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

October 27, 2005

EA-05-196

Tennessee Valley Authority
ATTN.: Mr. K. W. Singer
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 05000259/2005004, 05000260/2005004, AND 05000296/2005004 AND EXERCISE OF ENFORCEMENT DISCRETION

Dear Mr. Singer:

On September 30, 2005, the US Nuclear Regulatory Commission (NRC) completed an inspection at your operating Browns Ferry Unit 2 and 3 reactor facilities. The enclosed integrated quarterly inspection report documents the inspection results, which were discussed on October 6, 2005, with Mr. B. O'Grady, and other members of your staff.

Also, please note that, per our letter to you on December 29, 2004, Browns Ferry Unit 1 inspections in the Reactor Oversight Program (ROP) cornerstones of Occupational Radiation Safety, Public Radiation Safety, Emergency Preparedness, and Physical Protection were incorporated into the routine ROP baseline inspection program effective January 1, 2005. Although this report period did not contain any site inspection in those cornerstones, they will continue to be documented in ROP integrated quarterly reports such as this one. Results from our inspection of your Unit 1 Recovery Project in the remaining cornerstones will continue to be documented in a separate Unit 1 integrated inspection report.

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.

This report also provides details of a violation regarding an inoperable primary containment isolation valve in excess of Technical Specification allowable outage time. As discussed in the NRC's Enforcement Policy, the NRC may refrain from issuing enforcement action for violations resulting from matters not within the licensee's control, such as equipment failures that were not avoidable by reasonable licensee quality assurance measures or management controls. Based on the circumstances of this violation, the NRC considers it appropriate to exercise enforcement discretion in accordance with Section VII.B.6 of the Enforcement Policy.

In accordance with 10 CFR 2.390 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/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Stephen J. Cahill, 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: Inspection Report 05000259/2005004, 05000260/2005004 and 05000296/2005004
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:	05000259/2005-004, 05000260/2005-004, 05000296/2005-004
Licensee:	Tennessee Valley Authority (TVA)
Facility:	Browns Ferry Nuclear Plant, Units 1, 2, and 3
Location:	Corner of Shaw and Nuclear Plant Roads Athens, AL 35611
Dates:	July 1, 2005 - September 30, 2005
Inspectors:	T. Ross, Senior Resident Inspector R. Monk, Resident Inspector E. Christnot, Resident Inspector B. Holbrook, NRC Contractor
Approved by:	Stephen J. Cahill, Chief Reactor Project Branch 6 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000259/2005-004, 05000260/2005-004, 05000296/2005-004; 07/01/2005 - 09/30/2005; Browns Ferry Nuclear Plant, Units 1, 2, and 3; resident inspector integrated report.

The report covered a three-month period of routine inspections by the resident inspectors. No findings of significance were identified by the NRC or the licensee during the period. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

None.

B. Licensee Identified Findings

None.

Report Details

Summary of Plant Status

Unit 1 was defueled and in a recovery status for the entire report period.

Unit 2 operated at essentially full power for the entire report period, expect for two noteworthy events. On July 30, 2005, unit power was reduced to 20% and the main turbine-generator was taken off line for approximately eight hours to effect planned repairs to the main generator neutral overvoltage protection system. Then on August 5, Unit 2 experienced an automatic reactor trip from 100% power due to low reactor vessel level caused by an unexpected mechanical failure of the 2C reactor feedwater pump (RFP), closely followed by a spurious trip of the 2B RFP. The unit was returned to full power on August 8.

Unit 3 operated at essentially full power for the entire report period, expect for planned downpowers of limited duration and a reactor trip. On September 17, 2005, Unit 3 experienced an automatic reactor trip from 74% power due to a main turbine trip caused by a loss of main condenser vacuum during maintenance on a main feedwater heater level control valve. The unit was restarted on September 18, and returned to full power on September 23.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope

On August 29, prior to the onset of the tropical storm remnant from Hurricane Katrina, the site was notified that a Tornado Watch was in affect. The inspectors verified that the licensee entered Abnormal -Operating Instruction (AOI) 0-AOI-100-7, Tornado. The inspectors also discussed severe weather preparations with site management and conducted comprehensive tours of vulnerable areas outside and around the power block, switchyard, etc., to verify that the licensee had made appropriate preparations per the AOI. Furthermore, the inspectors attended status meetings conducted by personnel responsible for implementing severe weather preparations onsite.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

