radionuclides released in liquid and gaseous effluents during the CY 2000 and the radiation doses to the public resulting from those releases. The inspectors evaluated the report to determine whether it included the information and data required to be reported to demonstrate conformance with 10 CFR 20.1302, 10 CFR 50.36a, and 10 CFR 50, Appendix I. The inspectors reviewed the recent changes to Offsite Dose Calculation Manual (ODCM) and evaluated whether those changes were technically justified and consistent with the guidance provided by Regulatory Guide 1.109 and NUREG-0133. The inspectors toured the plant and assessed whether the major components of the radioactive effluent release and monitoring equipment were configured as described in Chapter 9 of the Updated Final Safety Analysis Report (UFSAR). During the tours the inspectors observed twenty two effluent monitoring instruments to evaluate their material condition and to determine whether they were in service as specified by the ODCM.

The inspectors assessed whether compensatory sampling and analyses were performed as required for four randomly selected monitors which were out-of-service at various times during the previous twelve months. The inspectors observed the collection and analysis of samples from the plant stack and assessed whether sampling and analytical procedures were followed. The inspectors reviewed the records for the most recent calibrations of six effluent monitoring instruments and one gamma spectroscopic instrument in the count room to determine whether their calibrations were current with respect to ODCM requirements. The inspectors reviewed the results of inter-laboratory comparisons made during CY 2000 and the first quarter of CY 2001 for samples typical of plant effluents and determined whether the licensee had maintained the quality of analyses consistent with the program guidance provided by Regulatory Guide 4.15. The effectiveness of characterization and resolution of selected effluent monitoring related issues identified since April 2001 were evaluated by the inspectors. The following licensee documents and procedures were examined during the inspection:

- Annual Radioactive Effluent Release Report - January through December 2000
- Offsite Dose Calculation Manual
- 0-SI-4.8.B.2-8 Airborne Effluent Analysis - Stack Noble Gas
- CI-738 Sampling Effluent Monitors for Tritium and Gamma Isotopics
- CI-702 Data Acquisition and Data Reduction
- CI-703 Sample Preparation for Gamma Ray Spectroscopy
- CI-720 Determining Vent Flow
- CI-719 Process Gaseous Permits from a Gas Sample
- 0-SI-4.2.D.1 Liquid Radwaste Monitor Calibration and Functional Test
- 0-SI-4.2.D.4 Liquid Radwaste Effluent Flowrate Calibration and Functional Test
- 3-SI-4.2.D.2 Raw Cooling Water Radiation Monitor Calibration and Functional Test
- 0-SI-4.2.K.1 Airborne Effluents - Main Stack Monitoring System Calibration
- 2-SI-4.2.K-6B1 Off-Gas Post-Treatment Radiation Monitoring Sample Flow Calibration
- 2-SI-4.2.K-6A1 Off-Gas Post-Treatment Radiation Monitoring System Calibration
- 2-SI-4.2.K-6A2 Off-Gas Post-Treatment Radiation Monitoring System Calibration
- CI-303.13 Energy Calibration and Daily Checks (Gamma Spectroscopy System)
- CI-1101 Quality Assurance/Quality Control
- CI-303.15 Efficiency Calibration (Gamma-Ray Spectrometry System)
- Audit Report No. SSA0102 - Plant Support Functional Area Audit

b. Findings

No findings of significance were identified.

2PS2 Radioactive Material Processing and Transportation

a. Inspection Scope

The inspectors evaluated the licensee’s facilities, processes and programs for the collection, processing, treatment, shipping, storage and disposal of radioactive materials and radwaste. The inspectors conducted reviews of the following: in-plant liquid and solid waste systems; waste processing and sampling program; shipment activities and records; assurance of quality, including corrective action reports; and training.

System reviews included system descriptions in Chapter 9 of the UFSAR, facilities tours, liquid waste and recycle system flow diagrams and a review of system changes in accordance with 10 CFR 50.59. The inspectors also toured radwaste equipment, and storage locations used for processed radwaste including low level waste storage modules.

