August 22, 2000

Mr. G. Rainey, President
PECO Nuclear
Nuclear Group Headquarters
Correspondence Control Desk
P.O. Box 195
Wayne, Pennsylvania 19087-0195

SUBJECT: NRC'S PEACH BOTTOM INSPECTION REPORT 05000277/2000-005, 05000278/2000-005

Dear Mr. Rainey:

On July 13, 2000, the NRC completed a team inspection of the design and performance capability of the feedwater system and the standby liquid control system and the evaluation of changes, tests and experiments at the Peach Bottom Atomic Power Station Units 2 and 3. The results of this inspection were discussed on July 20, 2000, with Mr. G. Johnson and other members of your staff.

The inspection was an examination of 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. Within these areas, the inspection consisted of a selected examination of procedures and representative records, observations of activities, and interviews with personnel.

The NRC team identified one issue of very low safety significance (Green). The issue involved the inservice tests for the standby liquid control pumps. A two-minute wait was not mandated, as required in the applicable Code, by the test procedure before pump flow and pressure measurements were recorded. Because of the very low safety significance, this violation was not cited. If you contest this non-cited violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Peach Bottom Station.

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Sincerely,

/RA/John White for/

Wayne D. Lanning, Director
Division of Reactor Safety

Docket Nos. 05000277, 05000208
License Nos. DPR-44, DPR-56

Enclosure: NRC Inspection Report 05000277/2000005 and 05000278/2000005

cc w/encl:
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J. Doering, Vice President, Peach Bottom Atomic Power Station
M. Warner, Plant Manager, Peach Bottom Atomic Power Station
J. A. Hutton, Director, Licensing, PECO Nuclear
G. D. Edwards, Chairman, Nuclear Review Board
R. Boyce, Director, Nuclear Quality Assurance
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U. S. NUCLEAR REGULATORY COMMISSION

REGION I

| | |
|---|---|
| Docket Nos: | 05000277, 05000278 |
| License Nos: | DPR-44, DPR-56 |
| Report Nos: | 05000277/2000-005, 05000278/2000-005 |
| Licensee: | PECO Energy Company<br>Correspondence Control Desk<br>P.O. Box 195<br>Wayne, PA 19087-0195 |
| Facility: | Peach Bottom Atomic Power Station Units 2 and 3 |
| Location: | 1848 Lay Road, Delta, PA 17314 |
| Dates: | June 26 - 30, July 10 - 13, 2000, and July 20, 2000 |
| Inspectors: | L. Cheung, Senior Reactor Inspector, Team Leader, DRS<br>R. Bhatia, Reactor Inspector, DRS (part time)<br>T. Burns, Reactor Inspector, DRS<br>J. Carrasco, Reactor Inspector, DRS<br>L. James, Reactor Engineer, DRS<br>K. Kolaczyk, Reactor Inspector, DRS<br>A. Smith, Reactor Inspector (Trainee) , DRS<br>K. Young, Reactor Inspector, DRS |
| Approved by: | William H. Ruland, Chief<br>Electrical Branch<br>Division of Reactor Safety |

## SUMMARY OF FINDINGS

IR 05000277-00-005, IR05000278-00-005, on 06/26-06/30/2000, 07/10-07/13/2000, PECO Energy Company, Peach Bottom Atomic Power Station; Units 2 and 3. Annual baseline inspection of System Design and Functional Capability, Inservice Testing

The inspection was conducted by a team of region-based inspectors. This inspection identified one green issue. The significance of issues is indicated by their color (green, white, yellow, red) and was determined by the Significance Determination Process.

**Cornerstone: Mitigating Systems**

- Green. The inspectors identified a non-cited violation for failure to properly conduct inservice tests of the standby liquid control system tests. Specifically, the test procedure did not mandate a two-minute wait, as required by the applicable Code, before pump flow and pressure measurements were recorded. PECO demonstrated that the failure to wait the required two-minute period did not affect system operability. This finding had very low safety significance because the two-minute wait period was only to assure that a valid assessment of the pump performance under normal operating conditions was obtained. (Section 1R21.3)

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# Report Details

## 1. REACTOR SAFETY

**Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity**

1R21 Safety System Design and Performance Capability

.1 Standby Liquid Control System (SLCS)

.1.1 Design - Mechanical, Electrical, and Instrumentation and Controls

a. Inspection Scope

The team reviewed the SLCS design and licensing basis documents, including the Updated Final Safety Analysis Report (UFSAR), plant Technical Specifications (TS) and design basis documents (DBD)s to determine the system and component functional requirements during normal and accident conditions. For the documents reviewed, which included mechanical and electrical calculations and analyses, the team verified that the assumptions were appropriate, that proper engineering methods and models were used, and that there was an adequate technical basis to support the conclusions. If appropriate, the team performed independent calculations to evaluate the document adequacy. The review was performed to determine that: (1) the design basis was in accordance with the licensing commitments and regulatory requirements; and (2) the design output documents such as drawings and design calculations were correct.