.1 Partial Walkdown

a. Inspection Scope

The inspectors performed partial walkdowns of the three safety systems listed below to verify train operability as required by the plant Technical Specifications (TSs), while the other redundant or diverse trains were out of service. These inspections included reviews of applicable TSs, plant lineup procedures, operating procedures, and/or piping and instrumentation drawings (P&IDs) which were compared with observed equipment configurations to identify any discrepancies that could affect operability of the redundant train or backup system. The systems selected for walkdown were also chosen due to their relative risk significance from a Probabilistic Safety Assessment (PSA) perspective for the existing plant equipment configuration. The inspectors verified that selected breaker, valve position, and support equipment were in the correct position for system operation.

- Unit 3 Reactor Core Isolation Cooling (RCIC) System While Unit 3 High Pressure Coolant Injection (HPCI) System Was Out-of-Service (OOS) per 3-OI-71 Attachments 1, 2, 3 and 4 and drawing 3-47E813 (7/20/2005)
- Unit 2 Core Spray Div II per 2-OI-75 Attachments 1, 2, 3, and 4 and Drawing 2-47E814
- Unit 3 RCIC While Unit 3 HPCI OOS per 3-OI-71 Attachments 1, 2, 3, and 4 and drawing 3-47E813 (9/15/2005)

b. Findings

No findings of significance were identified.

.2 Complete Walkdown

a. Inspection Scope

The inspectors completed a detailed alignment verification of the Unit 2 Residual Heat Removal (RHR) System Loop II using P&ID 2-47E811, and applicable training guides to walkdown and verify equipment alignment and operability. The inspectors reviewed relevant portions of the Updated Final Safety Analysis Report (UFSAR) and TSs. This detailed walkdown also verified electrical power alignment, the condition of applicable system instrumentation and controls, component labeling, pipe hangers and support installation, and associated support systems status. Furthermore, the inspector examined TS action statements, OOS and Operator Work Around (OWA) lists; control room operator logs; active open work orders (WOs); System Health Reports; and any PERs that could affect system alignment and operability.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors reviewed licensee procedures, Standard Program and Process (SPP) -10.10, Control of Transient Combustibles, and SPP-10.9, Control of Fire Protection Impairments, and conducted a walkdown of the seven fire areas (FA) and one hot work listed below. Selected fire areas/zones, and/or hot work activities, were examined in order to verify licensee control of transient combustibles and ignition sources; the material condition of fire protection equipment and fire barriers; and, operational lineup and operational condition of fire protection equipment or measures. Also, the inspectors verified that selected fire protection impairments were identified and controlled in accordance with procedure SPP-10.9. Furthermore, the inspectors reviewed applicable portions of the Site Fire Hazards Analysis (FHA), Volume 1 and 2 and Pre-fire Plan drawings to verify that the necessary fire fighting equipment, such as fire extinguishers, hose stations, ladders, and communications equipment, were in place.

- Unit 2 Reactor Building Elevations 593' & 621'
- Unit 3 Diesel Bldg. (FA 21)
- Unit 2 Battery and Board Rooms (FA-18)
- Welding on Refueling Floor (Hot Work Permit 05-718306)
- Unit 3A Electric Board Room (FA-13)
- Unit 2 Aux Instrument Room (FA 18)
- Reactor Building 4-kV Shutdown Board C (FA-9)
- Reactor Building 4-kV Shutdown Board D (FA-8)

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope

The inspectors performed a review of the Unit 2 RHR and Containment Spray (CS) pump rooms, and Under-Torus area, for internal flood protection measures. The inspectors reviewed plant design features and licensee procedures intended to protect the plant and its safety-related equipment from internal flooding events. The inspectors reviewed flood analysis documents including: UFSAR; Design Criteria BFN-50-C-7105, Internal Flooding Design Basis; Emergency Operating Instruction 3, Secondary Containment Control; Browns Ferry Unit 2 Individual Plant Examination, Browns Ferry Internal Floods Analysis; and the Browns Ferry Nuclear Plant Probabilistic Safety

Assessment Initiating Event Notebook, Initiating Event Frequencies, for licensee commitments. The inspectors also discussed possible flood events, flooding scenarios, and mitigating strategies with licensee personnel knowledgeable about flood protection measures and plans.

