January 25, 2006

Mr. Christopher M. Crane
President and CNO
Exelon Nuclear
Exelon Generation Company, LLC
200 Exelon Way KSA 3-E
Kennett Square, PA 19348

SUBJECT: PEACH BOTTOM ATOMIC POWER STATION - NRC INTEGRATED INSPECTION REPORT 05000277/2005005 AND 05000278/2005005

Dear Mr. Crane:

On December 31, 2005, the United States Nuclear Regulatory Commission (NRC) completed an inspection at your Peach Bottom Atomic Power Station Units 2 and 3. The enclosed inspection report documents the inspection results, which were discussed on January 13, 2006, with Mr. R. Braun and other members of your staff.

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

The report documents two NRC-identified findings of very low safety significance (Green). Both of these findings were determined to involve violations of NRC requirements. However, because of the very low safety significance and because they were entered into your corrective action program, the NRC is treating these two findings as non-cited violations (NCVs) consistent with Section VI.A.1 of the NRC's Enforcement Policy. If you contest any NCVs in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at Peach Bottom.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosures, 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 the NRC's document system (ADAMS). ADAMS is accessible from the NRC Website at http://www.nrc.gov/reading-rm/adams.html (The Public Electronic Reading Room).

Sincerely,

/RA/

James Trapp, Chief
Reactor Projects Branch 4
Division of Reactor Projects

Docket Nos.: 50-277, 50-278
License Nos.: DPR-44, DPR-56

Enclosure: Inspection Report 05000277/2005005 and 05000278/2005005
w/Attachment: Supplemental Information

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U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket Nos.:	50-277, 50-278
License Nos.:	DPR-44, DPR-56
Report No.:	05000277/2005005 and 05000278/2005005
Licensee:	Exelon Generation Company, LLC
Facility:	Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3
Location:	Delta, Pennsylvania
Dates:	October 1 through December 31, 2005
Inspectors:	F. Bower, Senior Resident Inspector D. Schroeder, Resident Inspector F. Bonnett, Senior Reactor Analyst, NRR J. Caruso, Senior Operations Engineer, DRS J. Josey, Reactor Inspector, DRS J. Krafty, Acting Resident Inspector P. Kaufman, Senior Reactor Inspector, DRS R. Nimitz, Senior Health Physicist, DRS D. Silk, Sr. Emergency Preparedness Inspector, DRS
Approved by:	James Trapp, Chief Reactor Projects Branch 4 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000277/2005-005, 05000278/2005-005; 10/01/2005 - 12/31/2005; Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3; Post-Maintenance Testing, Access Control to Radiologically Significant Areas.

The report covered a 13-week period of inspection by the resident inspectors and announced inspections by specialist inspectors including: a regional health physicist, a regional emergency preparedness inspector, a regional operations inspector, regional reactor inspectors and a senior reactor analyst from the Office of Nuclear Reactor Regulation. Two Green findings, both of which were non-cited violations (NCVs), were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be 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. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a Green non-cited violation (NCV) of 10 CFR Part 50, Appendix B, Criterion XI, Test Control, for not adequately testing the high pressure service water (HPSW) sub-system following a valve replacement The post-maintenance test did not account for the known degraded condition of the 3B residual heat removal (RHR) heat exchanger HPSW outlet throttle valve. The leaking valve allowed unmeasured bypassing flow to occur while recording the sub-system flow of the 3D HPSW loop. PBAPS entered this performance deficiency into their corrective action program (CAP). Planned corrective actions include revising the surveillance test procedure and re-sizing the orifice plate that is located downstream of the 3D RHR HX.

  The finding is greater than minor because it is associated with the procedure quality attribute of the Mitigating Systems Cornerstone and affects the cornerstone objective to ensure the capability of systems that respond to initiating events to prevent undesirable consequences. Improper test control on two occasions following the design change to MO-3-10-89D and the downstream orifice plates, did not identify that HPSW flow through this loop was below the design basis flow of 4500 gpm. The finding was determined to be a Green finding of very low safety significance using Phase 1 of the SDP, since the finding is a qualification deficiency confirmed not to result in a loss of function. The cause of this finding is related to the cross-cutting element of problem identification and resolution. (Section 1R19)

Cornerstone: **Occupational Radiation Safety**

- Green. The inspectors identified an NCV of Technical Specification 5.4 and Regulatory Guide 1.33, 1972, associated with failure to follow initial containment access radiation protection program procedures. Specifically, on September 19,

2005, personnel made an initial entry into the Unit 3 reactor drywell, after reactor shut down, and did not collect and analyze a drywell radiation monitoring system (RMS) sample for airborne particulate and iodine, prior to the entry, as required by Radiation Protection Program Procedure HP-315, Initial Drywell Entry, Rev. 12. PBAPS entered this performance deficiency into their CAP to develop corrective actions for resolution.

The finding is greater than minor, in that, it is associated with the Occupational Radiation Safety Cornerstone attribute of exposure control and affects the cornerstone objective. Specifically, PBAPS could not ensure adequate protection of worker health and safety from exposure to airborne radioactive material. The finding is suitable for SDP review, in that there was a potential for a significantly greater unplanned, unintended dose if airborne radioactivity concentrations had been significantly elevated. Using the Occupational Radiation Safety Significance Determination Process, the finding is of very low safety significance (Green), in that it did not involve an ALARA finding, did not result in an overexposure, did not result in a substantial potential for an overexposure, and did not compromise the ability to assess dose. The two individuals who made the initial entry did not sustain any significant dose. (Section 2OS1)

B. Licensee-Identified Violation

A violation of very low safety significance (Green), which was identified by the licensee was 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 2 began the inspection period at approximately 100 percent rated thermal power (RTP), where it remained, except for brief periods for the conduct of planned testing and rod pattern adjustments.