The team also reviewed the UFSAR, plant TS, and DBDs for two interfacing systems; the instrument air system and reactor water cleanup system. For these systems, piping and instrumentation drawings, electrical schematics and configuration baseline documents were examined to assess the capability of the systems to ensure that the SLCS satisfied its design functions.

Finally, the team performed a walkdown of the SLCS in the field and compared it to the design drawings and specifications. Components/conditions examined included piping and pipe supports, system heat tracing, operator aids, area heating and ventilation systems, and storage of transient combustibles.

b. Issues and Findings

There were no findings identified.

.1.2 Operations and Maintenance

a. Inspection Scope

The team reviewed a number of activities to verify that the SLCS was installed, operated and maintained consistent with the design and licensing bases. The operational readiness and material condition of the SLCS was assessed by conducting system walkdowns and by reviewing procedures, operator logs, design documents, component maintenance history records, and system health reports. The team also interviewed licensed and non-licensed operators and engineers. As part of this review, the team evaluated a sample of licensee-identified problems documented in their performance enhancement program (PEP) reports to assess the effectiveness of the licensee's corrective actions.

The documents reviewed included: six SLCS system operating procedures, four reactor operating procedures that interfaced with the SLCS, three design-bases documents (SLCS, reactor transients, and reactor internals) that discussed the SLCS operations and maintenance, four preventive maintenance and five corrective-maintenance records, seven PEP reports and Action Requests (AR) related to SLCS corrective maintenance and operations.

b. Issues and Findings

There were no findings identified.

.1.3 Surveillance and Testing

a. Inspection Scope

The team reviewed test procedures and recent performance data to verify that the SLCS piping and components met their design and licensing bases. Components and auxiliary systems examined included the following:

- SLCS positive displacement pumps
- SLCS tank heater and level system
- System heat tracing
- Squib valves and circuitry
- System relief and check valves
- System containment isolation valves
- Standby liquid control tank boron concentration
- Instrument air and reactor water cleanup system interfaces
- SLCS alarm and indications (local and control room)
- Local leak rate testing for the SLCS containment isolation check valves

b. Issues and Findings

With one exception, the team found the SLCS piping and components were tested in accordance with the applicable regulatory requirements. The exception involved how the SLC pumps were tested. Specifically, Section 5.6 of Part 6 of the ASME "Operations and Maintenance of Nuclear Power Plants" (OM-1990) Code states that, after pump conditions are as stable as the system permits, each pump shall be run at least two minutes; and, at the end of this time, at least one measurement or observation of each of the quantities required shall be made and recorded. While reviewing procedure ST-O-011-301-2, "Standby Liquid Control Pump Functional Test for IST," the team noted that during the quarterly system performance testing of the SLCS, a two-minute wait was not mandated by the procedure before pump flow and pressure measurements were recorded. There were no other procedures that provided the equivalence of meeting the two-minute wait requirement. Without a two-minute wait period, the team determined that, under certain conditions, a valid assessment of pump performance may not be obtained.

After reviewing pump surveillance test data and interviewing operators, PECO determined that the failure to wait the required stabilization period did not affect system operability, because, during testing, significant pressure fluctuations and flow oscillations had not occurred. Further, recent test results indicated that SLCS pump flow had been significantly higher than the TS minimum value of 43 gpm. Therefore, absent the occurrence of significant pressure fluctuations and flow oscillations during testing, it was unlikely that pump pressure and flow would fall below the TS minimum values. The team independently reviewed prior test results to confirm the validity of this conclusion. Since this procedure inadequacy did not affect system operability, the team determined that the procedure inadequacy was a Green finding. This finding had very low safety significance because the two-minute wait period was only to assure that a valid assessment of the pump performance under normal operating conditions was obtained.