The inspectors evaluated the licensee’s Process Control Program Manual (PCP) Revision 2, including: process documentation; scaling factors (derivation, sampling type, sampling frequency, and effect of changing plant conditions); and determination of waste characteristics and waste classification and the Radioactive Material Shipment Manual.

The inspectors reviewed the licensee’s 10 CFR 61 Analysis for waste characterization and scaling factors. The inspectors selected eight solid radwaste shipping records for detailed review against the requirements contained in 10 CFR Parts 20, 61, and 71, and 49 CFR Parts 100-177.

The inspectors evaluated the licensee’s program for assurance of quality in the radwaste processing and radioactive materials transportation program by reviewing a quality assurance audit, self assessments, and seven corrective action problem investigation process reports and safety assessment/screening review/safety evaluations involving the radwaste and transportation program.

The inspectors evaluated the licensee’s program for training personnel involved in the radwaste and radioactive materials transportation program with regard to the requirements contained in NRC IE Bulletin 79-19 and DOT 49 CFR, Subpart H.

The following licensee documents and procedures were examined during the inspection:

Shipping Records:

- Shipment NO. 010711
- Shipment NO. 010712
- Shipment NO. 010405
- Shipment NO. 010407
- Shipment NO. 010419
- Shipment NO. 010412
- Shipment NO. 010703
- Shipment NO. 010704

Safety Assessment/ Screening Review/Safety Evaluations:

- DCN W39816A
- ECN P0320

Corrective Action Documents:

- PER 00-006763-000
- PER 00-007091-000
- PER 00-007278-000
- PER 00-007889-000
- PER 00-009389-000
- PER 00-009618-000
- PER 00-002296-000
- PER 01-007170-000
- PER 01-007291-000

Self Assessments:

- BFN ENVR-00-003
- BFN ENVR-01-002
- Audit Report No. SSA0102- Plant Support Functional Area Audit
- Radioactive Material Shipment Manual

b. Findings

No findings of significance were identified.

2PS3 Radiological Environmental Monitoring Program (REMP)

a. Inspection Scope

The inspectors reviewed the licensee’s most recent Annual Radiological Environmental Operating Report which described implementation of the REMP during CY 2000 and provided an assessment of the program results. The review assessed whether the report included the information required to be reported regarding surveillance results, analysis of data, land use census, interlaboratory comparison program results, and permitted program deviations. The review also evaluated whether the REMP was implemented as required with respect to sampling locations, monitoring and measurement frequencies. The inspectors observed collection of air particulate filters and charcoal cartridges at four air sampling stations and collection of soil samples at four locations to determine whether the samples were collected in accordance the sampling procedures and whether good techniques were used. Calibration procedures and records for the each of the air sampling stations were reviewed to determine whether the calibrations were current. The inspectors also observed the location of eight thermoluminescence dosimeters to determine whether they were located as described in the ODCM. The inspectors reviewed the results of the interlaboratory comparisons made during CY 2000 for environmental type samples to determine whether the licensee had maintained the quality of the analyses consistent with REMP requirements.

Meteorological monitoring instrument calibration procedures and records were reviewed to determine whether instrument calibrations were current with respect to ODCM and UFSAR requirements. The inspectors assessed whether the instruments were operable and whether current meteorological conditions were available in the control room. Surveys of potentially contaminated materials being released from the RCA for unrestricted use were also observed. The inspectors assessed whether appropriate criteria were used for unrestricted release of potentially contaminated materials, whether appropriate instrumentation was used for those surveys, and whether the instruments were calibrated with appropriate sources. The inspectors reviewed REMP-related self-assessment reports to determine whether substantive issues were identified and adequately addressed. The effectiveness of characterization and resolution of selected REMP-related issues identified by the licensee were evaluated by the inspectors. Through the above reviews and observations, the licensee’s practices and implementation of the radiological monitoring program, meteorological monitoring program, and radioactive material control program were evaluated by the inspectors for consistency with the ODCM, the UFSAR, TS, and 10 CFR Part 20 requirements. The following licensee documents and procedures were examined during the inspection:

- Annual Radiological Environmental Operating Report for CY 2000
- Offsite Dose Calculation Manual
- UFSAR Chapter 2.3, 2.4, and 2.6
- Technical Requirement 3.3.7, Meteorological Monitoring Instrumentation
- Meteorological Station Calibrations for last 2 years (last calibration 6/21/01)
- Meteorological Tower Sensor Exchanges for last 2 years
- BFN Stack Quarterly Meteorological Data recovery (Graph 1987 to 2001)
- BFN Ground Level Meteorological Data recovery (Graph 1987 to 2001)
- Meteorological Measurement Site Inspection Checklist, 6/19/2001 & 5/12/2000
- Evaluation of Meteorological Monitoring Facilities at TVA Nuclear Plant Sites 1/10/2001 (QA report)

Self Assessments:

- CRP-ENV-01-002, 1/22-2/16/2001
- CRP-ENV-00-002, 10/4-11/19/1999
- CRP-RP-00-002, 6/1-30/2000
- CRP-ERMI-01-003, 4/9-27/2001

Plant Procedures:

- RCI-1.1, Field Operations Program Implementation
- SPP 5.1, Radiological Controls
- RCDP-1, Conduct of Radiological Controls
- RCDP-8, Radiological Instrumentation/ Equipment Controls
- SC-01, Collection of Environmental Monitoring Samples

Corrective Action Documents:

PER 00-011629-000
PER 00-012252-000
PER 01-000059-000
PER 01-000135-000
PER 01-006898-000

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

Licensee records were reviewed by the inspectors under the guidance of Inspection Procedure 71151 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.

.1 Unplanned Scrams per 7000 Critical Hours

a. Inspection Scope

The inspectors verified the accuracy and completeness of the licensee's PI data on unplanned scrams per 7000 critical hours on Units 2 and 3. Data reviewed included the operators' logs, the licensee's PI submittal to the NRC covering the past five quarters up to the second quarter of 2001, licensee event reports (LERs) published over the past year, and the licensee's monthly operating reports.

b. Findings

No findings of significance were identified.

.2 Safety System Functional Failures

a. Inspection Scope

The inspectors verified the accuracy and completeness of the licensee's PI data for safety system function failures on Units 2 and 3. The period covered was the third and fourth quarters of 2000 and the first and second quarters of 2001. Records reviewed included LERs, corrective action program records, the licensee's maintenance rule database, and PI data appearing on the NRC web site.

b. Findings

No findings of significance were identified.

.3 Reactor Coolant System Leakage

a. Inspection Scope

The inspectors verified the accuracy and completeness of the licensee's third and fourth quarter 2000 and first quarter 2001 Unit 2 and 3 PI data pertaining to reactor coolant system total leakage. Records reviewed included surveillance procedure 2/3-SR-2, Instrument Checks and Observations, total leakage data sheets; NEI 99-02 Revision 0, Reactor Coolant System Leakage Performance Indicator portion; and PI data for the specified period from the NRC web site. In addition to record reviews, the inspector observed activities associated with determining the RCS leak rates in conjunction with Section 1R22.

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up

.1 Units ½ C and D EDGs Rendered Inoperable

a. Inspection Scope

On July 10, 2001, the licensee discovered the fire dampers on the Unit ½ C and D EDG room air intake plenums in the closed position. Operators immediately declared the EDGs inoperable. The EDGs were returned to an operable status within the 2-hour LCO completion time. The licensee determined that preventive maintenance was performed on the fire damper actuators approximately two days prior to the discovery of the closed dampers. Independent verification and routine operator checks of the dampers confirmed that the dampers were left in the open position following the maintenance. The dampers may have inadvertently closed earlier on July 10, when an operator started the room ventilation fans; however the licensee could not confirm this as the actual closure time. The inspectors reviewed the event to determine whether additional inspection response was warranted.