Unit 3 began the inspection period shutdown for its 15th refueling outage (3R15). The unit was restarted on October 16, 2005 and achieved full RTP on October 19, 2005. On October 20, 2005, power was reduced to 88 percent in response to 3A recirculation pump seal cavity high temperature. Recirculation pump speed was raised in a graduated manner to address the problem and the unit returned to full power on October 21, 2005. Power was reduced to 82 percent on October 24, 2005, in response to a rod drift. The unit returned to full power on October 24, 2005. On October 27, 2005, to support maintenance and testing of five control rods, power maneuvers were made to as low as 84 percent before being returned to full power on October 29, 2005. On October 30, 2005, power was reduced to 85 percent in response to main turbine number one control valve oscillations. Power was further reduced to as low as 70 percent to support troubleshooting and maintenance before returning to full on power on November 1, 2005, where it remained, except for brief periods for the conduct of planned testing and rod pattern adjustments.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01 - 2 Samples)

b. Inspection Scope

.1 Evaluate Readiness for Winter Seasonal Susceptibilities

The inspectors performed a detailed review of Peach Bottom Atomic Power Station's (PBAPS) and Exelon's written procedures for winter readiness and low temperatures to evaluate PBAPS's implementation of the adverse weather preparation procedures and compensatory measures for the affected conditions before the onset of and during adverse weather conditions. The inspectors selected, for inspection, the intake structure and traveling screens supporting both units, Units 2 and 3, which constituted two samples that included the following three systems on both units:

- Circulating water
- Emergency service water
- High pressure service water

Documents reviewed to verify that the selected systems would remain functional when challenged by adverse weather included the Updated Final Safety Analysis Report (UFSAR), Technical Specifications (TSs), and selected plant documents. The review also verified that plant features and procedures for operation and continued availability of the ultimate heat sink, the Conowingo Pond. The three plant systems listed above

were walked down to verify the physical condition of the cold weather protection features and to verify features, such as, space heaters and weatherized enclosures are monitored sufficiently to ensure they support operability of the system, structure, or component (SSC) they protect. The inspectors also verified that operator actions defined in the licensee's adverse weather procedures are adequate to maintain readiness of essential systems. Documents, procedures and drawings reviewed during the inspection are listed in the attachment.

1R04 Equipment Alignment (71111.04 - 3 Samples)

a. Inspection Scope

The inspectors performed three partial system walkdowns during this inspection period to verify system and component alignment and to note any discrepancies that could impact system operability. The partial walkdowns included verification of the alignment of selected portions of redundant or backup systems and risk-significant systems that were recently realigned following an extended system outage, maintenance, modification, or testing. The inspectors reviewed selected valve positions, electrical power availability, and the general condition of major system components. The partial walkdowns included the following systems:

- Emergency diesel generator (EDG) during the 343 startup power source outage
- E1, E2, E4 EDGs while E-3 EDG was out-of-service
- E1, E3, E4 EDGs while E-2 EDG was out-of-service

1R05 Fire Protection (71111.05 - 7 Samples)

.1 Fire Protection - Tours

a. Inspection Scope

The inspectors reviewed PBAPS's Fire Protection Plan, Technical Requirements Manual, and the respective pre-fire action plan procedures to determine the required fire protection design features, fire area boundaries, and combustible loading requirements for the areas examined during this inspection. The fire risk analysis was reviewed to gain risk insights regarding the areas selected for inspection. The inspectors then performed walkdowns of the following areas to assess the material condition of active and passive fire protection systems and features. The inspection was also performed to verify the adequacy of the control of transient combustible material and ignition sources, the condition of manual firefighting equipment, fire barriers, and the status of any related compensatory measures. The following seven fire areas were reviewed for impaired fire protection features:

- E-2 and E-3 Diesel Generator Rooms (Fire Zone 132)
- Emergency Cooling Tower (Fire Zone 136)
- Recombiner Building, Elevation 135' (Fire Zone 158)
- Recombiner Building, Elevation 157' (Fire Zone 159)

- Unit 2 Refuel Floor, Elevation 234' (Fire Zone 57)
- Unit 2 Reactor Building High Pressure Coolant Injection (HPCI) Room, Elevation 88' (Fire Zone 59)
- E-1 and E-4 Diesel Generator Rooms (Fire Zone 132)

The inspectors verified that housekeeping issues (IR 397211) noted during these walkdown inspections were entered into the CAP.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06 - 2 Internal Samples)

.1 Internal Flooding

a. Inspection Scope

To select risk-important plant design features intended to protect the plant and its safety-related equipment from internal flooding events, the inspectors reviewed the Peach Bottom Level 1 Probabilistic Safety Assessment (PSA), 2002 update. The internal flood analysis for the Unit 2 and 3 reactor building closed cooling water system (RBCCW) pump rooms was selected as samples for detailed inspection. The inspectors walked down the Unit s 2 and 3 RBCCW pump rooms. The walkdown was conducted to verify internal flooding design features were as described in UFSAR - Appendix J, Section J.3.3.4, Station-Site Flood Protection Studies Statement and the PSA. This inspection activity represented two internal flooding samples.

b. Findings

No findings of significance were identified.

1R08 In-service Inspection (ISI) (71111.08 - 7 Samples)

a. Inspection Scope

The inspectors observed in-process non-destructive examination (NDE) activities and reviewed documentation of NDE and repair/replacement activities. The sample selection was based on the inspection procedure objectives and risk priority of those components and systems where degradation could result in a significant increase in risk of core damage. The direct observations and documentation reviews were performed to verify activities were performed in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section IX and XI requirements. The inspector reviewed the following selected sample of in-service inspection (ISI) examination reports initiated to document the performance and record results of ISI examinations completed during the 15th Unit 3 refueling outage (3R15) and the previous

Unit 3 refueling outage, 3R14. Also, the inspectors evaluated PBAPS's effectiveness in resolving relevant indications identified during ISI activities.

- The inspectors reviewed selected documentation, including results of ultrasonic testing (UT) of the Unit 3 H3 and H4 core shroud circumferential welds. The core shroud wall is 2 inches thick. The licensee conducted a UT examination of a known H4 weld flaw (crack) located from 141E-148E, a span of 7E and 1.39 inches deep which had been monitored by interior wall UT examinations during outages in 1995 and 1999. A visual examinations of the core shroud exterior wall at this H4 flaw location did not confirm the presence of a through-wall crack. PBAPS conservatively assumed the H4 weld flaw to be through-wall for evaluation and analysis purposes. The inspectors reviewed the evaluation of the Unit 3 shroud examination results for welds H3, H4 and V3 performed by Structural Integrity Associates. The evaluation concluded that no repairs were needed on the H4 weld location during P3R15 refueling outage. The inspectors verified that the licensee was appropriately implementing the core shroud inspection and flaw evaluation guidelines in Boiling Water Reactor Vessel Internals Project (BWRVIP)-76.

- The inspectors reviewed a videotape of a remote visual examination of the jet pump 2-AD-3B structural welds. The review was conducted to assess and evaluate any changes to these welds since the previous inspection.

- The inspectors examined Exelon's evaluation and disposition for continued operation without repair or rework of non-conforming conditions identified by Issue Report (IR) 00379330 and GE INR-PB3R15-05-02 during ISI activities. The IR documented a large dent on a Unit 3 core spray upper elbow located inside the reactor vessel that was observed during the visual inspection (IVVI). Review of previous core spray videotapes revealed that the dent was also visible in a 3R11 (1997) video clip.