At the end of the inspection period, PECO was in the process of reviewing the SLCS quarterly testing procedures to ensure that the testing requirements contained in OM-6 were satisfied. 10 CFR 50.55a(b)(2)(vii), "Section XI References to OM Part 4, OM Part 6 and OM-Part 10 (Table IWA-1600-1)," states that PECO shall meet the requirements set forth in OM-6 of the ASME Boiler and Pressure Vessel Code. The failure to meet the OM-6 testing requirements when testing the SLCS was a violation. This violation will be considered a non-cited violation in accordance with the NRC enforcement policy. **(NCV 05000277; 05000278/2000-005-01).** This issue was entered into the PECO corrective action program as PEP item number 10011470.

.2 <u>Feedwater System (FWS)</u>

.2.1 <u>Mechanical, Electrical, Instrumentation and Control System Design</u>

a. <u>Inspection Scope</u>

The team reviewed the FWS design and licensing basis documents to determine the system functional requirements during normal and accident conditions. For the documents reviewed, which included the licensee's DBD, reactor feed pump vendor manuals, system modifications and repairs, reactor feed pump turbine speed controls design data, electrical and control design and logic drawings for the digital feedwater control system and the reactor feed pump controls, the team verified that the assumptions were appropriate, that proper engineering methods and engineering standards were used, that the design basis was maintained, and that there were adequate technical bases to support conclusions. Review was performed to determine that: (1) the design basis was in accordance with the licensing commitments and regulatory requirements; (2) the design output documents such as drawings and system calculations and analyses were correct.

The team reviewed the UFSAR to identify the design and licensing basis for the FWS and its interfacing systems. The applicable piping, electrical, instrumentation and control drawings, the logic channel configuration documents and the installed configuration were also reviewed to assess the capability of the system to satisfy the design intent.

The team also performed a walkdown of the FWS in the field and compared it to the design drawings and specifications. Components examined included piping and pipe supports, reactor feed pumps (RFP), feedwater check valves and isolation valves, digital feedwater controls and reactor feed pump turbine (RFPT) speed controls.

b. Issues and Findings

There were no findings identified.

.2.2 Operations and Maintenance

a. Inspection Scope

The team reviewed a number of activities and procedures to verify that the FWS was operated and maintained consistent with the design and licensing bases. The operational readiness and material condition of the FWS was assessed by conducting system walkdowns and reviewing procedures, design documents, component maintenance history records, and system health reports. The team also interviewed licensed and non-licensed operators and engineers. As part of this review, the team evaluated a sample of licensee-identified problems documented in their performance enhancement program (PEP) reports to assess the effectiveness of the licensee's corrective actions.

The documents reviewed included: three system operating and two abnormal operating procedures for the FWS, alarm response procedures for the RFP trip and trouble conditions, six preventive maintenance and six corrective maintenance records, two PEP reports and 19 ARs associated with the RFP operations and corrective maintenance.

b. Observations and Findings

There were no findings identified.

.2.3 Surveillance and Testing

a. Inspection Scope

The team reviewed test procedures and recent performance data, to verify that the FWS piping, components, instrumentation and control met their design functional requirements and licensing bases. The following tests were examined:

- Reactor feed pump turbines (RFPT) mechanical overspeed trip test
- RFPT overspeed trip mechanism trip dump valve test
- RFPT protection systems testing
- RFPT speed control system uninterruptible power supply (UPS) battery condition check
- RFPT high water level trip test
- Feedwater control system stability test
- Feedwater stop valve test
- RFPT oil system accumulator nitrogen pressure tests
- Local leak rate testing for the feedwater system containment isolation check valves

In addition, the team also reviewed the FWS transient response data (real performance data) for seven reactor transients to verify that the feedwater control systems were stable under various transient conditions:

- Unit 2 scram generator lockout, September 30, 1999
- Unit 2 feedwater pump trip, July 13, 1999
- Unit 2 recirculation pump runback, June 7, 1998
- Unit 2 circulating pump trip, January 6, 1998
- Unit 3 scram-shutdown for instrument N2 supply line repair, November 27, 1998
- Unit 3 feedwater pump trip, August 25, 1999
- Unit 3 recirculation pump runback, June 7, 1998

b. Issues and Findings

There were no findings identified.

