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

October 30, 2006

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 05000259/2006004, 05000260/2006004, 05000296/2006004, AND 07200052/2006001

Dear Mr. Singer:

On September 30, 2006, the United States 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 5, 2006, with Brian O'Grady and other members of your staff.

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

Additionally, the enclosed report also documents some inspection of Unit 1 that was performed per our letter to you on December 29, 2004, regarding the transition of Unit 1 into the Reactor Oversight Program (ROP). In that letter we indicated that the NRC had determined that the ROP cornerstones of Occupational Radiation Safety, Public Radiation Safety, Emergency Preparedness, and Physical Protection would be incorporated into the routine ROP baseline inspection program effective January 1, 2005. Remaining results from our inspection of your Unit 1 Recovery Project continue to be documented in a separate Unit 1 integrated inspection report.

This report documents an NRC-identified finding which was determined to involve a violation of NRC requirements. However, because this finding was of very low safety significance and was entered into your corrective action program, the NRC is treating the violation as a non-cited violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy. In addition, a licensee-identified violation which was determined to be of very low safety significance is listed in this report. If you contest any non-cited violation or finding 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 United States 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 Senior Resident Inspector at the Browns Ferry Nuclear Plant.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure and your response, if any, 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/

Malcolm T. Widmann, Chief
Reactor Projects Branch 6
Division of Reactor Projects

Docket Nos.: 50-259, 50-260, 50-296, 72-052
License Nos.: DPR-33, DPR-52, DPR-68

Enclosure: Inspection Report 05000259/2006004, 05000260/2006004, 05000296/2006004, and 07200052/2006001 w/Attachment: Supplemental Information

cc w/encl.: (See page 3)

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure and your response, if any, 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/

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Distribution w/encl: (See page 4)

Report to Karl W. Singer from Malcolm T. Widmann dated October 30 , 2006.

SUBJECT: BROWNS FERRY NUCLEAR PLANT - INTEGRATED INSPECTION REPORT 05000259/2006004, 05000260/2006004, 05000296/2006004, and 07200052/2006001

Distribution w/encl.:
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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.:	05000259/2006004, 05000260/2006004, 05000296/2006004, and 07200052/2006001
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 - September 30, 2006
Inspectors:	T. Ross, Senior Resident Inspector R. Monk, Resident Inspector S. Freeman, Senior Resident Inspector (Section 1R07) G. Laska, Senior Operations Examiner (Section 1R11.2) S. Rose, Senior Operations Examiner (Section 1R11.2)
Approved by:	Malcolm T. Widmann, Chief Reactor Project Branch 6 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000259/2006004, 05000260/2006004, 05000296/2006004, 07200052/2006001; 07/01/2006 - 09/30/2006; Browns Ferry Nuclear Plant, Units 1, 2, and 3; Maintenance Effectiveness, and Identification & Resolution of Problems

The report covered a three-month period of routine inspections by the resident inspectors, a senior resident inspector from another site, and two senior operations examiners from Region II. One Severity Level IV non-cited violation was identified. The significance of most findings are indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or assigned a severity level after NRC management review. 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. <u>NRC-Identified and Self-Revealing Findings</u>

- A Severity Level IV, non-cited violation (NCV) of 10 CFR 50.73(a)(2)(i)(B) was identified by the inspectors for the licensee's failure to submit the required licensee event reports for several instances of multiple main steam relief valve (MSRV) test failures that resulted in a condition prohibited by Technical Specifications. This issue was documented in the licensee's corrective action program as Problem Evaluation Report 112190.

  The licensee's failure to provide written event reports was considered greater than minor because it could potentially impact the NRC's ability to carry out its regulatory function. However, since this failure to report per 10 CFR 50.73 did not actually impede or influence regulatory action, and the condition that required reporting under 10 CFR 50.73 was determined to be of very low safety significance because the as-found MSRV lift pressures were bounded by accident analyses, the NRC has characterized the significance of this reporting violation as a Severity Level IV in accordance with Section IV.A.3 and Supplement I of the NRC Enforcement Policy (Section 1R12)

B. <u>Licensee-Identified Violations</u>

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 have been entered into the licensee's corrective action program. The violation and corrective actions are listed in Section 4OA7 of this report.

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, except for two unplanned downpowers. On July 4, operators reduced power to 82% to remove the 2B reactor feedwater pump (RFP) from service in order to repair a through wall leak on the 2B RFP minimum flow valve body. The repairs were completed and the unit was returned to full power the same day. On September 20, operators conducted a rapid down power of Unit 2 to 91% power due to loss of level control of the 2A1 main feedwater (MFW) heater. Repairs were made and the unit was returned to full power the same day.

Unit 3 operated at essentially full power for the entire report period, except for two manual reactor scrams, a runback, and a rapid down power. On July 22, operators reduced power to 85% due to unexpected isolation of the 3A3 MFW heater from a spurious high level indication. The 3A3 MFW heater was returned to service the same day. On August 16, an unexpected runback of the 3A recirculation pump to 96% power occurred due to an automatic bypass of the variable frequency drive (VFD) A4 power cell following an electrical fault. Full power was restored later that day with the A4 cell still bypassed. On August 19, Unit 3 was manually scrammed from 100% power due to the sudden, simultaneous loss of both VFD's caused by an ethernet data storm that locked up the VFD central processing units (CPU). The VFD's were restored and the unit was returned to full power on August 22. On August 29, Unit 3 was manually scrammed from 100% power due to a large, unisolable electrohydraulic (EHC) fluid leak caused by an O-ring failure of the fast acting solenoid valve (FASV) connected to main turbine Control Valve #2. Repairs were made and the unit was returned to full power on September 1.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R04 Equipment Alignment

.1 Partial Walkdown

a. Inspection Scope

Partial System Walkdown. The inspectors performed partial walkdowns of the safety systems listed below to verify train operability, as required by the plant Technical Specifications (TS), while the other redundant trains were out of service or after the specific safety system was returned to service following maintenance. These inspections included reviews of applicable TS, operating instructions (OI), 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.