- The inspectors reviewed welding on pressure boundary ASME Class 1 and 2 piping systems and NDE activities associated with: (1) the residual heat removal (RHR) injection check valve (AO-3-10-046B) equalizing line socket-weld; and, (2) the high pressure coolant injection (HPCI) motor operated valve (MO-3-23-014), ASME Class 2 component. The inspectors directly observed baseline UT examination and data recording of the results for the 10" pipe to valve welds FW 23-0-49 and FW 23-0-50 for the replacement of MO-3-23-014. In addition, the inspectors reviewed the radiographs of these same two HPCI welds taken a day prior to the UT examinations.

- Increasing unidentified reactor coolant system leakage was noted on December 28, 2003 (A1448564). During a drywell walkdown on September 20, 2005, to investigate the source of the unidentified leakage, PBAPS personnel observed a weld on a small-bore pipe coupling leaking steam and water. The leak was through the lower socket-welded joint of a 1" pipe coupling, for the above seat equalizing line connection to the RHR injection check valve

(AO-3-10-046A). Metallurgical analysis of the failed socket weld was performed by BWXT Services in Virginia. The analysis showed that the weld failure resulted from crack initiation and subsequent crack propagation. The crack initiator was a significant lack of fusion at the weld root that extended approximately 120 degrees around the fillet weld (a weld flaw). The crack propagation was caused by low cycle fatigue and transgranular stress corrosion cracking (TGSCC). PBAPS replaced this small-bore pipe coupling and the flawed socket weld during 3R15. As part of the extent of condition per IR 375299, six similar 1" socket welds in three couplings were UT examined. The inspectors observed the UT examination of four of these socket welds.

- The inspectors reviewed UT examination procedure, GE-UT-112, "Procedure for Manual Ultrasonic Examination of Socket Welds." The procedure was used during this 3R15 at PBAPS for the first time. The inspectors observed the ISI technicians being tested on mock-up socket welds with flaws. The inspectors also observed the ISI technicians performing calibrations according to GE-UT-112 before entering the drywell for the examination

These reviews were conducted to verify that activities, indications and defects were being properly assessed, evaluated, and dispositioned in accordance with the applicable ASME Sections IX and XI code requirements and the BWRVIP guidelines. The reviews were also conducted to verify the effectiveness of Exelon's program for monitoring degradation of risk significant structures and components.

The inspectors also discussed component and piping degradation operating experience (OE) issues with various engineering personnel. Specifically, the inspectors reviewed how PBAPS evaluated and assessed failed bolts in torus tee-quencher at Edwin I. Hatch Nuclear Power Plant. PBAPS reviewed the operating experience and performed an evaluation as appropriate with support from GE Nuclear Energy. The evaluation showed that the tee-quencher supports at Peach Bottom Unit 3 were made of stainless steel and of welded joint type. As a result, the components are not subject to the type of degradation experienced at plant Hatch.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11 - 2 Samples)

.1 Resident Inspector Quarterly Review

a. Inspection Scope

On November 2, 2005, the inspectors observed one training crew during licensed operator requalification training to verify that operator performance was adequate and that evaluators were identifying and documenting crew performance problems. The inspectors also verified that performance errors were discussed in the crew's

post-scenario critiques. The inspectors focused on the control room supervisor's satisfactory completion of critical tasks, including proper and timely identification and classification of emergencies. The inspectors also evaluated whether the operators adhered to the emergency operating procedures. The inspectors discussed the training, simulator scenarios, and critiques with the operators, shift supervision, and the training instructors. The scenarios observed are listed below:

- SE-11, Loss of Offsite Power
- T-101, Reactor Pressure Vessel (RPV) Control
- T-102, Containment Spray

b. Findings

No findings of significance were identified.

.2 Biennial Review of Licensed Operator Requalification

a. Inspection Scope

As a follow-up to last year's Limerick and Peach Bottom Dual Site Limited to Refueling Senior Reactor Operator (LSRO) Licensed Operator Requalification Program Inspection, an additional limited scope follow-up inspection was performed since recent LSRO inspection observations were performed only at Limerick and this is a dual site license.

The following inspection activities were performed using NUREG-1021, Rev. 9, "Operator Licensing Examination Standards for Power Reactors," Inspection Procedure Attachment 71111.11, "Licensed Operator Requalification Program," and NRC Manual Chapter 0609, Appendix I, "Operator Requalification Human Performance Significance Determination Process (SDP)," as acceptance criteria.

The inspector reviewed documentation of operating history since the last requalification program inspection. The inspector also discussed facility operating events with the resident staff. Documents reviewed included NRC inspection reports and PBAPS condition reports (CRs) that involved human performance issues for licensed operators to ensure that operational events were not indicative of possible training deficiencies.

The inspector reviewed one set of five job performance measures (JPMs) administered during this current exam cycle and a written exam administered at Peach Bottom (i.e., during the 2004 LSRO exam cycle) to ensure the quality of these exams met or exceeded the criteria established in the Examination Standards and 10 CFR 55.59.

During this inspection, the inspector observed the administration of operating examinations to five licensed LSROs on the refueling floor at Peach Bottom. The operating examinations consisted of one set of five JPMs administered to each individual.

On December 22, 2005, the inspector conducted an in-office review of PBAPS requalification exam results. These results included the annual operating tests administered this year. The inspection assessed whether pass rates were consistent with the guidance of NRC Manual Chapter 0609, Appendix I, "Operator Requalification Human Performance Significance Determination Process (SDP)." The inspector verified that:

- Individual failure rate on the walkthrough test (JPMs) was less than or equal to 20 percent. (Failure rate was 0.0 percent)
- More than 75 percent of the individuals passed all portions of the exam (100.0 percent of the individuals passed all portions of the exam).