1R02 Evaluations of Changes, Tests, or Experiments

a. Inspection Scope

The team reviewed administrative control procedures for the 10 CFR 50.59 program to determine their adequacy. Three procedures were reviewed: (1) Procedure LR-CG-13, “Performing 10 CFR 50.59 Reviews,” Revision 3; (2) Procedure LR-CG-13-2, “10 CFR 50.59 Review Determination Checklist,” Revision 3; and (3) Procedure LR-CG-13-3, “10 CFR 50.59 Screening,” Revision 5.

The team reviewed and assessed selected 10 CFR 50.59 Safety Evaluations (SE) representing activities associated with the three cornerstones: initiating events, mitigating systems, and barrier integrity. The reviews were to verify that changes made to the facility or procedures as described in the UFSAR were reviewed and documented in accordance with 10 CFR 50.59, and that the safety issues pertinent to the changes were properly resolved. The team also reviewed 12 screened-out safety evaluations associated with the SLCS and the FWS to verify that the screen-out process was appropriately implemented. In addition, the team reviewed five PEP reports associated with 10 CFR 50.59 issues to ensure that all 10 CFR 50.59 issues were properly resolved and appropriate corrective actions were implemented.

The team interviewed engineering personnel engaged in the preparation and the review of the selected 10 CFR 50.59 safety evaluations. The team conducted a number of meetings with licensee's engineering personnel to resolve questions and observations made during the course of the reviews. The following 10 CFR 50.59 safety evaluations were selected for reviews:

10 CFR 50.59 for ECR 97-000533 - evaluated the changes being made to the description of the RHR/HPSW cross-tie contained in the DBD's and the UFSAR.

10 CFR 50.59 for ECR 98-01931 - evaluated UFSAR and other DBD changes for the PBAPS Fire Safe Shutdown Analysis.

10 CFR 50.59 for ECR 00-00459 - addressed a change in Technical Specification Surveillance Requirements and associated bases regarding the verification testing of Excess Flow Check Valves.

10 CFR 50.59 for SO 12.1.A-2(3) and SO 12.12.3.A-2(3) - evaluates temporary changes to procedures (SO 12.1.A-2(3) and SO 12.12.3.A-2(3) to prevent the RWCU MO-15 (the inboard isolation valve from isolating on a spurious high system flow signal or high system temperature.

10 CFR 50.59 for ECR 99-02764 - supports a Technical Specification and bases change. This change would increase the required completion time for restoring an inoperable Emergency Service Water subsystem from 7 to 14 days.

10 CFR 50.59 for ECR 99-00456 - evaluated the change to the Peach Bottom UFSAR to reflect new ECCS-LOCA peak clad temperature (PCT) values.

10 CFR 50.59 for ECR 98-01931 - evaluated UFSAR and other DBD changes for the PBAPS Fire Safe Shutdown Analysis.

10 CFR 50.59 for NCR 99-02492 - evaluated the interim “Use As Is” disposition for the Unit 2 recirculation pump cover cracks.

10 CFR 50.59 for ECR 99-01814 - evaluates the design change to de-energize alternate control station lights on two panels to maintain the load margin on two safety-related battery supplies.

10 CFR 50.59 for ECR 97-02522 - justified the proposed activities of steady state operation with one Main Steam Line (MSL) isolated.

b. Issues and Findings

There were no findings identified.

## 4. OTHER ACTIVITIES

4OA1 Identification and Resolution of Problems (IP 71152)

a. Inspection Scope

For the SLCS and FWS, the team reviewed the activities for identifying, evaluating, and correcting problems as documented in the licensee's PEP reports which could impact the cornerstone objectives.

b. Issues and Findings

There were no findings identified.

4OA6 Management Meetings

The team presented the inspection results to Mr. G. Johnson, Plant Manager, and other members of licensee management at the conclusion on the inspection on July 20, 2000. The licensee acknowledged the inspection findings presented.