- 3A Control Rod Drive (CRD) Hydraulic System per PI&D flow diagrams 0-47E820 and 3-47E820, and 3-OI-85, Control Drive System
- Unit 3 Division II Core Spray (CS) system per PI&D flow diagram 3-47E814 and 3-OI-75, Core Spray System
- Unit 2 High Pressure Coolant Injection (HPCI) system per PI&D flow diagram 2-47E812 and 2-OI-73, High Pressure Coolant Injection System
- Unit 2 Reactor Core Isolation Cooling (RCIC) system per PI&D flow diagram 2-47E813 and 2-OI-71, Reactor Core Isolation Cooling System

b. Findings

No findings of significance were identified.

.2 Complete Walkdown

a. Inspection Scope

The inspectors completed a detailed alignment verification of the Unit 3 Standby Liquid Control (SLC) system, using the applicable P&ID flow diagram 3-47E854-1 and 3-OI-63, Standby Liquid Control System, to walkdown and verify equipment alignment and operability. The inspectors reviewed relevant portions of the Updated Final Safety Analysis Report (UFSAR) and TS. 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 inspectors examined control room operator logs; the applicable System Health Report; and any problem evaluation reports (PERs) that could affect system alignment and operability for the past year.

b. Findings

1R05 Fire Protection

a. Inspection Scope

Walkdowns. 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 fire areas (FA) and fire zones (FZ) listed below. Selected fire areas/zones 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 features 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, Volumes 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 Elev 621 and North 639 (Fire Zone (FZ) 2-5)
- Unit 2 Reactor Building South Elev 639 (FZ 2-6)
- Unit 3 Reactor Building Elev 621 and North 639 (FZ 3-4)
- 1A Electric Board Room (Fire Area (FA) 5)
- Unit 3 Reactor Building Elev 519 thru 565 East (FZ 3-2)
- Unit 3 Reactor Building Elev 519 thru 565 West (FZ 3-1)
- Unit 2 Reactor Building Elev 593 North (FZ 2-3)
- Unit 2 Reactor Building Elev 593 South and Residual Heat Removal (RHR) Heat Exchanger Rooms (FZ 2-4)
- 1A 480V Shutdown Board Room (FA 6)
- 1B 480V Shutdown Board Room (FA 7)

b. Findings

1R06 External 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 external flooding events. The inspectors reviewed flood analysis documents including: UFSAR Section 2.4, Hydrology, Water Quality, and Marine Biology, which included Appendix 2.4A, Maximum Possible Flood; and UFSAR Section 12.2.9.2.3 Flood Gate, for licensee commitments. The inspectors also interviewed cognizant licensee personnel knowledgeable about site flood protection measures and plant drainage plans. The inspectors performed walkdowns of risk-significant areas, susceptible systems and equipment, including the residual heat removal service water (RHRSW) and emergency equipment cooling water (EECW) pump rooms “A”, “B”, “C” and “D”, including the associated structures. The inspectors performed walkdowns of the Units 3 emergency diesel generator (EDG) rooms, and the associated cable/piping wall penetrations in rooms “3A”,”3B”, “3C”, and “3D”. The inspectors’ review included flood-significant features such as level switches, room sumps, door seals and the Reactor Building Flood Gate. Plant procedures and calculations for coping with flooding events were also reviewed to verify that licensee actions and maintenance practices were consistent with the plant’s design basis assumptions. The procedures reviewed during this inspection are listed in the Attachment to this report.

The inspectors also reviewed licensee corrective action documents for flood-related items identified in PERs written from 2005 through early 2006 to verify the adequacy of the corrective actions. The inspectors reviewed selected completed preventive maintenance procedures and work orders for identified level switches, pumps and flood barriers (e.g., Flood Gate) for completeness and frequency.

b. Findings

No findings of significance were identified.

1R07 Annual Heat Sink Performance

a. Inspection Scope

The inspectors reviewed the licensee's biofouling controls to verify that the licensee had identified any potential deficiencies that could lead to heat exchanger flow blockage problems due to biological macro-fouling and to verify that the licensee had identified any problems that could affect multiple safety related heat exchangers. Specifically, the inspectors reviewed the licensee's raw water fouling, corrosion control, and chemical treatment procedures to ensure they had a program, based on industry standards, for controlling clams and other biological agents. The inspectors also reviewed the data for several RHR heat exchanger visual inspections and EECW system treatments to gage the effectiveness of the program. Documents reviewed are in the Attachment to this report.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

.1 Requalification Activities Review

a. Inspection Scope

On August 7, 2006, inspectors observed the as-found simulator evaluations for group A of crew #4 in accordance with OPL177.01, Failure of Reactor Protection System “A” With Failure of “C” Standby Gas Treatment to Start, Recirc Pump Trip, Reactor Power Oscillations, Anticipated Transient Without Scram, Stuck Open Safety Relief Valve, and Safety Relief Valve Tailpipe break In Containment. The scenario was challenging, and involved critical equipment failures, abnormal operational transients and accident conditions.