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12 - 1 Sample)

.1 Routine Maintenance Effectiveness Issues

a. Inspection Scope

The inspectors reviewed the follow-up actions for issues to assess the effectiveness of PBAPS's maintenance activities. The review included items such as: (1) appropriate 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) classification and reclassification in accordance with 10 CFR 50.65(a)(1) or (a)(2); and (8) appropriateness of performance criteria for structures, systems, and components (SSCs)/functions classified as (a)(2) and/or appropriateness and adequacy of goals and corrective actions for SSCs/functions classified as (a)(1). The item reviewed included the following:

- Main Condenser Mechanical Vacuum Pumps (IR 385904)

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13 - 7 Samples)

a. Inspection Scope

The inspectors reviewed PBAPS's planning and risk management actions for planned and emergent work activities to assess PBAPS's management of overall plant risk. The activities selected were based on plant maintenance schedules and systems that contributed to risk. As applicable, the inspectors reviewed PBAPS's probabilistic safety

assessment risk evaluation results forms. The inspectors compared the risk assessment results and the risk management actions against the requirements of 10 CFR 50.65(a)(4) and the information in Regulatory Guide 1.182, Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants, and Procedure WC-AA-101, On-line Work Control Process. The inspectors verified that risk assessments were performed when required and appropriate risk management actions were identified. The inspectors also reviewed control room operating logs, walked down protected equipment and maintenance locations, and interviewed personnel. These reviews were performed to determine whether PBAPS properly assessed and managed plant risk and performed activities in accordance with applicable Technical Specifications (TS) and work control requirements. The following seven planned and emergent work order (WO) activities were reviewed:

- #1 Turbine Control Valve Troubleshooting (TCV) (WO C0215576)
- MO-518 Valve Replacement Preparation (WO C0204514)
- Main Steam Isolation Valve (MSIV) 80A Closure Input to RPS Trip Troubleshooting (WO C0215714)
- Cardox Injection Test for E-2 Emergency Diesel Generator (WO R0971464)
- Unit 2 Condenser Vacuum Switch Calibration with 2A Electrohydraulic Control (EHC) Pump Out-of-Service (WO R1002296-9)
- Restoration of Unit 3 Control Rod 54-19 to Position 48 (A1544131) (IR 325408)
- E-2 EDG Exhaust Manifold Gasket Replacement (WO C0215907)

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15 - 5 Samples)

b. Inspection Scope

The inspectors reviewed five issues that were selected based on risk insights to assess the technical adequacy of the evaluations, the use and control of compensatory measures, and compliance with the licensing and design bases. As applicable, associated adverse condition monitoring (ACM) plans, engineering technical evaluations (TE) and operational and technical decision making (OTDM) documents were also reviewed. The inspectors verified these processes were performed in accordance with the applicable procedures listed in the attachment. The inspectors used the Technical Specifications, Technical Requirements Manuals, the Updated Final Safety Analysis Report, and associated Design Basis Documents as references during these reviews. The issues reviewed included:

- Emergency Service Water (ESW), MO-518 Valve Saddle Welding with System in Service (ECR 05-159)
- Unit 3 Average Power Range Monitor #3 Test Feature Partially Non-Functional (IR 393641)
- Emergency Cooling Tower, MO-2804B, Will Not Stroke Open (IR 391934)

- Evaluate Uncoupled Unit 3 Control Rod 30-15 Following Over Travel (IR 385431)
- E3 Diesel Generator Fuel Oil Transfer Pump Developed Head - Low (AR A1541353)

b. Findings

No findings of significance were identified.

1R16 Operator Work-Arounds (71111.16 - 2 Samples)

c. Inspection Scope

.1 Biannual Review of the Cumulative Effects of Operator Workarounds

The inspectors reviewed and discussed with PBAPS staff the cumulative effects of operator work-arounds and equipment deficiencies on the reliability, availability, and potential for misoperation of systems at both units. The inspectors evaluated the effects of identified items on the ability of operators to respond in a correct and timely manner to plant transients and accidents. The inspectors also reviewed deficiencies to determine if any items complicating the operators' ability to implement emergency operating procedures had not been identified by PBAPS as an operator work-around. The inspectors reviewed Exelon Administrative Procedure OP-AA-102-103, "Operator Work-Around Program," for implementation at the site. This inspection activity represented one cumulative sample.

.2 Selected Operator Workaround

The inspectors reviewed the following one selected burden to operators to determine if the functional capability of the system or human reliability in responding to an initiating event is affected by the operator workaround or operator challenge. The inspectors referred to the definitions and standards identified in Exelon Administrative Procedure OP-AA-102-103, Operator Work-Around Program. Specifically, the review was conducted to evaluate the effect of the operator burden on the operator's ability to implement abnormal or emergency operating procedures. The following one sample selected was for an operator challenge that was identified through other inspection activities:

- The inspectors performed a focused review of the challenge presented by oscillations in the speed of the 3A reactor recirculation pump. This problem first appeared in 2002 and was believed to have been corrected by maintenance conducted during the 3R14 (AR A1374347)and 3R15 (AR A1439041) refueling outages, that included locking in place of an abandoned scoop tube in the motor-generator set fluid drive. The inspectors noted that PBAPS has recognized the issue as a reactivity management concern and an impact on the ability of the operating shifts to maintain Unit 3 generation within the desired band.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19 - 7 Samples)

a. Inspection Scope

The inspectors observed portions of post-maintenance testing activities in the field and reviewed selected test data at the job site. The inspectors observed whether the tests were performed in accordance with the approved procedures and assessed the adequacy of the test methodology based on the scope of maintenance work performed. In addition, the inspectors assessed the test acceptance criteria to verify whether the test demonstrated that the tested components satisfied the applicable design and licensing bases and the TS requirements. The inspectors reviewed the recorded test data to evaluate whether the acceptance criteria were satisfied. The inspectors reviewed seven post-maintenance tests performed in conjunction with the following maintenance activities:

- Work Order (WO) C0215114, Scram Time Testing After Replacing Auxiliary Contacts
- WO C0210627, Post-Maintenance Testing on RHR HX Outlet Valve (3-10-89D) After Valve and Pipe Replacement
- WO R0898571, Core Spray Loop A Check Valve, AO-3-14-013A, Inspection & Cleaning
- WO C0212638 and WO R0881503, Core Spray Loop A Inboard Isolation Motor Operated Valve, MO-3-14-12A, Motor Replacement & Gear Ratio Change
- WO C0212638, High Pressure Coolant Injection (HPCI) Pump Mechanical Seal Replacement
- WO R00944894 and WO C0213592, Reactor Core Isolation Coolant (RCIC) Minor Maintenance and RCIC Pump Mechanical Seal
- WO M1537137, High Pressure Service Water Pump, Valve, and Flow Inservice Testing (IST) Following RHR Loop B Flow Transmitter Calibration

b. Findings

Introduction: The inspectors identified a Green non-cited violation (NCV) of 10 CFR Part 50, Appendix B, Criterion XI, “Test Control,” for not adequately testing the high pressure service water (HPSW) sub-system following a valve replacement.