## PARTIAL LIST OF PERSONS CONTACTED

Licensee

| | |
|---|---|
| P. Davidson | Director, Site Engineering |
| A. Hegedus | Manager, I&C Design Engineering |
| G. Johnson | Plant Manager |
| A. Knoll | PSA |
| W. Nelle | Lead Assessor |
| D. Warfel | Senior Manager, Design Engineering |
| D. Wielgoposki | Licensing Engineer |
| A. Winter | Manager, Experience Assessment |

NRC

| | |
|---|---|
| L. Doerflein | Chief, Systems Branch |
| A. McMurtray | Senior Resident Inspector |
| S. Morris | Acting Chief, Electrical Branch |
| J. Richmond | Acting Resident Inspector |

## ITEMS OPENED, CLOSED, AND DISCUSSED

Opened/Closed

| | | |
|---|---|---|
| 05000277; 05000278/2000-005-01 | NCV | IST of SLC pumps not in accordance with ASME Code. |

LIST OF ACRONYMS USED

| | |
|---|---|
| ASME | American Society of Mechanical Engineers |
| AR | Action Request |
| ATWS | Anticipated Transients Without Scram |
| CFR | Code of Federal Regulation |
| DBD | Design Basis Document |
| ECR | Engineering Change Request |
| EQ | Environmental Qualification |
| FWS | Feedwater System |
| gpm | Gallons per Minute |
| LOCA | Loss of Coolant Accident |
| NCV | Non-Cited Violation |
| NRC | Nuclear Regulatory Commission |
| PEP | Performance Enhancement Program |
| RFP | Reactor Feedwater Pump |
| RFPT | Reactor Feedwater Pump Turbine |
| RG | Regulatory Guide |
| SDP | Significancy Determination Process |
| SLCS | Standby Liquid Control System |
| TS | Technical Specification |
| UFSAR | Updated Final Safety Analysis Report |
| UPS | Uninterruptible Power Supply |

**ATTACHMENT 1**

**LIST OF DOCUMENTS REVIEWED**

Engineering Calculations

| | |
|---|---|
| EE-467-1 | Determine SLC Control Transformer Power Requirement, Squib Valve Firing Current, Revision 0 |
| PM-512 | Implications of having the Standby Liquid Tank Temperature between 100$^{O}$ F and 155$^{O}$ F, Revision 0 |
| PM-517 | Maximum Allowable SBLC Storage Tank Solution at “Zero” Tank Level, Revision 0 |
| M-28 | Standby Liquid Control Pumps, Revision 0 |
| ME-208 | Determine New SLC Instrument Setpoints with Enriched Sodium Pentaborate, Revision 1 |
| PI 00120 | Reactor Vessel High Water Level (Level 8) Trip in HPCI and RFPT Loop |
| PE 057 | Reactor Water Level Drop (ECR PB 99-01793 8/10/99) |
| PM 0231 | Determine Prop Setpoint for RFP Low Suction Press Alarm |
| ME 0694 | RPV Water Level Inst Max Run Temperature |
| PM 0620 | Determine Upstream and Downstream Line Pressure for MOV’s |
| 6280-26 | Feedwater From Pump Discharge |

Design Bases Documents

| | |
|---|---|
| P-S-11 | Feedwater System, Peach Bottom Atomic Power Station, Units 2 and 3, Revision 14 |
| P-S-38 | Standby Liquid Control System, Revision 7 |
| P-T-16 | Peach Bottom Atomic Power Station, Unit 2 & 3 Regulatory Guide 1.97 - Post Accident Monitoring, Revision 8 |
| P-T-12 | Design Basis Accidents, Transients, and Events, Revision 4 |
| P-T-18 | Reactor Vessel Internals |

Engineering Change Requests

| | |
|---|---|
| ECR 99-00382 | Standby Liquid or Pipe Hi-Lo Temperature Alarm |
| ECR 99-01041 | SBLC Hi/Lo Temperature Alarm, Unit 3 |
| ECR 98-01953 | Replace Unit 3 Feedwater Computer, Revision 0 |
| ECR 98-01179 | Upgrade Feedwater control system power Supplies-Unit 2, Revision 0 |
| ECR 98-00501 | POT-9091 Could not be Calibrated to within Calibration Sheet SPECS, Revision 1 |
| ECR 00-00178 | Update Feedwater and PMS to Incorporate flow test results, Revision 0 |
| ECR 00-00137 | Inspect/Repair Unit 2 Feedwater Heaters |
| ECR 99-01781 | Inspect/Repair Unit 3 Feedwater Heater Shells |
| ECR 99-00289 | Unit 2 RFPT Lube oil Coolers |
| ECR 00-00821 | Valve Leaks Thru |
| ECR 99-01793 | Change DFCS Tuning to Lessen Level Transient |
| ECR 98-01366 | Final Rework for Previous NCR PB 96-03343 |
| ECR 96-03343 | Repair of Leak at 2C RFPT HP Control Valve Flange |

| | |
|---|---|
| ECR 99-02584 | RFPT Vacuum Sensing Line Supports |
| ECR 99-02241 | Procurement Evaluation Parts for HV-3-06-51127 |
| ECR 99-02584 | Vacuum Sensing Line Support Design |
| ECR 95-04183 | Load changes on 120 Vac inverter backed power panels due to replacement of the reactor feedwater pump Turbine (RFPT) speed governor and associated engineering change ECRs, Revision 2 |