The inspectors specifically evaluated the following attributes related to the operating crews’ 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 (AOI), Emergency Operating Instructions (EOI) and Operational Contingencies
- Timely and appropriate Emergency Action Level declarations per Emergency Plan Implementing Procedures (EPIP)

- Control board operation and manipulation, including high-risk operator actions
- Command and Control provided by the Unit Supervisor and Shift Manager

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

b. Findings

No findings of significance were identified.

.2 Biennial Review

a. Inspection Scope

The inspectors reviewed the facility operating history and associated documents in preparation for this inspection. During the week of September 18, 2006, the inspectors reviewed documentation, interviewed licensee personnel, and observed the administration of operating tests associated with the licensee's operator requalification program. Each of the activities performed by the inspectors was done to assess the effectiveness of the licensee in implementing requalification requirements identified in 10 CFR Part 55, "Operators' Licenses." The evaluations were also performed to determine if the licensee effectively implemented operator requalification guidelines established in NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," and Inspection Procedure 71111.11, "Licensed Operator Requalification Program." The inspectors also evaluated the licensee's simulation facility for adequacy for use in operator licensing examinations using ANSI/ANS-3.5-1985, "American National Standard For Nuclear Power Plant Simulators for use in Operator Training and Examination." The inspectors observed two crews during the performance of the operating tests. Documentation reviewed included written examinations, Job Performance Measures (JPMs), simulator scenarios, licensee procedures, on-shift records, simulator modification request records and performance test records, the feedback process, licensed operator qualification records, re-mediation plans, and watchstanding and medical records. These records were inspected using the criteria listed in Inspection Procedure 71111.11. A list of the documents reviewed by the inspectors is included in the List of Documents Reviewed attachment to this report.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

Routine 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 (a)(1) goals and corrective actions (i.e., Ten Point Plan). Both of these systems had exceeded their reliability performance criteria and were 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; and SPP 3.1, Corrective Action Program. The inspectors also reviewed applicable work orders, surveillance records, PERs, system health reports, engineering evaluations, and MR expert panel minutes; and attended MR expert panel meetings to verify that regulatory and procedural requirements were met.

- RHRSW/EECW Motor Failures
- Main Steam Safety/Relief Valve (MSRV) Setpoint Drift and Pilot Valve Leakage

b. Findings

Introduction: A Severity Level IV, non-cited violation (NCV) of 10 CFR 50.73(a)(2)(i)(B) was identified by the inspectors for the licensee's failure to submit the required licensee event reports (LERs) for several instances of multiple MRSV test failures that occurred following previous Unit 2 and 3 refueling outages. These test failures resulted in conditions prohibited by TS 3.4.3 due to two or more MSRV as-found lift settings being found outside ± 3% of the TS lift setpoints.

Description: As-found MSRV lift setpoint testing is required by TS 3.4.3 and the licensee's American Society of Mechanical Engineers (ASME) Section XI Pump and Valve Program. This testing is accomplished every fuel cycle by an independent laboratory after the associated refueling outage. During a review of past surveillance test results of Unit 2 and 3 MSRVs from previous refueling outages, the inspector determined that numerous setpoint failures had been identified by the licensee in excess of TS 3.4.3 requirements. However, this condition was not recognized as a reportable event pursuant to 10 CFR 50.73(a)(2)(i)(B) for a condition prohibited by TS. More specifically, the as documented records in surveillance procedure 0-SR-3.4.3.1.b, Bench test Relief Valves - As-Found, indicated that multiple MSRV setpoint failures in excess of TS 3.4.3 were identified following the Unit 2 Cycle 11 (U2C11), U2C12, U2C13, U3C10 and U3C11 refueling outages which were not reported as an LER as required by 10CFR50.73.

Analysis: The licensee's failure to provide written event reports was considered greater than minor because it could potentially impact the NRC's ability to carry out its regulatory function. As discussed in Section IV of the NRC Enforcement Policy, the significance of violations involving the failure to make required reports is not dispositioned using the Reactor Oversight Program's Significance Determination Process. However, since this failure to report per 10 CFR 50.73 did not actually impede or influence regulatory action, and the condition that required reporting under 10 CFR 50.73 was determined to be of very low safety significance because the as-found MSRV lift pressures were bounded by accident analyses, the NRC has characterized the significance of this reporting violation as a Severity Level IV in accordance with Section IV.A.3 and Supplement I of the NRC Enforcement Policy.

Enforcement: Pursuant to 10 CFR 50.73, the licensee shall submit an LER for any type of event described therein within 60 days after discovery of the event. Contrary to 10 CFR 50.73, the licensee failed to recognize and report within 60 days several instances of multiple MSRV test failures which met the reporting requirements of 10 CFR 50.73(a)(2)(i)(B) for a condition or operation prohibited by TS. Because the criteria of Section VI.A of the Enforcement Policy were satisfied, which included the licensee's initiation of PER 112190, this violation will be considered an NCV: NCV 05000260 and 296/2006004-01, Failure To Report Several Instances of Multiple MSRV Setpoint Test Failures Per 10 CFR 50.73.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

For planned online work and/or emergent work that affected the risk significant systems as listed below, the inspectors reviewed licensee maintenance risk assessments and actions taken to plan and control work activities to effectively manage and minimize risk. The inspectors verified that risk assessments and risk management actions (RMA) were being conducted 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 Matrix. The inspectors also evaluated the adequacy of the licensee's risk assessments and the implementation of RMAs.