Description: In June 2003, the high pressure service water (HPSW) 3D residual heat removal (RHR) heat exchanger (HX) outlet throttling motor operated (MO) valve (3-10-89D) was replaced with a valve that had significantly higher pressure drop. The modification was completed in accordance with engineering change request (ECR) 96-4115, which approved the use of a cage and plug valve design. The replacement valve had a higher differential pressure drop, and therefore, was inappropriate for the existing system configuration. Following implementation of this modification, the 3D HPSW loop

could no longer meet the technical specification required design flow rate of 4500 gallons per minute (gpm). However, the test results indicated an acceptable flow rate of greater than 4500 gpm because the flow meter used during the test incorrectly measured the combined flow through both the 3B and 3D loops. Leakage through the closed 3B HPSW RHR heat exchanger outlet valve during testing masked the actual low flow condition in the 3D loop. The inspectors determined that with the known deterioration of the HPSW RHR heat exchanger outlet valves, that it was reasonable for PBAPS to have expected leakage through the 3B loop and have closed an inlet manual isolation valve in the 3B loop to provide additional isolation prior to the flow test.

On June 9, 2004, a flow test was being conducted with a blind flange installed on the 3B loop, thus isolating HPSW bypass flow through the 3B heat exchanger (HX). During a scheduled surveillance test, 3D loop flow could not reach the expected flow rate of 4500 gpm (maximum measured flow rate of 4380 gpm). By analysis, the 3D RHR HX was determined to be operable above a flow rate of 2900 gpm. Heat exchanger operability was established using design basis temperatures and actual heat transfer rates from HX thermal performance testing. Corrective action to restore design flow through the 3D loop was accomplished on July 1, 2004, by removing one of three existing orifice plates and replacing it with a full flow orifice plate.

On July 1, 2004, the post-maintenance test following the orifice plate replacement was inadequate. The test procedure sequence once again did not appropriately isolate the approximately 500 gpm of leakage flow through the 3B HPSW loop while measuring the flow through the 3D HPSW loop. Additionally, the test procedure acceptance criteria was inadequate in that it did not require a flow rate of 4500 gpm through the 3D loop.

On October 10, 2005, the leaking valve MO 3-10-89B was replaced. This eliminated the 500 gpm bypass flow through the 3B loop. Low flow through the 3D loop was identified during performance of a periodic test performed on October 20, 2005. A potential technical specification action statement was entered for the 3D HPSW loop low flow condition. An immediate safety concern did not exist because an operability evaluation determined that with the low flow condition, the 3D HPSW loop remained operable provided that river temperature remained below 80EF. During the time of the existing degraded condition, river water temperature never exceeded 80EF. At the end of the inspection period, the maximum flow through the 3D HPSW loop was less than the design flow of 4500 gpm. PBAPS is planning to correct the low flow condition prior to river temperature exceeding 80EF.

Analysis: PBAPS's inadequate test following modification of the Unit 3, HPSW D RHR HX outlet throttle valve is considered a performance deficiency since PBAPS is required to properly test systems in accordance with 10 CFR 50, Appendix B, Criterion XI, Test Control. The finding is considered greater than minor because it is associated with the procedure quality attribute of the Mitigating Systems Cornerstone and affects the cornerstone objective to ensure the capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). The finding was determined to be of very low safety significance (Green) using Phase 1 of the SDP, since the finding is a qualification deficiency confirmed not to result in a loss of function.

The inspectors identified that a contributing cause of the finding was related to the problem identification and resolution cross-cutting area. Recurrence of the low flow condition and inadequate post-maintenance testing that was previously a non-cited violation (NCV 05000278/2004003-01) and was documented in the CAP as CR 227081, dated June 9, 2004, and was not prevented. These problems were not adequately corrected by the design change and subsequent post-maintenance test performed on July 1, 2004.

Enforcement: 10 CFR 50, Appendix B, Criterion XI, "Test Control," requires that testing of systems be performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents. Contrary to the above, the post-maintenance test procedure for measuring flow through the 3D RHR heat exchanger loop following design changes to the MO-3-10-89D valve and downstream orifice plates in June 2003 and July of 2004, did not incorporate requirements to ensure that only flow through the 3D RHR HX loop was measured and an acceptance criteria of a minimum of 4500 gpm was met. Because this finding is of very low safety significance and has been entered into PBAPS's corrective action program (CR 388447), this violation is being treated as an NCV, consistent with section VI.A of the NRC Enforcement Policy: NCV 05000278/2005005-01, Inadequate Post-Maintenance Test Procedure for HPSW Flow to Residual Heat Removal Heat Exchanger. Planned corrective actions identified in CR 388447 include revising the surveillance test procedure and re-sizing the orifice plate that is located downstream of the 3D RHR HX.

1R20 Refueling and Other Outage Activities (71111.20 - 1 Sample)

1. Unit 3 Refueling Outage 15

a. Inspection Scope

The Unit 3 refueling outage (3R15) was conducted from September 19, 2005 through October 16, 2005. The inspectors performed the activities below to verify PBAPS controls over the outage activities.

- Refueling Activities - verified that PBAPS was using adequate controls to ensure the location of the fuel assemblies were properly tracked and that foreign material exclusion procedures were implemented on the refueling floor.
- Decay Heat Removal - verified the integrity of residual heat removal (RHR) system piping supports by sampling the ISI inspection of the supports.
- Drywell Closure - conducted a thorough inspection and walkdown of containment prior to reactor startup and ensured that all remaining debris, tools, and equipment were removed. Verified that newly installed insulation was accounted for in their emergency core cooling system (ECCS) suction strainer blockage calculation.
- Containment Integrity - verified local leak rate testing (LLRT) for the standby liquid control system was satisfactory.

- 3B Recirculation Pump - reviewed the Operations Technical Decision Making (OTDM) document on the decision to delay the replacement of the pump shaft. Verified that PBAPS had adequate measures in place to detect a pump shaft crack and take appropriate corrective action.
- Startup Requirements - verified that TS requirements were met by reviewing scram time testing and main steam isolation valve (MSIV) partial closure and reactor protection system (RPS) function input surveillance tests. The inspectors also verified that the proper post-maintenance testing had been completed on the RCIC and HPCI pump and turbine and selected motor-operated valves (MOVs) and air-operated valves (AOVs).
- Licensee Identification and Resolution of Problems - reviewed corrective action reports related to refueling outage activities to verify that PBAPS was identifying issues at the appropriate level and taking adequate corrective action.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22 - 6 Samples)

a. Inspection Scope

The inspectors reviewed and/or observed portions of surveillance tests, and compared test data with established acceptance criteria to verify the systems demonstrated the capability of performing the intended safety functions. The inspectors also verified that the systems and components maintained operational readiness, met applicable Technical Specification requirements, and were capable of performing the design basis functions. The surveillance tests reviewed and observed included:

- ST/LLRT 30.11.02, Rev. 5, Local Leak Rate Test (LLRT) Standby Liquid Control (CIV - Sample)
- ER-AA-335-016, Rev. 2, Inservice Inspection (ISI) Inspection of Residual Heat Removal (RHR) Supports
- ST-M-01G-450-3, Rev. 9, Main Steam Safety & Relief Valve (MSIV) Replacement (IST - Sample)
- ST-O-60F-405-3, Rev. 15, MSIV Partial Closure and Reactor Protection System (RPS) Input Functional Test
- S12F-20A-354-XXCQ, Calibration Check of Drywell Floor Drain Sump (RCS - Sample)
- S13F-20B-364-XXCQ, Calibration Check of Drywell Equipment Drain Sump Flow Instructions (RCS - Sample)

1R23 Temporary Plant Modifications (71111.23 - 1 Sample)

j. Inspection Scope

The inspectors reviewed one temporary modification to verify that implementation of the modifications did not place the plant in an unsafe condition. The review was also conducted to verify that the design bases, licensing bases, and performance capability of risk significant SSCs had not been degraded through these modifications. The inspectors verified the modified equipment alignment through control room instrumentation observations, UFSAR, drawings, procedures, and work order reviews, and plant walkdowns of accessible equipment. The following temporary modification was reviewed:

- Recirculation Pump Remote Oil Fill Pump Installation During 3R15 (ECR 05-00437)

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01 - 21 Samples)

a. Inspection Scope

The inspectors reviewed selected activities and associated documentation in the below listed areas. The evaluation of PBAPS's performance in these areas was against criteria contained in 10 CFR20, applicable Technical Specifications, and applicable Exelon procedures.

Plant Walkdowns, Radiation Work Permit (RWP) Reviews, and Jobs in Progress Reviews

The inspectors walked down selected radiologically controlled areas and reviewed housekeeping, material conditions, posting, barricading, and access controls to radiological areas. The inspectors reviewed exposure significant work areas to determine if radiological controls were acceptable and conducted selective radiation surveys. The inspectors selectively walked down these areas to determine the adequacy of posting and controls.

The inspectors selectively reviewed the radiological controls for various Unit 3 outage work activities, including drywell work activities, in-service inspection (ISI) activities, refueling activities, valve work activities, condenser bay work activities, control rod drive removal and packaging activities, reactor vessel work activities, reactor water clean-up

work activities, and feed water heater work activities. The inspectors also selectively reviewed ongoing Unit 3 condenser/turbine work including blast cleaning activities and use of respiratory protection. The reviews included evaluation of the adequacy of applied radiological controls including radiation work permits, procedure adherence, radiological surveys, job coverage, system breach surveys, air sampling, airborne radioactivity controls, and contamination controls. The reviews included, where applicable, barrier integrity and the application of engineering controls for potential airborne radioactivity areas. The inspectors also reviewed applicable radiation work permits and electronic personnel dosimetry alarm set points to verify the set points were commensurate with ambient/expected conditions and radiation work permits. The inspectors selectively interviewed workers during the inspection and verified if workers knew what actions were required when their dosimeters alarmed. The inspectors also verified workers knew the ambient radiological conditions that they were working in. The inspectors observed portions of the worker briefings for work activities.

The inspectors reviewed, observed, and discussed ongoing work in TS controlled high radiation areas, including Unit 3 drywell. The inspectors reviewed radiation protection job coverage including use of audio and visual surveillance and use of integrating alarming dosimetry.

The inspectors reviewed work activities with radiation dose rate gradients (e.g., control rod drive work activities) to verify that PBAPS had applied appropriate radiological controls including use of multiple dosimeters or repositioning of dosimetry, as appropriate. The inspectors reviewed posting and locking of entrances to high dose rate and very high radiation areas, as appropriate. The inspectors reviewed the high radiation area controls for the underwater storage of materials.

The inspectors reviewed and discussed internal dose assessments for 2005, including the Unit 3 outage and post-outage data, to identify any apparent actual occupational internal doses greater than 50 millirem committed effective dose equivalent (CEDE). The review also included the adequacy of evaluation of selected dose assessments, as appropriate, and included selected review of the program for evaluation of potential intakes associated with hard-to-detect radionuclides (e.g., airborne transuranics).

The inspectors reviewed radiological controls and exposure results associated with reactor cavity drain down and cavity decontamination, including airborne radioactivity sample results and radiological survey results (RWP Nos. 63, 68).

High Risk Significant, High Dose Rate HRA and VHRA Controls

The inspectors discussed procedure changes for high radiation area access controls since the last inspection with the radiation protection manager and selected supervisors to determine if the changes resulted in a reduction in the effectiveness and level of worker protection. The inspectors conducted a selective review of high radiation area controls (e.g., adequate posting and locking of entrances).

During the week of November 15, the inspectors observed the on-going work on the Unit 3 refueling floor. The inspectors observed radiological controls associated with removal of items from underwater.

Radiation Worker/Radiation Protection Technician Performance and Radiation Protection Technician Proficiency

The inspectors observed radiation worker performance with respect to stated radiation protection requirements to determine if the workers were aware of significant radiological conditions in their work place, and the RWP controls/limits in place, and that their performance took into consideration the levels of radiological hazards present. The inspectors also evaluated radiation protection technician performance and proficiency relative to control of hazards and work activities, as applicable. In addition, the inspectors reviewed problem reports to identify problems with worker or radiation protection technician performance.

b. Findings

Introduction: A Green non-cited violation (NCV) of Technical Specification 5.4 and Regulatory Guide 1.33, 1972, was identified by the NRC. Specifically, PBAPS failed to follow a containment access radiation protection program procedure and collect and analyze a drywell radiation monitoring system (RMS) air sample for particulate and iodine airborne radioactivity prior to initial entry of personnel into the Unit 3 reactor drywell on September 19, 2005.

Description: Radiation Protection Program Procedure HP-315, Initial Drywell Entry, Rev.12, requires, as a prerequisite for initial entry (Step 3.6), that a drywell RMS air sample be taken and analyzed for particulate and iodine activity. On September 19, 2005, at about 11:14 p.m., two PBAPS radiation protection personnel made an initial entry into the Unit 3 reactor drywell to support drywell opening and conduct of initial radiological surveys. The inspectors's review identified that no drywell RMS air sample was collected for analysis.