This event was determined to be of very low safety significance because the duration of the loss of function of the two EDGs was less than two days and the other Unit ½ EDGs (A and B) remained operable during the event. Therefore, no additional inspection response was warranted.

b. Issues and Findings

No findings of significance were identified.

.2 Unit 2 SCRAM Due to Electro-Hydraulic Control System

a. Inspection Scope

On July 25, 2001, Unit 2 automatically scrammed while testing was performed on the electro-hydraulic control (EHC) system. The inspectors responded to the control room to observe licensee activities and verify that the reactor was shutdown and in a safe condition. The inspectors also reviewed the licensee's scram report which is produced from Attachment 1 of Procedure 2-AOI-100-1, Reactor Scram. The inspectors also reviewed post scram data using graphs of key plant parameters and discussed some items with engineering personnel.

b. Issues and Findings

No findings of significance were identified.

.3 Fire Protection Inspection Followup

(Closed) LER 50-260/2000-002-00: Failure to Meet Appendix R Criteria Resulting in a Condition not Covered by Plant Operating Procedures. See Section 4OA5 for a detailed discussion and disposition of a previously identified NRC finding concerning this issue. This LER is closed.

4OA5 Other

.1 Unit 1 Lay-up and Equipment Preservation Program Inspection

a. Inspection Scope

The inspectors utilized the applicable guidance in Inspection Procedure 92050 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. Although this equipment was not currently performing safety-significant functions, this inspection provided a periodic quality status of Unit 1 equipment. The inspectors reviewed the licensee's overall application of the Unit 1 equipment preservation process with emphasis on preservation equipment condition and operation, compared with the requirements of Procedure 0-TI-373. In addition, the inspectors reviewed the preventive maintenance program for the HPCI system, and the water chemistry history of the suppression pool, spent fuel pool, and reactor coolant system.

The inspectors also reviewed the results of a self-assessment of the Unit 1 Layup Program conducted by the Chemistry Department on July 12, 2001. The inspectors reviewed the assessment findings requiring corrective action and documented in PERs 01-006965-000 and 01-007017-000. The inspectors also reviewed the licensee's lay-up punch list and outstanding PERs to gain insight over what deficiencies were being identified and corrected. The inspectors conducted a walkdown inspection of the risk significant systems with emphasis on a sampling of Core Spray, HPCI, and RHR Loop I.

Unit 1 SSCs shared by Unit 2 and Unit 3 have been inspected under the normal ROP for operating reactors.

b. Findings

No findings of significance were identified.

.2 Fire Protection Inspection Followup

(Closed) Unresolved Item (URI) 50-260,296/00-08-01: Determination of the Risk Significance of Dead-Heading the RHR Pump. During the triennial fire protection inspection (NRC Inspection Report 50-259,260,296/00-08, dated August 8, 2000), the NRC identified an issue involving the potential unintended closure of residual heat removal (RHR) pump minimum flow control valves due to fire damage to control cables. After the inspection, the licensee submitted LER 50-260/2000-002-00, Failure to Meet Appendix R Criteria Resulting in a Condition not Covered by Plant Operating Procedures. This LER reported the problem with closure of the RHR minimum flow control valves, and stated it was a non-compliance with 10 CFR 50, Appendix R, Criterion III.L.2.b and a condition not covered by the plant's operating and emergency procedures. The NRC agreed with that conclusion. Criterion III.L.2.b stated that, when using "alternative" shutdown in response to a fire, one performance goal for the shutdown functions is that the reactor coolant makeup function shall be capable of maintaining the reactor coolant level above the top of the core.

The issue had a credible impact on safety because closure of the minimum flow control valves could cause the RHR pumps to run dead-headed in the early stages of several fire scenarios, which could lead to pump failure. The inspectors also noted that, reliance on RHR appeared to be an important part of the licensee's safe shutdown analysis. Hence the issue impacted the ability to safely shutdown the plant for the credible initiating event of fire due to loss of function of a mitigating system, specifically the RHR system.