- 2B and 2D CS Pumps, 1B CRD Pump, Battery and Battery Board #3 Out of Service (OOS)
- Battery and Battery Board #3, 3A CRD Pump, and 3EA LPCI MG Set OOS
- Unit 3 SLC System (both trains), and 3A CRD Pump OOS
- 1B 480 Shutdown Board, D EDG, and D3 EECW Pump OOS
- 2A CRD Pump and Unit 2 Division 1 CS System OOS
- 2A CRD Pump, and 3A and 3C RHR Pumps OOS
- Work week 2629

b. Findings

No findings of significance were identified

1R15 Operability Evaluations

Routine Baseline Review

a. Inspection Scope

The inspectors reviewed the operability/functional evaluations listed below to verify technical adequacy and ensure that the licensee had adequately assessed TS operability. The inspectors also reviewed applicable sections of the UFSAR to verify that the system or component remained available to perform its intended function. In addition, where appropriate, 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 on a daily basis to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations.

- Unit 1 and 2 EDG's B, C & D Misaligned Brush Spring Clips (PER 106157)
- 3-FCV-74-68, RHR Loop Low Pressure Coolant Injection (LPCI) Inboard Injection Valve Seat Leakage (PER 104292)
- A RHRSW Pump Room Sump Pump A Vacuum Breaker Blocked By Plastic Cover (PER 106821)
- Unit 1, 2 & 3 Fuel Pool Cooling Discharge Check Valves Failing To Fully Close (PER 108398)
- Unit 2 CS Injection Valves 2-CKV75-25 & 26 Seat Leakage (PER 103954)

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed the post-maintenance tests (PMT) listed below to verify that procedures and test activities confirmed system, structure, or component (SSC) operability and functional capability following maintenance. The inspectors reviewed the licensee's completed test procedures to ensure any of the SSC safety function(s) that may have been affected were adequately tested, that the acceptance criteria were consistent with information in the applicable licensing basis and/or design basis documents, and that the procedure had been properly reviewed and approved. The inspectors also witnessed the test and/or reviewed the test data, to verify that test results adequately demonstrated restoration of the affected safety function(s). The inspectors also verified that PMT activities were conducted in accordance with applicable work order (WO) instructions, or procedural requirements, including SPP-6.3, Post-Maintenance Testing, and MMDP-1, Maintenance Management System. Furthermore, the inspectors reviewed problems associated with PMTs that were identified and entered into the CAP.

- Unit 2 PMT for EDG “C” 24 Hour Run for Generator Brush Spring Clip Replacement per 0-SR-3.8.1.7C)
- Unit 3 PMT for 3EA LPCI Motor Generator Set Bearing Replacement per 0-TI-230, Predictive Monitoring Program
- PMT for D2 RHRSW Pump Low Flow per 3-SI-4.5.C.1(3), RHRSW Pump and Header Operability Flow Test
- PMT for Diesel Driven Fire Pump Strainer Leak 0-SI-4.11.B.2.a, Diesel Driven Fire Pump Operability
- Unit 2 PMT for the Preferred Transformer per Load Testing Sequence in WO 06-718844-006
- PMT for the B Control Bay Chiller Slide Valve and Fan Replacements per Work Order step1.9 and 0-O1-31, Control Bay and Off-Gas Treatment Building Air Conditioning System

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

.1 Unit 3 Force Outage Due To Manual Scram

a. Inspection Scope

On August 19 - 21, 2006, the inspectors examined critical activities associated with an Unit 3 unplanned shutdown to verify that they were conducted in accordance with TS, applicable procedures, and the licensee’s outage risk assessment and management plans. Some of the more significant outage activities monitored, examined and/or reviewed by the inspectors were as follows:

- Control of Hot Shutdown conditions, and critical plant parameters
- Outage risk assessment and management
- Control and management of forced outage and emergent work activities
- Reactor Startup and Power Ascension activities

The inspectors also verified that selected TS, license conditions, license commitments, and administrative prerequisites were being met prior to Unit 3 mode changes. Furthermore, the inspectors examined RCS identified and unidentified leakage tests.

Corrective Action Program

The inspectors reviewed PERs generated during the Unit 3 forced outage to verify that initiation thresholds, priorities, mode holds, and significance levels were assigned as required. Certain aspects of the resolution and implementation of corrective actions of several PERs were also examined and/or verified.

b. Findings

No findings of significance were identified.

.2 Unit 3 Force Outage Due To Manual Scram

a. Inspection Scope

On August 29 - 31, 2006, the inspectors examined critical activities associated with the Unit 3 unplanned shutdown to verify that they were conducted in accordance with TS, applicable procedures, and the licensee's outage risk assessment and management plans. Some of the more significant outage activities monitored, examined and/or reviewed by the inspectors were as follows:

- Control of Hot Shutdown conditions and critical plant parameters
- Outage risk assessment and management
- Control and management of forced outage and emergent work activities
- Plant Oversight Review Committee (PORC) meeting for restart
- Power Ascension activities

The inspectors also verified that selected TS, license conditions, license commitments, and administrative prerequisites were being met prior to Unit 3 mode changes. Furthermore, the inspectors examined RCS identified and unidentified leakage tests.

Corrective Action Program

The inspectors reviewed PERs generated during the Unit 3 forced outage to verify that initiation thresholds, priorities, mode holds, and significance levels were assigned as required. Certain aspects of the resolution and implementation of corrective actions of several PERs were also examined and/or verified.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors witnessed portions and/or reviewed completed test data of the following surveillance tests of risk-significant and/or safety-related systems 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.