The inspectors performed walkdowns of risk-significant areas, susceptible systems and equipment, including the Unit 2 RHR and CS pump rooms and Under-Torus area to review flood-significant features such as area level switches, room sumps and sump pumps, flood protection door seals, conduit seals and instrument racks that might be subjected to flood conditions. Plant procedures for mitigating flooding events were also reviewed to verify that licensee actions were consistent with the plant's design basis assumptions.

The inspectors also reviewed a sampling of the licensee's corrective action documents with respect to flood-related items to verify that problems were being identified and corrected. Furthermore, the inspectors reviewed selected completed preventive maintenance procedures, work orders, and surveillance procedures to verify that actions were completed within the specified frequency and in accordance with design basis documents.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

On September 26, the inspector observed operator crew conduct during an as-found simulator evaluation per Simulator Exercise Guide OPL 177.058 to verify that crew performance was in accordance with licensee procedures and regulatory requirements.

The inspector specifically evaluated the following attributes related to the operating crew's performance:

- Clarity and formality of communication
- Ability to take timely action to safely control the unit
- Prioritization, interpretation, and verification of alarms
- Correct use and implementation of Abnormal Operating Instructions, Emergency Operating Instructions and Operational Contingencies
- Timely and appropriate Emergency Action Level Declarations per Emergency Plan Implementing Procedures
- Control board operation and manipulation, including high-risk operator actions
- Command and Control provided by the Unit Supervisor and Shift Manager

The inspector also attended the post-exercise critique to assess the effectiveness, and verify that licensee-identified issues were comparable to issues identified by the inspector.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed the two systems listed below with regard to some or all of the following attributes: (1) work practices; (2) identifying and addressing common cause failures; (3) scoping in accordance with 10 CFR 50.65(b) of the maintenance rule (MR); (4) characterizing reliability issues for performance; (5) trending key parameters for condition monitoring; (6) charging unavailability for performance; (7) appropriateness of performance criteria in accordance with 10 CFR 50.65(a)(2), (8) system classification in accordance with 10 CFR 50.65(a)(1); and (9) appropriateness and adequacy of goals and corrective actions (i.e., Ten Point Plan) for structures, systems, and components (SSCs)/functions classified as (a)(1). The inspectors also compared the licensee's performance against site procedure SPP-6.6, Maintenance Rule Performance Indicator Monitoring, Trending and Reporting; Technical Instruction 0-TI-346, Maintenance Rule Performance Indicator Monitoring, Trending and Reporting; 0-TI-362, Inservice Testing of Pumps and Valves; and SPP 3.1, Corrective Action Program. The inspectors also reviewed applicable work orders, engineering evaluations and system testing to verify that regulatory and procedural requirements were met.

- Unit 2 RHR Valve 2-FCV-74-57 (Containment Isolation Valve)
- 2A RHR Heat Exchanger and Pump Train (Excessive Unavailability)

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control

j. Inspection Scope

For the seven risk and emergent work assessments listed below, the inspectors reviewed licensee actions taken to plan and control the work activities to effectively manage and minimize risk. The inspectors verified that risk assessments were being performed as required by 10 CFR 50.65(a)(4), and applicable procedures, such as SPP-6.1, Work Order Process Initiation; SPP-7.1, Work Control Process; and 0-TI-367, BFN Dual Unit Maintenance. The inspectors also evaluated the adequacy of the licensee's risk assessments and the implementation of compensatory measures.

- Shutdown Board A Battery OOS
- Unit 2 Division 1 Core Spray System, Essential Equipment Cooling Water (EECW) Pump D3, Reactor Water Cleanup Up (RWCU) Pump 3A, and RHR Service Water (SW) Pump A1 OOS
- Emergency Diesel Generator A, EECW Pump D3, and RHRSW Pump A1 OOS
- RHRSW A2 and D1 Pumps, Unit 1 Main Bank Transformer, 3A Control Rod Drive (CRD) Pump, and EDG D OOS
- EDG 3C, RHRSW Pump D1, Common Service Station Transformer (CSST) A, 1B 480V Shutdown Board, Unit 1 Main Bank Transformer, and Switchyard Bus 1 OOS
- Emergent Work Related to Leak on 2B and 2D RHRSW Piping
- Emergent Work Related to Leak on 3C EDG

b. Findings

No findings of significance were identified.