The NRC determined that the issue was of very low safety significance (Green). Some of the factors causing the issue to be of very low safety significance were:

- Automatic and manual suppression systems were available which reduced the estimated likelihood rating
- Due to redundancy in the RHR system two cables would have to be damaged in a particular manner to cause loss of function
- The plant power conversion system, which was not included in the safe shutdown analysis, remained available through operator action to remove core heat

Because the licensee has included this item in the corrective action program (PER-00-006682-000), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy, and is identified as NCV 50-260,296/01-03-02, Error in Analysis of RHR System Results in Failure to Meet 10 CFR 50, Appendix R, Criterion III.L.2.b. The URI is closed.

4OA6 Management Meetings

The inspectors presented the inspection results to Mr. Ashok Bhatnagar and Mr. R. G. Jones, and other members of licensee management during exit meetings on July 13, July 27, September 14, September 26, October 12, and October 22, 2001. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

4OA7 Licensee Identified Violations

The following finding of very low safety significance was identified by the licensee, is a violation of NRC requirements, and meets the criteria of Section VI.A.1 of the NRC's Enforcement Policy, NUREG-1600, for being treated as an NCV.

NCV Tracking Number	Requirement Licensee Failed to Meet
NCV 50-260/01-03-03	TS 5.4.1.a requires that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33 requires procedures for performing maintenance that can affect the performance of safety-related equipment. Fire Protection Instruction FP-0-039-PM001, Preventive Maintenance for Horizontal CO_2 Activated Fire Dampers in Diesel Generator Buildings, Revision 8, contained maintenance procedures for the CO_2 actuated dampers supplying cooling air to the emergency diesel generators. The procedure did not contain adequate checks for damper chain alignment and blow off clip engagement in accordance with vendor recommendations necessary to assure proper operation including preventing inadvertent actuation. This finding had a credible impact on safety because, on July 10, 2001, approximately two days after performance of the maintenance procedure, the Unit ½ C and D EDG room air intake fire dampers were found in the closed position, thereby rendering both of the associated EDGs inoperable. This issue was determined to be of very low safety significance (Green) because the duration of the loss of function of the two EDGs was less than two days and the other Unit ½ EDGs (A and B) remained operable. This problem was entered into the licensee's corrective action program as PER 01-006911-000.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

T. Abney, Licensing Manager
A. Bhatnagar, Site Vice President
L. Clardy, Site Quality Assurance Manager
J. Corey, Radiation Protection and Chemistry Manager
T. Cornelius, Emergency Preparedness Supervisor
W. Hargrove, Radcon Supervisor
R. Jones, Plant Manager
G. Little, Operations Manager
T. Niessen, Jr., Site Support Manager
D. Sanchez, Maintenance and Modifications Manager
M. Scaggs, Assistant Plant Manager
T. Trask, Acting Site Engineering Manager

NRC

R. Bernhard, Region II Senior Reactor Analyst

LIST OF ITEMS OPENED AND CLOSED

Opened and Closed

50-260,296/01-03-01	NCV	Failure to Meet 10 CFR 50.59 Requirements (Section 1R20)
50-260,296/01-03-02	NCV	Error in Analysis of RHR System Results in Failure to Meet Appendix R, Criterion III.L.2.b (Section 4OA5.2)
50-260/01-03-03	NCV	Inadequate Fire Damper Maintenance Procedure Results in Inoperable Emergency Diesel Generators (Section 4OA7)

Closed

50-260,296/00-08-01	URI	Determination of the Risk Significance of Dead-Heading the RHR Pump (Section 4OA5.2)
50-260/2000-002-00	LER	Failure to Meet Appendix R Criteria Resulting in a Condition not Covered by Plant Operating Procedures (Section 4OA3.3)