- 3-SR-3.3.5.1.6(ADS A), ADS Logic System Functional Test - Bus A Time Delay Relay Calibration and Bus Power Monitor Test
- 2-SR-3.5.3.3, RCIC System Rated Flow at Normal Operating Pressure *
- O-SR-3.6.4.3.1, Standby Gas Treatment System Train Operation
- 0-SR-3.8.1.1(D), Diesel Generator D Monthly Operability Test
- 2-SI-4.2.E-1(B), Drywell Equipment Drain Sump Flow Integrator Calibration **

* Quarterly IST
** RCS leak detection

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 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.

- TACF 2-06-009-099, Temporary Transformer for Unit 2 alternate RPS power supply
- TACF 3-06-005-024, Temporary setpoint change to temperature controllers 3-TIC-024-0080 and 3-TIC-024-0085 for RBCCW Heat Exchangers to support data collection to support 3 Unit operation

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Identification & Resolution of Problems

.1 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 and Observations

There were no significant findings identified from this overall review of the PERs issued each day.

.2 Focused Annual Sample Review - Operator Workarounds

a. Inspection Scope

The inspectors conducted a review of existing Operator Workarounds (OWA) to verify that the licensee was identifying OWAs at an appropriate threshold, entering them into the corrective action program, establishing adequate compensatory measures, prioritizing resolution of the problem, and implementing appropriate corrective actions in a timely manner commensurate with its safety significance. The inspectors examined all active OWAs listed in the Limiting Condition of Operation Tracking (LCOTR) Log, and reviewed them against the guidance in OPDP-1, Attachment L, Operator Workarounds. The inspectors also discussed these OWAs in detail with onshift operators to assess their familiarity with the degraded conditions and knowledge of required compensatory actions. Furthermore, the inspector walked down selected OWAs, and verified the ongoing performance, and/or feasibility of, the required actions. Lastly, for each Priority 1 OWA the inspector reviewed the applicable PER, including the associated functional evaluation and corrective action plans (both interim and longterm).

b. Findings and Observations

Described below are two separate inspector identified issues that are being characterized as unresolved items (URIs).

Lack of Compensatory Measures To Ensure Sufficient Alternate Shutdown Cooling Flow During Appendix R Events

On August 30, 2006, during a review of the licensee's list of outstanding PERs involving degraded and/or non-conforming conditions per Regulatory Issue Summary 2005-20 (a.k.a. Generic Letter (GL) 91-18), the inspectors identified that the compensatory measures of the functional evaluation for PER 108439 had not been implemented.

On August 9, 2006, during a review of RHR pump net positive suction head (NPSH) calculations, Engineering determined that the reactor vessel pressure operating range specified in their Appendix R calculations for safe shutdown was too broad to ensure adequate RHR pump flow for Alternate Shutdown Cooling. The reactor pressure operating range for an Appendix R event was originally specified to be greater than 100 psig but less than 320 psig. The licensee's Safe Shutdown Instructions (SSI) for mitigating Appendix R events required operators to depressurize the reactor vessel and maintain this pressure band for Alternate Shutdown Cooling. However, the licensee's engineering evaluation identified that at reactor pressures approaching 320 psig the RHR pump flow would be significantly reduced and would not provide sufficient cooling. The analyzed RHR pump flow required for Alternate Shutdown Cooling was 6000 gpm. In the Alternate Shutdown Cooling Mode, RHR flows less than 6000 gpm would have an adverse impact on peak suppression pool temperature. It was subsequently determined by Engineering, that reactor pressure must be reduced and maintained below 220 psig to achieve the minimum required RHR flowrate.

In the PER 108439 functional evaluation (i.e., operability determination) that was approved on August 18, Engineering concluded that the currently specified reactor pressure range of 100 - 320 psig did not allow for adequate Alternate Shutdown Cooling flow during Appendix R events to maintain suppression pool temperatures within Appendix R analyzed limits. Consequently, Engineering stated in their functional evaluation that "the reactor pressure band must be maintained between 120 - 200 psig," and "until the appropriate design outputs are revised, this action will be implemented by an Operator Work Around." This compensatory measure (i.e., OWA) would be required until completion of the PER corrective action plan (e.g., SSI revision). However, on August 30, the inspectors discovered that the compensatory measure, prescribed by the functional evaluation for ensuring sufficient Alternate Shutdown Cooling flow for safe shutdown during Appendix R events, was not implemented due to a communication breakdown between Engineering and Operations. The licensee promptly instituted a Priority 1 OWA to implement the required compensatory measures and initiated another PER 109829 to address their process breakdown. Subsequently, on October 1, the licensee revised the applicable Appendix R SSIs accordingly. This revision to the SSIs implemented the necessary permanent corrective actions and eliminated any further need for an OWA.

In order to fully assess the enforcement implications and safety significance of this issue, additional information from the licensee will be needed. Consequently, pending further review by the NRC (e.g., determination of the safety significance), this issue will be identified as URI 05000260, and 296/2006004-02, Lack of Compensatory Measures To Ensure Sufficient Alternate Shutdown Cooling Flow During Appendix R Events.

<u>Incomplete and Unfeasible Compensatory Measures For Ensuring RHR Pump NPSH During Appendix R Events</u>

On August 25, 2006, the inspectors identified that the required actions of the Priority 1 OWAs listed in the LCOTR as #2-074-OWA-2006-0070 and #3-074-OWA-2006-0071 were inconsistent with the required compensatory measures of the applicable functional evaluation for PER 107105. In addition, the inspectors determined that the required compensatory measures of PER 107105 were not feasible as Appendix R manual actions.