1R14 Operator Performance During Non-Routine Evolutions and Events

a. Inspection Scope

Unit 2 Restart Following Reactor Scram

Following the Unit 2 reactor trip on August 5 (see section 4OA3.3), the inspectors witnessed significant portions of the Unit 2 restart, including the approach to criticality and power ascension, in accordance with 2-GOI-100-1A, Unit Startup and Power Operation. The inspectors evaluated operator performance through interviews, observations, and examining available information (e.g., operator logs, plant computer data, strip charts, etc.). The inspectors also examined a non-safety significant human performance error and reactivity event that involved an out of sequence rod withdrawal during the approach to criticality as documented by PER 87188. Furthermore, the inspectors discussed the results of the subsequent human performance investigation with Operations Management, and reviewed the immediate and proposed long-term corrective actions.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

Routine Baseline Review

a. Inspection Scope

The inspectors reviewed seven operability/functional evaluations listed below to verify technical adequacy and ensure that the licensee had adequately assessed TS operability. The inspectors reviewed appropriate sections of the UFSAR to verify that the system or component remained available to perform its intended function. In addition, where applicable, the inspectors reviewed licensee procedure SPP-3.1, Corrective Action Program, Appendix D, Guidelines For Degraded/Non-conforming Condition Evaluation and Resolution of Degraded/Non-conforming Conditions, to ensure that the licensee's evaluation met procedure requirements. Furthermore, where applicable, inspectors reviewed implemented compensatory measures to verify that they worked as stated and that the measures were adequately controlled. The inspectors also reviewed PERs daily to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations.

- PER 83178, RHRSW System Check Valve 0-CKV-023-0594 Failure To Close
- PER 87661, Containment Atmosphere Dilution System Tank B Pressure Regulator Leak
- PER 87900, 2B & 2D RHRSW Supply Piping Integrity Per Code Case 513
- PER 88490, Unit 2 Recirculation Piping Snubber 068-5009 Missed Inspection
- PER 88802, 161-kV Capacitor Battery Bank With Three Inoperable Cells
- PER 95636, Unit 3 Pipe Support R-38 Inaccurate Analysis
- PER 85316, Shutdown Board Battery A Individual Cell Voltage Degradation

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (OWAs)

a. Inspection Scope

The inspectors performed a semi-annual evaluation of the potential cumulative effects of all outstanding Unit 1 and 2 OWAs regarding: (1) Reliability, availability, and potential for mis-operation of a system; (2) Increasing an initiating event frequency or affecting multiple mitigating systems; and (3) Ability of the operators to respond in a correct and timely manner to plant transients and accidents. The inspectors reviewed the current list of OWAs as defined by Operations Department Procedure (OPDP)-1, Conduct of Operations, and Section 4.11 of the Operations Directive Manual. The inspectors also attended plan of the day meetings in which the priority and operational impact of all OWAs were discussed.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (PMT)

a. Inspection Scope

The inspectors witnessed and/or reviewed documentation of post-maintenance test (PMT) activities of the five risk-significant SSCs listed below, to verify that system operability and functional capability following completion of associated work was adequately demonstrated. For each of these PMTs, some or all of the following aspects were inspected: (1) effect of testing on the plant was recognized and addressed by control room, maintenance and/or engineering personnel; (2) testing was consistent with maintenance performed; (3) acceptance criteria demonstrated operational readiness consistent with design and licensing basis documents such as TSs, UFSAR, and others; (4) range, accuracy and calibration of test equipment; (5) step-by-step compliance with test procedures, and applicable prerequisites satisfied; (6) control of installed jumpers or lifted leads; (7) removal of test equipment; and, (8) restoration of SSCs to operable status. The inspectors also verified that PMT activities were conducted in accordance with applicable procedural requirements, including SPP-6.3, Post-Maintenance Testing, and MMDP-1, Maintenance Management System. Furthermore, the inspectors also reviewed problems associated with PMTs that were identified and entered into the corrective action program.