Performance Enhancement Process (PEP) Documents

| | |
|---|---|
| I0007188 | TS 2-11-059C (Standby Liquid Tank HI/LOW Temperature for Unit 2) |
| I0009799 | Functional Failure of Breaker Associated with SLC Pump Operation |
| I0007670 | Locked Valve on SBLC was not Locked Appropriately |
| I0007688 | Oil in 3A SBLC Pump Crankcase has History of High Viscosity |
| I0008776 | SBLC Control Switch Keylock has Key Stuck in Wrong Position |
| I0005405 | Review of System for Generic Implications |
| I0010110 | Repeat Maint-GENIP Pump Failure |
| I0010030 | 2B Reactor Feed Pump Tripped-Broken Sensing Line |
| I0011259 | B2 Feedwater Heater Tube Leak |
| I0010325 | U/2 Main Turbine Generator Trip Caused Reactor Scram |
| I0010377 | Reactor Feed Pump Discharge Valve |
| I0010167 | 3A RFPT Trip Due to Loss of UPS Power to Governor |
| I0008852 | Isolation Valve for 2C Reactor Feedwater Pump (RFP) Suction PT Found Closed |
| I0008414 | 2A RFP was Inadvertently Placed in Manual Operation |

Action Requests

| | |
|---|---|
| A1236465 | 3A Reactor Feedpump Turbine |
| A1079378 | Feedwater Check Valve Loop B Outboard |
| A0974019 | Feedwater Check Valve Loop B Outboard |
| A1049264 | 2C RX Feed Pump Turbine High Pressure Stop Valve |
| A1256975 | Secondary Plant Pressure Vessel Wall Thinning |
| A1248730 | Reactor Scram-Injection of Cold Feedwater |
| A1240725 | RX Power Ascension-Loss of FW Heating |
| A1216838 | FW Heater Shell Rupture |
| A1214111 | LGS LER 1/99/03 Loss of FW Flow |
| A1194613 | FW Heater Shell side Leak at Susquehanna |
| A1238474 | SIL 623 HPCI & RCIC System Peak Pump Discharge |
| A1240496 | SEN 204 Water Chemistry Induced Fuel Leaks |
| A1238339 | SEN 203 Drain Line Rupture-Moisture Separator Reheater |
| A1159503 | INN 98-24 Stem Binding in Turbine Governor Valves |
| A1214210 | SEN 198 Flooding of Steam Lines |
| A1100969 | INN 91-50 Water Hammer Events |
| A0827504 | SIL 452S1R1 Feedwater Flow Element Transmitter |
| A0364653 | SER 2-92 Feedwater Piping Over pressure |
| A1220436 | 2A Reactor Feedpump Turbine Oil Cooler “A” |
| A1257342 | 2A Reactor Feedpump Turbine Oil Cooler “B” |
| A1244125 | RX Feedwater Pump/Turbine “A” |
| A1228157 | Feed Pump Turbine Bearing Oil Pressure Regulator |
| A1225003 | “A” Reactor Feed Pump Bearing Oil Pressure Regulator |