On July 19, 2006, the licensee notified the NRC, that the operation of drywell coolers had not been modeled in the General Electric containment analyses. The results from the licensee's preliminary re-analysis, which assumed drywell coolers remained in operation, indicated that containment pressure during Appendix R events would be reduced below the necessary overpressure required to provide for adequate RHR pump NPSH. To address this degraded/non-conforming condition per RIS 2005-20, the licensee initiated PER 107105, which required a functional evaluation (i.e., operability determination). On July 27, licensee Engineering issued its approved functional evaluation which concluded that a "re-evaluation of the Appendix R event shows there will be sufficient available NPSH to the RHR pump, based on the drywell blowers being secured within 2 hours after the start of the Appendix R event." The licensee further stated in their functional evaluation that "appropriate actions must be taken to secure all drywell blowers within two hours of the start of an Appendix R event ... Until appropriate design outputs are revised, this action will be implemented by an Operator Work Around." However, on August 25, the inspectors discovered that the Priority 1 OWA instituted by Operations contained inconsistent and misleading guidance which stated "by securing 8 of the 10 drywell blowers prior to T=3 hours ... RHR pump NPSH will be adequate." This guidance, and the onshift operators interpretation of it, was not consistent with the functional evaluation and would not have ensured adequate RHR pump NPSH during Appendix R events. Once notified, Operations promptly corrected the misleading OWA guidance and initiated PER 109516.

In addition to reviewing the functional evaluation of PER 107105, and the associated Priority 1 OWAs, the inspectors reviewed the applicable Fire Protection License Condition Impact Evaluation (LCIE) and specifically examined the feasibility of implementing the required compensatory measure (i.e., Appendix R manual action). Based on a walkdown of the OWA prescribed Appendix R manual actions, and detailed discussions with responsible Operations and Engineering personnel, the inspectors concluded that the manual actions were not feasible. These actions would require the auxiliary operator(s) to open ten 480V AC breakers that are located in five different fire areas within two hours, in order to de-energize all ten drywell cooler blowers. Although two hours was more than sufficient time under normal circumstances to locate and open these breakers, the operator's path could require him to traverse through fire affected areas or even require operating breakers in a fire affected area. After further deliberation, the licensee decided their initial manual actions were not really feasible and subsequently revised the required manual actions. On August 30, the licensee instituted revised manual actions for #2-074-OWA-2006-0070 and #3-074-OWA-2006-0071. These revised manual actions required that "the RBCCW [Reactor Building Closed Cooling Water] Outboard Supply (2-70-312) or Return (2-70-311) will be Manually Closed within 2 hours of entering an Appendix R event for All Fires." The inspectors concluded, that these actions were much more feasible and met the intent of securing all drywell coolers.

Subsequently, on October 1, the licensee revised all applicable Appendix R SSIs to require tripping both RBCCW pumps or isolating 2/3-FCV-70-47 to secure all RBCCW flow to the drywell coolers. This revision to the SSIs implemented the necessary permanent corrective actions and eliminated any further need for an OWA.

In order to fully assess the enforcement implications and safety significance of this issue, additional information from the licensee will be needed. Consequently, pending further review by the NRC (e.g., determination of the safety significance), this issue will be identified as URI 05000260, and 296/2006004-03, Incomplete and Unfeasible Compensatory Measures For Ensuring RHR Pump NPSH During Appendix R Events.

4OA3 Event Follow-up

.1 Unit 3 Manual Reactor Scram

a. Inspection Scope

On August 19, 2006, Unit 3 was manually scrammed from 100% power due to the simultaneous loss of both the 3A and 3B Variable Frequency Drives (VFD) that was caused by an apparent lockup of the VFD central processing units (CPU). 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 interviewed responsible onshift Operations personnel, examined the implementation of the applicable ARPs and AOIs, including 3-AOI-100-1, Reactor Scram, and reviewed the written notification made in accordance with 10 CFR 50.72. The inspector discussed the preliminary cause of the loss of both VFDs with responsible Operations and Engineering personnel. This review included only initial event followup.

b. Findings

No findings of significance were identified during the initial event followup.

.2 Unit 3 Manual Reactor Scram

a. Inspection Scope

On August 29, 2006, Unit 3 was manually scrammed from 100% power due a large, unisolable electro-hydraulic control (EHC) fluid leak caused by an O-ring failure at the connection between the fast acting solenoid valve (FASV) and the main turbine #2 Control Valve. An inspector 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 interviewed responsible onshift Operations personnel, examined the implementation of 3-AOI-100-1 and reviewed the written notification made in accordance with 10 CFR 50.72. The inspector discussed the preliminary cause of the EHC system leak with knowledgeable Operations and Maintenance personnel. The inspector also reviewed the licensee's post-trip review report, restart checklist, and attended the Plant Oversight Review Committee meeting for approving restart. This review included only initial event followup.

b. Findings

No findings of significance were identified during the initial event followup.

.3 (Closed) LER 05000296/2006-001-00, Main Steam Relief Valve Inoperability LCO Exceeded as a Result of Lift Setpoint Drift

a. Inspection Scope

The inspectors reviewed the LER dated June 23, 2006, and the applicable PER 102298, including associated apparent cause determination and corrective action plans.

Following the Unit 3 Cycle 12 (U3C12) refueling outage, the licensee tested 13 MSRVs that had been in service during the previous fuel cycle. However, during surveillance testing, two of the 13 MSRVs lifted at a pressure outside the TS 3.4.3 allowed limit of plus or minus 3% from the required setpoint. The cause of the MSRV as-found setpoints being outside their TS limits was determined to be a corrosion bonding between the pilot valve seat and disc, which is a recognized industry problem. The failure of these two MSRVs to lift within the allowed setpoint limits constituted a condition prohibited by TS 3.4.3. However, subsequent Pressure Transient Analysis by the licensees concluded that the as-found condition of the MSRVs from U3C12 would have been sufficient to fulfill the pressure relief safety function during design basis over-pressure transient events.