- Reactor Building Crane PMT for replacement of a control relay per 0-SI-4.10.D, Reactor Building Crane
- Unit 2 PMT for 2-FCV-71-17, RCIC Suppression Chamber Inboard MOV per 2-SR-3.6.1.3.5 (RCIC), Unit 2 RCIC System MOV Operability Test
- Unit 2 PMT for 2-FCV-75-58, Core Spray Drain Pump A Bypass Line per 2-SR-3.6.1.3.5 (CS I), Unit 2 Core Spray System MOV Operability Test
- PMT for BFN-0-RLY-082-A/PFDA2, A EDG Aux Relay per EPI-0-082-DGZ003, Diesel Generator A Redundant Start Test.
- D3 EECW Motor replacement PMT per 3-SI-4.5.C.1 (2-Comp), EECW Comprehensive Pump Test

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors either witnessed portions of surveillance tests or reviewed test data for the five risk-significant and/or safety-related systems listed below to verify that the tests met TS surveillance requirements, UFSAR commitments, and in-service testing (IST) and licensee procedure requirements. The inspectors' review confirmed whether the testing effectively demonstrated that the SSCs were operationally capable of performing their intended safety functions and fulfilled the intent of the associated surveillance requirement. Applicable IST data was compared against the requirements of licensee procedures 0-TI-362, Inservice Testing of Pumps and Valves; and 0-TI-230, Vibration Monitoring and Diagnostics.

- 2-SR-3.4.5.2, Drywell Leak Detection Radiation Monitor Functional Test 2-RM-90-256**
- 3-SR-3.5.1.7, HPCI Main and Booster Pump Set Developed Head and Flow Rate Test at Rated Reactor Pressure*
- O-SI-4.11.B.3.b, Quarterly Check for Diesel Fire Pump Batteries 1 and 2
- 3-SR-3.5.1.6, Quarterly RHR System Rated Flow Test Division 1
- 2-SR-3.3.6.1.3, RCIC Steam Line Spaces High Temperature Calibration and Functional Test

*This procedure included inservice testing requirements.
**This procedure included a leak detection system surveillance.

b. Findings

No findings of significance were identified

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed licensee procedures 0-TI-405, Plant Modifications and Design Change Control; 0-TI-410, Design Change Control; SPP-9.5, Temporary Alterations; and the two temporary modifications listed below to ensure that procedure and regulatory requirements were met. The inspectors reviewed the associated 10 CFR 50.59 screening and evaluation, and applicable system design bases documentation. The inspectors reviewed selected completed work activities and walked down portions of the systems to verify that installation was consistent with the modification documents and Temporary Alteration Control Form (TACF).

- TACF 0-05-004-248, A Shutdown Battery Cell Jumpers
- TACF 2-05-009-003, Bypass of 2B Reactor Feedwater Pump Turbine Thrust Bearing Wear Detector Trip

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

Residual Heat Removal Unavailability

a. Inspection Scope

By the end of the second quarter of 2005, Unit 2 RHR system unavailability for the previous 36 months was only about 3.5 hours below the White PI threshold (i.e., 1.5% unavailability). Due to this slim margin to a color change, NRC regional management determined that an independent review of the licensee's PI determination by the resident staff would be a prudent measure.

This inspection was conducted in accordance with NRC Inspection Procedure 71151, Performance Indicator Verification, in conjunction with NEI 99-02, Revision 3, Regulatory Assessment Performance Indicator Guidelines, and SPP-3.4, Performance Indicator for NRC Reactor Oversight Process. The inspector specifically reviewed the raw data used as input to the Unit 2 RHR Unavailability PI from the third quarter 2002 through the third quarter 2005. The inspector also compared graphical representations from the most recent PI report to the raw data to verify that the data was correctly reflected in the report. Furthermore, the inspector reviewed relevant PERs (e.g., PER 89036) and engineering evaluations; selected control room operator chronological logs; and applicable licensee records to verify whether the PI data was appropriately captured for inclusion into the PI report, and that the PI was calculated correctly.

b. Findings

No findings of significance were identified.

4OA2 Identification & Resolution of Problems

Routine Review of Problem Evaluation Reports

a. Inspection Scope

The inspectors performed a daily screening of all PERs entered into the licensee's corrective action program. The inspectors followed NRC Inspection Procedure 71152, Identification and Resolution of Problems, in order to help identify repetitive equipment failures or specific human performance issues for follow-up.