Analysis: The failure of PBAPS to implement radiation protection and containment entry procedures required by Technical Specification 5.4 is a performance deficiency. The finding was greater than minor because it is associated with the occupational radiation safety cornerstone attribute of exposure control and monitoring and affected the cornerstone objective. Specifically, PBAPS could not ensure the adequate protection of worker health and safety from airborne radioactive materials since initial airborne activity levels were not known. This aspect was important since known system leaks were present in the drywell and the reactor had experienced fuel leakage. The finding is suitable for SDP review in that there was a potential for a significantly greater unplanned, unintended dose had significant airborne radioactivity been present. Using the Occupational Radiation Safety Significance Determination Process, the finding is of very low safety significance (Green), in that, it did not involve an ALARA finding, did not result in an overexposure, did not result in a substantial potential for an overexposure,

and did not compromise the ability to assess dose. The workers did not sustain significant unplanned intakes of radioactive material.

Enforcement: Technical Specification 5.4 and Regulatory Guide 1.33, 1972, require that the licensee establish and implement program procedures for containment access. Radiation Protection Program Procedure HP-315, Initial Drywell Entry, Rev. 12, Step 3.6 requires that a drywell RMS air sample be collected and analyzed for particulate and iodine airborne radioactivity. This step is a prerequisite for initial entry. Contrary to the above, on September 19, 2005, two PBAPS radiation protection personnel made an initial entry into the Unit 3 drywell, but no drywell RMS air sample was collected and analyzed for particulate and iodine activity. Because this finding is of very low safety significance (Green), and has been entered into PBAPS's CAP (IR 378342), this violation is being treated as a Non-Cited Violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000278/2005005-02, Failure to Follow Radiation Protection Program Procedures.

2OS2 ALARA Planning and Controls (71121.02 - 6 Samples)

a. Inspection Scope

The inspectors conducted the following activities to determine if PBAPS was properly implementing operational, engineering, and administrative controls to maintain personnel occupational radiation exposure as low as is reasonably achievable (ALARA). Implementation of these controls was reviewed against the criteria contained in 10 CFR 20, applicable industry standards, and applicable PBAPS procedures.

Inspection Planning, Radiological Work Planning, Job Site Inspections and ALARA Controls

The inspectors reviewed ongoing Unit 3 outage work activities and selected six work activities likely to result in the highest personnel collective exposures or presented challenges for ALARA control and reviewed the current and expected collective radiation exposure for these work activities. The work activities reviewed included control rod drive change-out, in-service inspection, feedwater heater activities, reactor water clean-up work activities, turbine work, and main steam isolation valve work. The inspectors also reviewed work activities that presented unusual conditions or situations (i.e., in-vessel inspection activities, condenser work activities). The inspectors selectively reviewed implementation of applicable ALARA plans and procedures for these activities, including tracking of exposures. The inspectors reviewed ALARA work activity evaluations, exposure estimates, and mitigation requirements. The inspectors evaluated the adequacy of PBAPS's engineering and work controls and the grouping of the activities relative to work activity. The inspectors reviewed the integration of ALARA requirements into procedures, as applicable, and RWP documents. The inspectors reviewed the implementation of applicable ALARA procedures.

The inspectors toured selected areas of the radiological controlled area (RCA), including the Unit 3 drywell and observed ongoing radiological work activity. The inspectors

evaluated whether workers were utilizing low dose waiting areas, were effective in maintaining their doses ALARA, and received appropriate on-the-job supervision to ensure ALARA requirements. The inspectors selectively reviewed the level of on-the-job supervision to ensure ALARA requirements were met.

The inspectors reviewed exposures of individuals from selected work groups to identify significant exposure variations which may exist among workers.

Verification of Dose Estimates and Exposure Tracking

The inspectors reviewed PBAPS's method for adjusting exposure estimates, or re-planning work, when unexpected changes in scope, radiation levels, or emergent work were encountered to determine if the adjustments were based on sound radiation protection and ALARA principles. The inspectors also reviewed the frequency of these adjustments to evaluate the original ALARA planning process. The inspectors reviewed re-forecast work activity dose estimates, including for work activities in the turbine building, Unit 3 drywell, and balance of plant.

The inspectors reviewed the Unit 3 refueling outage revised estimate. The inspectors compared the results achieved (person-rem, dose rates) with estimated exposures and determined the reasons for inconsistencies between intended and actual exposure. The comparison included evaluation of person-hour estimates, expected dose rates, emergent work, and use of supplemental shielding, as necessary. The inspectors evaluated the reasons for inconsistencies between intended and actual work activity doses. The inspectors compared the person-hour estimates provided by maintenance planning and other groups to the radiation protection group with the actual work activity time requirements and evaluated the accuracy of these time estimates.

The inspectors determined if work activity planning included consideration of the benefits of dose rate reduction activities, such as shielding provided by water filled components/piping, job scheduling, and shielding and scaffolding installation and removal activities.

The inspectors evaluated the interfaces between operations, radiation protection, maintenance, maintenance planning, scheduling, and engineering groups for interface problems or missing program elements.

During the week of November 15, the inspectors reviewed re-forecast work activity dose estimates and selectively reviewed dose results for completed Unit 3 outage work activities. The inspectors compared the results achieved (person-rem sustained) with the intended dose established in the ALARA plans for these activities. The inspectors evaluated differences between initial radiation dose estimates and actual doses sustained for the work activities. Work-in-progress reviews as well as post-job ALARA evaluations were reviewed, as appropriate. Tasks reviewed included balance of plant work activities, refueling floor work activities, drywell work activities, under vessel work activities (control rod drive removal and replacement activities), valve work, refueling

activities, reactor disassembly/reassembly, drywell head removal, feedwater heater change out, and reactor cavity decontamination activities.

The inspectors reviewed ALARA council meetings (September 30, 2005, October 1, 2005). Also reviewed was Common Cause Analysis 343877, 371432 and AR 387627.

Source-Term Reduction and Control

The inspectors reviewed the implementation and effectiveness of PBAPS's contingency plans for managing the elevated radiation levels following the Unit 3 shutdown for refueling. In particular, the inspectors evaluated PBAPS's response in the area of ALARA planning and controls. The inspectors reviewed re-forecasts of estimated person-rem and exposure mitigation activities.

The inspectors reviewed and discussed the effectiveness of PBAPS's supplementary shielding, flushing strategies, filtration efforts, and work control/deferrals to minimize the impacts on person-rem. The inspectors reviewed the site ALARA procedures including job exposure estimates and tracking. The inspectors evaluated PBAPS's use of engineering controls to achieve dose reductions.

The inspectors evaluated PBAPS's understanding of the plant source term, including knowledge of input mechanisms to reduce the source term and whether PBAPS had a source-term control strategy in place. The review included a review of the source-term control strategy to determine if specific sources have been identified for exposure reduction actions and what priorities had been established for implementation of these actions. Also reviewed were fluid clean-up methods used to remove radioactivity.