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b. Findings

This LER is closed. Since the setpoint drift problems were found during surveillance testing this LER was dispositioned as an NCV in Section 4OA7 of this report.

4OA5 Other

.1 Independent Spent Fuel Storage Installation Operations Inspection and 10CFR72.48 Evaluations Review

a. Inspection Scope

The inspectors examined routine performance of normal independent spent fuel storage installation (ISFSI) operations activities. [Note, there were no ISFSI unloading/loading campaigns conducted in 2006.] In particular, the inspector reviewed licensee implementation of 0-SR-DCS3.1.2.1, Spent Fuel Storage Inspection, and 2-SR-2, Table 1.39, Hi-Storm/Overpack Heat Removal System Operability. The inspectors also reviewed the special nuclear material (SNM) inventory forms of SPP-5.8, Special Nuclear Material Control, for each of the three loaded Hi-Storm casks currently located at the ISFSI pad. Furthermore, the inspectors toured the ISFSI to verify configuration control of the loaded Hi-Storm casks in accordance with Certificate of Compliance (CoC) surveillance requirements. During this tour the inspectors also verified the locations of environmental dosimetry, examined radiological postings and radioactive material labels, and reviewed recent radiological dose rate and contamination surveys. In addition to routine operations activities, the inspectors also reviewed seven 10CFR72.48 Screening Reviews for various ISFSI procedure and design changes, to verify these changes were consistent with the license and CoC, and did not reduce program effectiveness. [Note, none of the procedural or design changes conducted by the licensee since the last ISFSI inspection in 2005 required a full 10CFR72.48 evaluation, all changes were screened out by the screening reviews.]

b. Findings

No findings of significance were identified.

4OA6 Management Meetings

.1 Exit Meeting Summary

On October 5, 2006, the resident inspectors presented the integrated inspection results to 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.

4OA7 Licensee Identified Violations

The following violation of very low significance (green) was identified by the licensee and is a violation of NRC requirement which meets the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for disposition as a NCV.

Technical Specification 3.4.3 requires that twelve of thirteen Main Steam Safety/Relief Valves lift at a setpoint within plus or minus 3% of a specified value. Contrary to this, during surveillance testing following the U3C12 refueling outage, the licensee identified that at least two valves tested outside the TS allowed band as described in the licensee's PER 102298. This finding is of very low safety significance because the as found lift setpoint conditions of the Unit 3 MSRVs were analyzed and determined to meet the design basis criteria for an over-pressurization event.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

B. Aukland, Nuclear Plant Manager
T. Brumfield, Site Nuclear Assurance Manager
J. Burton Design Engineering Manager
D. Campbell, Lead Requalification Training Instructor
P. Chadwell, Operations Superintendent
J. Corey, Radiation Protection Manager
W. Crouch, Nuclear Site Licensing & Industry Affairs Manager
J. DeDimenico, Asst. Nuclear Plant Manager
R. DeLong, Site Engineering Manager
A. Elms, Nuclear Plant Operations Manager
A. Feltman, Emergency Preparedness Supervisor
A. Fletcher, Field Maintenance Superintendent
R. Jones, General Manager of Site Operations
D. Langley, Site Licensing Supervisor
D. Matherly, Human Performance Manager
J. Mitchell, Site Security Manager
D. Nye, Maintenance & Modifications Manager
B. O'Grady, Site Vice President
C. Ottenfeld, Chemistry Manager
C. Rasby, Supervisor - Medical
D. Sanchez, Training Manager
E. Scillian, Operations Training Manager
J. Sparks, Outage Manager
J. Steele, Outage Manager

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Open
None.

Opened and Closed

05000260 and 296/2006004-01	NCV	Failure To Report Several Instances of Multiple MSRV Setpoint Test Failures Per 10 CFR 50.73. (Section 1R12)

Opened

05000260 and 296/2006004-02	URI	Lack of Compensatory Measures To Ensure Sufficient Alternate Shutdown Cooling Flow During Appendix R Events (Section 4OA2.2)
05000260 and 296/2006004-03	URI	Incomplete and Unfeasible Compensatory Measures For Ensuring RHR Pump NPSH During Appendix R Events (Section 4OA2.2)

Closed

05000296/2006-001-00	LER	Main Steam Relief Valve Inoperability LCO Exceeded as a Result of Lift Setpoint Drift (Section 4OA3.3)

Discussed

None.

LIST OF DOCUMENTS REVIEWED

Section 1R06: Flood Protection Measures

Procedure 0-AOI-100-3, Revision 29, Flood Above Elevation 558'
MPI-0-260-DRS001, Revision 29, Inspection and Maintenance of Doors
MPI-0-000-INS001, Revision 10, Inspection of Flood Protection Devices
1-SR-2(DF), Revision 22, Instrument Checks and Observations
MAI-3.4B, Revision 7, Installation of Flood and Moisture Intrusion Seals
RHRSW Pump Compartment Sump and Sump Pump Capacity MD-Q0023-870149
PER 109318 Leakage outside RHRSW Pump Room B, C, D while sump pumps running
PER 109946 Flood Gate non-functional for 19 months
Flood Gate Past Operability Determination
WO 04-724220, Door Preventative Maintenance
WO 05-711560, Door Preventative Maintenance
WO 05-719013, Door Preventative Maintenance
WO 05-722393, Door Preventative Maintenance
WO 05-723348, RHRSW pump room level switch Preventative Maintenance
WO 04-712757, RHRSW pump room level switch Preventative Maintenance
WO 05-710867, Flood Gate Corrective Maintenance
WO 05-720154, Flood Gate Preventative Maintenance
WO 06-710849, Flood Gate Preventative Maintenance
DWG 44N236, Flood Gate Machinery Arrangement
DWG 3IN209, Concrete Flood Protection Wall Outline

