

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.

Mr. G. Rainey

-2-

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

Sincerely,

/RA/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:

J. Hagan, Senior Vice President, Nuclear Operations  
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  
A. F. Kirby, III, External Operations - Delmarva Power & Light Co.  
A. A. Winter, Manager, Experience Assessment  
J. W. Durham, Sr., Senior Vice President and General Counsel  
H. C. Kresge, Manager, External Operations, Connectiv  
N. J. Sproul, Manager, Financial Control & Co-Owner Affairs, Connectiv  
R. McLean, Power Plant Siting, Nuclear Evaluations  
D. Levin, Acting Secretary of Harford County Council  
R. Ochs, Maryland Safe Energy Coalition  
J. H. Walter, Chief Engineer, Public Service Commission of Maryland  
Mr. & Mrs. Dennis Hiebert, Peach Bottom Alliance  
Mr. & Mrs. Kip Adams  
Commonwealth of Pennsylvania  
State of Maryland  
TMI - Alert (TMIA)

Mr. G. Rainey

-3-

Distribution w/encl:

Region I Docket Room (with concurrences)

NRC Resident Inspector

H. Miller, RA/J. Wiggins, DRA (1)

C. Cowgill, DRP

D. Florek, DRP

D. Cullison, DRP

C. O'Daniell, DRP

Distribution w/encl: (Via E-Mail)

J. Shea, OEDO

E. Adensam, NRR

J. Clifford, NRR

B. Buckley, NRR

W. Scott, NRR

C. See, NRR

Inspection Program Branch, NRR (IPAS)

DOCUMENT NAME: G:\ELECTRICAL\CHEUNG\PB00-05REV2.WPD

After declaring this document "An Official Agency Record" it **will** be released to the Public.

To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure "E" = Copy with attachment/enclosure "N" = No copy

OFFICE	RI/DRS		RI/DRS		RI/DRP		RI/DRS			
NAME	LCheung/LSC		WRuland/RLF for:		CCowgil/cjc		WLanning/JRW for:			
DATE	08/11/00		08/11/00		08/14/00		08/22/00		08/ /00	

OFFICIAL RECORD COPY

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  
Correspondence Control Desk  
P.O. Box 195  
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  
R. Bhatia, Reactor Inspector, DRS (part time)  
T. Burns, Reactor Inspector, DRS  
J. Carrasco, Reactor Inspector, DRS  
L. James, Reactor Engineer, DRS  
K. Kolaczyk, Reactor Inspector, DRS  
A. Smith, Reactor Inspector (Trainee) , DRS  
K. Young, Reactor Inspector, DRS

Approved by: William H. Ruland, Chief  
Electrical Branch  
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)

## TABLE OF CONTENTS

	PAGE
1. REACTOR SAFETY .....	1
1R21 Safety System Design and Performance Capability .....	1
.1 Standby Liquid Control System (SLCS) .....	1
.1.1 Design - Mechanical, Electrical, and Instrumentation and Controls .....	1
.1.2 Operations and Maintenance .....	1
.1.3 Surveillance and Testing .....	2
.2 Feedwater System (FWS) .....	3
.2.1 Mechanical, Electrical, Instrumentation and Control System Design .....	3
.2.2 Operations and Maintenance .....	4
.2.3 Surveillance and Testing .....	4
1R02 Evaluations of Changes, Tests, or Experiments .....	5
4. OTHER ACTIVITIES .....	7
4OA1 Identification and Resolution of Problems .....	7
4OA6 Management Meetings .....	7
PARTIAL LIST OF PERSONS CONTACTED .....	8
ITEMS OPENED, CLOSED, AND DISCUSSED .....	8
LIST OF ACRONYMS USED .....	9
ATTACHMENT 1 .....	10
ATTACHMENT 2 .....	15

## 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 Feedwater System (FWS)

### .2.1 Mechanical, Electrical, Instrumentation and Control System Design

#### a. Inspection Scope

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° F and 155° 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