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up

.1 (Closed) Licensee Event Report (LER) 05000260/2005-002-00, Primary Containment Isolation Valve Inoperable in Excess of Technical Specification Allowable Outage Time

On February 10, 2005, the outboard primary containment isolation valve (2-FCV-69-02) of the Unit 2 RWCU System was discovered to be damaged and determined to be inoperable during Mode 1 operation. The inspector reviewed the LER for accuracy, completeness, and adequacy of corrective actions. The inspector also reviewed the associated PER 76546 for more details regarding the cause of the event and extent of condition. The damage to 2-FCV-69-02 was caused when the packing follower became bound to the valve stem as the valve was being opened. As the valve stem moved outward, the packing follower bolts broke, allowing the follower to be pulled out of the valve body. The most probable root cause of this failure was a gradual relaxation of the packing follower bolt torque, which allowed the follower to slightly cock as the stem moved outward, until it galled and bound on the stem. The last outward stroke occurred February 2, 2005, but the apparent failure was undetectable due to the location of the valve. Therefore, TS limiting condition of operation (LCO) actions were not completed in a timely manner. The licensee discovered that 2-FCV-69-02 was damaged a week later upon entry into the area for other system related maintenance activities. Subsequent examination determined that the damaged valve would not have been be able to close to perform its containment isolation safety function. The inspector determined that upon discovery of the inoperable containment isolation valve, the licensee took the necessary prompt actions to meet the applicable actions statements for the TS 3.6.1.3 LCO.

The inoperable outboard containment isolation valve was assessed by the inspectors using the Significance Determination Process (SDP). Because 2-FCV-69-02 was inoperable while the inboard valve (2-FCV 69-01) was open for more than 4 hours, this was a violation of TS 3.6.1.3, Action A.1, for Modes 1-3. This violation was considered greater than minor because if left uncorrected it would decrease the defense-in-depth capability to reliably isolate a critical interfacing system located outside containment. Furthermore, the violation directly affects the containment isolation reliability attribute of the Barrier Integrity cornerstone objective. This violation was evaluated per Phase 1 of the SDP and was determined to be a finding of very low safety significance because the RWCU system itself is a closed system rated for full RCS pressure, and the inboard RWCU containment isolation valve was verified to be fully operational.

After further review, the inspectors concluded that the failure mechanism of the RWCU isolation valve was not attributable to inadequate or improper maintenance. Also, the maintenance history of 2-FCV-69-02, the history of similar designed valves installed in both operating units, and industry operating experience provided no indication that this failure mode was probable. Therefore, the inspectors determined that the valve failure did not constitute a licensee performance deficiency. Consequently, as discussed in the NRC's Enforcement Policy, the NRC may refrain from issuing enforcement action for violations resulting from matters not within the licensee's control, such as equipment failures that were not avoidable by reasonable licensee quality assurance measures or management controls. Based on the circumstances of this violation, the NRC considers it appropriate to exercise enforcement discretion in accordance with Section VII.B.6 of the Enforcement Policy and refrain from issuing enforcement action for this violation. This LER is closed.

.2 (Closed) LER 05000260/2005-001-00, Loss of High Pressure Coolant Injection (HPCI) Discharge Piping Keep-Fill

On January 20, 2005, the Unit 2 HPCI pump suction unexpectedly transferred itself automatically from its normally aligned water source (i.e., Condensate Storage Tank) to the Suppression Pool. The cause was determined to be a shorted wire in a Condensate Storage Tank level switch. Coincidently, at the same time the pump suction realigned, the HPCI inboard discharge isolation valve was in the open position, with power removed, due to ongoing testing activities on the associated 250-VDC electrical system. For about five hours, until the HPCI suction path was realigned, the licensee was unable to assure that the discharge piping remained filled due to potential system voiding. Based on this uncertainty, the licensee declared the system inoperable for this period of time and considered it reportable as a safety system functional failure. After the suction path was realigned the licensee subsequently verified that the system was adequately filled and vented within about 12 hours. No abnormal indication of voiding or entrained air was identified. The Unit 2 RCIC system was operable throughout this event. Pursuant to TS LCO 3.5.1, Action C, HPCI is allowed to be inoperable for up to 14 days as long as RCIC is operable. This event did not involve a violation of TS or licensee procedures. The LER, and the associated PER 75274, were reviewed by the inspectors and no findings of significance were identified. This LER is closed.