Section 1R07: Annual Heat Sink Performance

0-OI-23, Residual Heat Removal Service Water System, Revision 72
CI-137, Raw Water Chemical Treatment, Revision 17
CI-137.5, Raw Water Chemical Treatment Molluscicide Control, Revision 27
0-TI-389, Raw Water Fouling and Corrosion Control, Revision 10
Performance of CI-137.5, dated June 13, 2006
Performance of 0-TI-389, dated June 13, 2006
WO 05-716953-000, Routine Maintenance and Cleaning of RHR Heat Exchanger 3A
WO 05-716954-000, Routine Maintenance and Cleaning of RHR Heat Exchanger 3B

Section 1R11: Licensed Operator Requalification - Biennial Review

Procedures:
TRN 11.4, Continued Training for Licensed Personnel Rev 0011. 0605/2006.
TRN 11.10, Annual Requalification Examination Development and Implementation. Rev 0011 09/01/2006.
TRN 11.14, TVA Operator Licensing Examination Security Program. Rev 0004.9/1/2006
TRN 11.12, Job Performance Measures Development, Administration, and Evaluation manual Rev. 3. 3/10/2003.

Records:
Badge Access Transaction Reports for Reactivation of Licenses (3)
Licensed Operator Medical Records (12)
Feedback Summaries
Remedial Training Records

Written Examinations Reviewed:
Inspectors reviewed two written examinations that were administered for the 2005 biennial requalification Examinations:
RO biennial Requalification exam 05C6W3RO
SRO biennial Requalification exam 05C6W3SRO

JPMs:
JPM 23F, Rev 4
JPM 341F, Rev.1
JPM 62, Rev 5
JPM 339, Rev 2. " Operate the RHR system I with suction from the CST in Accordance with 3-EOI Appendix 10C required to align CST to RHR system I as Directed by the Unit 3 operator".
JPM 34, Rev 8
JPM 190 Rev. 0, Respond to off-gas post-treatment radiation hi-hi-hi
JPM 177TC, Rev 9, Classify the event per the REP (secondary containment radiation)
JPM 606F, Rev 3, 2-EOI appendix 3A - SLC injection
JPM 23, Rev 7, 2-EOI appendix 6E - injection subsystem lineup - CS system II
JPM 172TCF, Rev 0, Classify the event per the REP (unisolable leak outside primary containment)
JPM 94, Rev. 8, Shut down HPCI and return to standby readiness.

Simulator Scenarios:
LOR-Exam-22, Rev 0, Loss of RPS MG set A, RCIC steam line break, HPCI auto start failure, loss of 4KV shutdown board D, low suppression pool water level, ADS valve failure
LOR-Exam-32, Rev 0, NI failure, loss of DG 'D', loss of off-site power, HPCI failure, LOCA, diesel and core spray failures.

Simulator Performance Testing:

Transient Tests:
100% Unisolable Main Steamline Break, IST-7.5, Revision 12.
Loss of Coolant Accident, IST-7.4, Revision 13.
Trip of One Reactor Coolant Pump, IST-7.9, Revision 12.
Simultaneous Closure of All MSIVs, IST-7.1, Revision 13.
Saturated Conditions (ECCS Inhibited), IST-7.6, Revision 14.

Normal Tests:
Normal Operations test procedure 100% Power to Hot Standby, IST-2.1, Revision 10.

Malfunction tests:
Loss of Instrument Air, IST-6.1.1.9, Revision 7.
Steam Generator Safety Valve Fails Open, IST-6.8.10, Revision 4.

Rods Fail to Move, IST-6.4.1, Revision 5.
Leak in Charging Line, IST-6.5.12, Revision 6.
Loss of ESF BUS 1DA, IST-6.6.5.1, Revision 2.
Failure of ESF Transformer XTF-4, IST-6.6.5.1, Revision 5.
Failure of ESF Transformer XTF-5, IST-6.6.18.2, Revision 4.
Loss of 125 VDC Bus 1HA, IST-6.6.7.1, Revision 6.
Loss of Unit Auxiliary Transformer, IST-6.6.9, Revision 3.
FW Break between FE-496 and FW Isol Valve 1611 (SGC), IST-6.7.16.1, Revision 4.
Loss of Service Water System, IST-6.1.4.2, Revision 5.
Ejected Rod (48 Rods), IST-6.4.5, Revision 5.
Pressurizer Safety Valve Fails Open, IST-6.ll.7, Revision 6
Loss of Normal and Emergency Feedwater, IST-6.7.1.2, Revision 7.
Reactor Trip Breaker Inadvertent Open, IST-6.10.9.1, Revision 7.
RCS RTD Loop Failure (Cold Leg, Fails low), IST-6.12.8.2, Revision 5.
RB Spray pump failure, IST-6.13.7, Revision 3.

Simulator Problem Reports (Prs):
4387
4376
4329
4378
4337

4450 8/10/06
4371 03/02/06
4310 12/12/05
4049 6/30/04
4093 11/03/04
4066 7/25/04
4192 9/27/05
4197 9/26/05
4198 9/26/05
4200 9/26/05