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,

	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
RT-R-59C-500-2	Checkout of Nuclear Steam Supply System Computer Calculation of Core Thermal Power
SI2L-11-45-XXCE	Calibration of SBLC System Level Instrumentation, Revision 3
SI3L-11-45-XXCE	Cal. Check of SBLC System Level Instrumentation, Revision 2
COL 11.1.A-3	Standby Liquid Control System (Lineup), Revision 11
SO 11.1.A-2	SBLC System Setup for Operation, Revision 8
SO 11.1.B-3	Standby Liquid Control System Initiation, Revision 3
SO 11.7.A-2	SBLC Chemical Makeup, Revision 9
SO 11.7.B-2	Standby Liquid Control Accumulator Charging, Revision 1
SO 11.8.A-2	Standby Liquid Control System Routine Inspection, Revision 7
SO 6C01.A-2	"C" Reactor Feedwater Pump (RFP) Startup with Vessel Level Control Established through Air Operated Valve 8091
SO 6C.1.D-3	Reactor Feedwater Automatic Level Control
SO 6C.1.C-2	Startup of Second or Third RFP
ST-O-O11-301-2	Standby Liquid Control Pump Functional Test For IST, Revision 13
ST-O-O11-611-2	Standby Liquid Control System Piping Pressure Test, Revision 0
ST-O-O11-612-2	Standby Liquid Control System Piping Hydro Test Inspection, Revision 3
ST-C-095-801-3	SBLC Tank Boron Solution Analysis, Revision 9
ST-C-095-802-2	Unit 2 Tank Chemical Addition, Revision 2
ST-C-095-802-3	Unit 3 Tank Chemical Addition, Revision 1
ST-I-006-101-2	"A" Reactor Feedwater Turbine High Water Level Trip Logic System Functional Test, Revision 3
ST-O-011-350-2	SBLC Explosive Valve Charge Continuity Check and Valve Position Verification, Revision 0
ST-O-011-350-3	SBLC Explosive Valve Charge Continuity Check and Valve Position Verification, Revision 0
ST-O-011-400-2	SBLC System "A" Loop Injection, Revision 2
ST-O-011-405-2	SBLC System "B" Loop Injection, Revision 3
ST-O-011-405-3	Standby Liquid Control System B Loop Injection Test, Revision 4
ST-X-011-400-3	SBLC System Injection Test, Revision 0
T-101	Reactor Pressure Vessel Control Procedure and Bases, Revision 22
T-117	Level/Power Control Procedure and Bases
T-211-2	CRD System Nonenriched Boric Acid and Borax Injection, Revision 6
T-210-2	CRD System SBLC Injection, Revision 4

T-210-3	RWCU System SBLC Injection, Revision 5
T-212-3	RWCU System SBLC Injection, Revision 5
T-244-2	Alternate Injection Using the SBLC Test Tank, Revision 4
T-244-3	Alternate Injection Using the SBLC Test Tank, Revision 4

### Design Drawings

6280-M1-DD-3-6	Process Diagram, Standby Liquid Control System, Revision 3
6280-M-358	P&I Diagram, Standby Liquid Control System, Sheets 1 and 2, Revision 39
6280-M-308	P& I Diagram, Feedwater & Feed Pumps, Sheet 3, Revision 53
E-29	Single Line Diagram, Instrumentation & Un-interruptible ac System-Unit 3 Revision 56
E-1700	Single Line Meter and Relay, Revision 36
E-7	Single Line Meter and Relay Diagram 13.8 kV AUX Power System-Unit 3, Revision 13
6280-NE-226-41	RFPT GOV Wiring Diagram, Sheet 20A, Revision 0
6280-M-1-S-25,	Electrical Schematic Diagram Feedwater Control System, Sheets 1 through 10
M-1-S-46	Electrical Schematic Diagram Standby Liquid Control System, Sheets 1 through 6

## 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:

<b>Reactor Safety</b>	<b>Radiation Safety</b>	<b>Safeguards</b>
<ul style="list-style-type: none"> <li>● Initiating Events</li> <li>● Mitigating Systems</li> <li>● Barrier Integrity</li> <li>● Emergency Preparedness</li> </ul>	<ul style="list-style-type: none"> <li>● Occupational</li> <li>● Public</li> </ul>	<ul style="list-style-type: none"> <li>● Physical Protection</li> </ul>

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>.