.3 Unit 2 Reactor Trip

a. Inspection Scope

On August 5, 2005, Unit 2 experienced an automatic trip of from 100% power due to low reactor water level (Level 3). This trip was caused by a sudden loss of the turbine-driven 2C RFP due to a mechanical failure of the steam admission control valve linkage, followed almost immediately by a trip of the 2B RFP due to a spurious high thrust bearing pressure signal. An inspector promptly responded to the control room and verified that the unit was stable in Mode 3 (Hot Shutdown), and confirmed that all safety-related mitigating systems had operated properly. The inspector evaluated safety equipment and operator performance before and after the event by examining existing plant parameters, strip charts, plant computer historical data displays, operator logs, the alarm typewriter Sequence of Events printout, and the critical parameter trend charts in the post-trip report. The inspector also discussed the event with onshift Operations personnel, reviewed 2-AOI-100-1, Reactor Scram, and written notifications made in accordance with 10 CFR 50.72. The inspector discussed the preliminary cause of the trip with licensee management, and Operations and Maintenance personnel. The inspector also reviewed the licensee’s post-trip review report, restart checklist, and Plant Oversight Review Committee minutes for approving restart.

b. Findings

No findings of significance were identified during the initial event followup. However, the inspectors are still following up on the licensee's root cause determination and corrective actions which will be addressed as an integral part of the LER closeout.

.4 Unit 3 Reactor Trip

a. Inspection Scope

On September 17, 2005, Unit 3 experienced an automatic reactor trip from 74% power due to a loss of main condenser vacuum caused during maintenance on the 3B1 moisture separator level control valve (2-LCV-006-0073A). The inspector responded from offsite and immediately met with Operations and Maintenance personnel to discuss the preliminary cause of the trip. The inspector then walked down the control room to examine existing plant parameters, conditions, and system alignments to verify that the unit was stable in Mode 3 (Hot Shutdown), and to confirm that all safety-related mitigating systems operated properly. The inspector also conducted an evaluation of safety equipment and operator performance before and after the event by interviewing the operating crew; and by examining applicable strip charts, plant computer historical data displays, operator logs, alarm typewriter Sequence of Events printout, and the critical parameter trend charts in the post-trip report. Furthermore, the inspector reviewed operator compliance in accordance with 3-AOI-47-3, Loss of Condenser Vacuum, and 3-AOI-100-1, Reactor Scram; and reviewed the written notifications made in accordance with 10 CFR 50.72. Lastly, the inspector discussed operator performance during the event with the Operations Manager, and verified that appropriate PERs were initiated. The inspector also attended a Unit 3 Scram Recovery Meeting and discussed the recovery schedule and restart items with licensee management.

b. Findings

No findings of significance were identified during the initial event followup. However, the inspectors are still following up on the licensee's root cause determination and corrective actions which will be addressed as an integral part of the LER closeout.

4OA5 Other

Operational Readiness of Offsite Power (Temporary Instruction (TI) 2515/163)

Completion of this TI was documented in NRC Inspection Report 50-259, 260, and 296/2005-003. However, after an NRC headquarters review of the data provided, additional information related to the TI was requested. The inspectors collected this information from licensee discussions, site procedures, and licensee documentation. The information was subsequently provided to the headquarters staff for further analysis.

4OA6 Management Meetings

Exit Meeting Summary

On October 6, 2005, the resident inspectors presented the integrated inspection results to the Site Vice President, Mr. Brian O'Grady, and other members of his staff, who acknowledged the findings. The inspectors confirmed that proprietary information was not provided or examined during the inspection period.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION
PARTIAL LIST OF PERSONS CONTACTED

Licensee

B. Aukland, Nuclear Plant Manager
W. Crouch, Nuclear Site Licensing & Industry Affairs Manager
R. DeLong, Site Engineering Manager
A. Elms, Nuclear Plant Operations Manager
A. Feltman, Emergency Preparedness Supervisor
R. Jones, Unit 1 Restart Manager
J. Kennedy, Human Performance Improvement Manager
R. Kerwin, Acting Site Nuclear Assurance Manager
R. Marks, Site Support Manager
R. Marsh, Operations Superintendent
M. Mitchell, Radiation Operations Manager
J. Mitchell, Site Security
D. Nye, Maintenance & Modifications Manager
B. O'Grady, Site Vice President
C. Ottenfeld, Radiation Protection Manager
J. Steele, Daily Scheduling Manager
K. Welch, Systems Engineering Manager

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

None

Closed

05000260/2005-001-00	LER	Loss of High Pressure Coolant Injection Discharge Piping Keep-Fill (Section 4OA3.2)
05000260/2005-002-00	LER	Primary Containment Isolation Valve Inoperable in Excess of Technical Specification Allowable Outage Time (Section 4OA3.1)

Discussed

None