| | |
|---|---|
| A1225444 | 3A Reactor Feed Pump Turbine Tripped |
| A1219424 | RFPT “B” Exh Vacuum Pressure Alarm |
| A1157211 | “B” RFPT Lockout Valve Normal Position Pressure |
| A1196528 | “B” RFPT Lockout Valve Locked Out-Pressure |
| A1196529 | “C” RFPT Lockout Valve Locked Out-Pressure |
| A1182992 | 3C Reactor Feed pump Turbine |
| A1180778 | “C” Reactor Feed Pump |
| A1179794 | “C” Reactor Feed Pump |
| A1141553 | “C” RFPT Lockout Valve Locked Out |
| A1150883 | 2C Reactor Feed Pump Turbine |
| A1133366 | 2A Reactor Feed Pump Turbine |
| A1119321 | “A” Reactor Feed Pump Failure to Trip |
| A1105963 | RFPT B AUX Panel |
| A1094455 | SV-7 Failed with 3B RFP on the Jack |
| A1270905 | Noncompliances Found during RT-O-0-100-505-2, Emergency Operating Procedure Tool Inventory |
| A0132342 | PI-2-11-053, SLC Pump Discharge Header Pressure |
| A0354761 | PI-2-11-052, SLC Pump Discharge Header Pressure |
| A0354772 | RV-2-11-39A, Remove/Bench Test Relief Valve |
| A0037689 | RV-2-11-39B, Remove/Bench Test Relief Valve |
| A1101190 | SLC Tank Hi/Low Temperature |
| A1188862 | Sodium Pentaborate Buildup Upon SLC Pumps |
| A1223203 | Sodium Pentaborate Buildup on U/2 B SLC Pump Plunger/pistons |
| A1242579 | Sodium Pentaborate Buildup on 2BP040 |
| A1124302 | 3SP040: Incorrect Oil in Pump Crankcase |
| A0895783 | 3B RFPT Speed Control Hydraulic Actuator Inspection |
| A1192338 | 3B RFPT Minor Inspection (PM) |
| A1146470 | Perform Functional Check Loop to Alarm |
| A1104229 | Calibrate LIC-9091 and Loop |
| A0378320 | 2BP001, Perform Pump Internal Inspection |
| A0940713 | 2BP001, RFP Minor Inspection |
| A1220436 | 2A RFPT Oil Cooler “A” |
| A1257342 | 2A RFPT Oil Cooler “B” |
| A1228157 | RFPT Bearing Oil Pressure Regulator |
| A1225444 | 3A RFPT Tripped |
| A1196528 | “B” RFPT Lockout Valve Locked Out-Pressure |

Station Procedures

| | |
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| AO 6D.1-2 RFP | Shutdown with Failed Minimum Flow Valve |
| AO 5.4-3 | Condensate System Draining |
| ARC 201 20C206L H-4 | B RFPT Trip |
| ARC 201 20C206R G-1 | A RFPT Controller Trouble |
| ARC-211 20C205R J-2 | SLC Tank HI-LO Level, Revision 5 |
| ARC-211 20C205R J-3 | Standby Liquid or Pipe Hi-Lo Temperature, Revision 4 |
| ARC-211 20C205R J-4 | Standby Liquid Tank Heater Power Off, Revision 1 |
| ARC-211 20C205R H-3 | Standby Liquid Squib Valve Loss of Continuity, Revision 2 |
| RT-I-006-230-3 | Feedwater Control System Stability/Response Test, Revision 4 |
| RT-I-006-710-2 | RFPT Speed Control System UPS Battery Condition Check, |

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| | Revision 0 |
| RT-M-080-611-2 | Standby Liquid Control System Pressure Test, Revision 1 |
| RT M-01E-640-2 | 3A, 4A, 5A Feedwater Heater Shell-side Pressurization Check |
| RT-N-06D-221-2 | A Reactor Feedwater Pump Turbine (RFPT) Mechanical Overspeed Trip Test, Revision 3 |
| RT-O-0-100-505-2 | Emergency Operating Procedure Tool Inventory, Revision 8 |
| RT-O-06D-261-2 | A RFPT Protection Systems Testing, Revision 4 |
| RT-O-06D-400-3 Rev 3 | Trip Mechanism Exercise Test for RFPT A, B & C |
| RT-O-006-450-3 Rev 0 | Feedwater Stop Valve MO-3-02-029A Alternative Control Test |
| RT-O-06B-2SO-2 Rev 4 | Pressure Check of RFPT Oil System Accumulators |
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| COL 11.1.A-3 | Standby Liquid Control System (Lineup), Revision 11 |
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| SO 11.7.B-2 | Standby Liquid Control Accumulator Charging, Revision 1 |
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| SO 6C.1.C-2 | Startup of Second or Third RFP |
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| ST-C-095-802-2 | Unit 2 Tank Chemical Addition, Revision 2 |
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| ST-I-006-101-2 | “A” Reactor Feedwater Turbine High Water Level Trip Logic System Functional Test, Revision 3 |
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| E-29 | Single Line Diagram, Instrumentation & Un-interruptible ac System-Unit 3 Revision 56 |
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## ATTACHMENT 2

## NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

**Reactor Safety**

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

**Radiation Safety**

- Occupational
- Public

**Safeguards**

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.