The inspectors evaluated what results have been achieved against these priorities since the last refueling cycle. The inspectors also evaluated whether source reduction evaluations have been made and actions have been taken to reduce the overall source-term compared to the previous year.

Radiation Worker/Radiation Protection Technician Performance

The inspectors observed radiation worker and radiation protection technician performance in the area of ALARA practices to identify acceptable performance in areas of greatest radiological risk to workers. The inspectors selectively questioned workers in-the-field to evaluate their understanding of ambient radiological conditions.

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation and Protective Equipment (71121.03 - 4 Samples)

a. Inspection Scope

The inspectors selectively reviewed radiation monitoring/measurement instrumentation in the below listed areas. The review was against criteria contained in applicable Technical Specifications and station procedures.

Instrument Calibration, Operability, and Alarm Setpoint Verification

The inspectors reviewed calibration and operability check records for a variety of radiological survey instrumentation in use for radiological job coverage and area monitoring during the Unit 3 outage. The instrumentation included:

- air sampling instrumentation, portable survey meters, scaler-counters, and portable area radiation monitors. (MGP-6698-007; Dosimeter Corp. -100-1192, 121-590; RO-2A-4003; Gilian Air Sampler -17067, 3817; AMP-100-5098-037, 5097-013; SAC-4-1275; RM-14-7891).
- The inspectors also selectively reviewed the most recent calibration and operational testing of the whole body counter. The inspectors reviewed completed calibration and operability check data records.
- The inspectors selectively reviewed program actions when during calibration, or source checks, an instrument was found out of calibration.
- The inspectors selectively evaluated the calibration program relative to the plant source term to determine adequacy of calibrations.
- The inspectors selectively reviewed calibration data for personnel contamination monitors (PM-7: 224, 227, 296, 332684; PMW: 9712002, 9712006, 9712003).
- The inspectors reviewed calibration of area radiation monitors (Unit 2/3 refueling floor area radiation monitors Channels 29 (two), 30, and 31 and Post-Accident Sampling Stations).
- The inspectors selectively reviewed the calibration of the Unit 2/ 3 High Range Drywell Radiation monitors (8103 A, B, C, D; 9103 A, B, C, D).
- The inspectors selectively reviewed the calibration of electronic dosimetry (ED) worn by workers (ED Nos. 27774, 71266, 30343).

Radiation Protection Technician Instrument Use

The inspectors selectively verified the calibration expiration and source check response on radiation detection instruments staged for use for the Unit 3 outage. The inspectors observed radiation protection technicians for appropriate instrument selection and use including self-verification of instrument operability.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification (71151 - 2 Samples)

.1 Occupational Exposure Control Effectiveness (1 Sample)

a. Inspection Scope

The implementation of the Occupational Exposure Control Effectiveness Performance Indicator (PI) Program was reviewed. The inspectors reviewed corrective action program records for occurrences involving high radiation areas, very high radiation areas, and unplanned personnel radiation exposures since the last inspection in this area. The inspectors also selectively reviewed exposure records. The review was against the applicable criteria specified in NEI 99-02, Regulatory Assessment Performance Indicator Guideline, Revision 3. The purpose of this review was to verify that occurrences that met NEI criteria were recognized and identified as Performance Indicators.

b. Findings

No findings of significance were identified.

.2 RETS/ODCM Radiological Effluent Occurrences (1 Sample)

a. Inspection Scope

The implementation of the RETS/ODCM PI was reviewed. The inspectors reviewed corrective action program records and projected monthly and quarterly dose assessment results due to radioactive liquid and gaseous effluent releases; for the fourth quarter 2004 to the fourth quarter 2005 (to date). The review was against the applicable criteria specified in NEI 99-02, Regulatory Assessment Performance Indicator Guideline, Revision 3. The purpose of this review was to verify that occurrences that met NEI criteria were recognized and identified as Performance Indicators.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152 - 2 Samples)

.1 Review of Relief Valve Testing (1 Sample)

a. Inspection Scope

The inspectors reviewed relief valve set pressure testing for the group of 14 Class 2 and 3 pressure relief valves in the HPCI, RCIC, and RHR systems to evaluate the effectiveness of PBAPS's corrective actions. The review also evaluated if the corrective

actions are based on the significance of the identified problem and if risk is a primary factor in determining the significance. The inspection included a review of corrective action reports and work orders associated with the relief valve testing.

b. Findings and Observations

No findings of significance were identified. The Unit 2 HPCI pump suction header relief valve, RV-2-23B-034, failed its set pressure test in May 2004. The ASME OM Code requirement of testing two additional valves for each valve that fails was initially thought to be completed in April 2005. A review by the inspectors identified that one of the two additional valves tested, the Unit 2 HPCI cooling water header relief valve, RV-2-23B-066, was actually completed before the RV-2-23B-034 failure and could not be counted for the scope expansion testing. The test on the Unit 3 RCIC pump suction relief valve, RV-3-13B-025, which was satisfactorily completed in September 2005, was subsequently counted as the second test in the scope expansion. The scope expansion testing was not completed timely, in that, this was 16 months after the initial set pressure test failure. The inspectors noted two other instances between November 2001 and September 2005 where required scope expansion for failed set pressure tests were delayed for 15 to 19 months.

The ASME OM Code does not give a time frame when the scope expansion testing must be completed. The Code does state that when a partial complement of pretested valves are installed, the valves that had been removed from service shall be pressure tested within three months. As a corrective action, AR 290719 was written on the RV-2-23B-034 test failure and stated that the IST Owners Group recommended performing the scope expansion within three months.

.2 Incorrect Value for the Drywell Radiation Monitor in the Fission Product Barrier Emergency Action Level Matrix (1 Sample)

a. Inspection Scope

The inspectors reviewed the root cause report, a technical evaluation, and corrective actions pertaining to the determination of an incorrect value for the drywell radiation monitor in the fission product barrier emergency action level (EAL) matrix. This inspection was conducted according to NRC Inspection Procedure 71152. The applicable planning standards, 10 CFR 50.47(b) and requirements of 10 CFR 50 Appendix E were used as reference criteria.

b. Findings and Observations

Once identified, Exelon took prompt action to address this issue. The root cause report was thorough, and provided detailed history surrounding the initiation of the problem and how the problem went undetected for about ten years. Exelon's extent of condition review was fleet-wide, in that, all numerical values in the EALs and their bases were reviewed for accuracy. Exelon's technical evaluation provided an accurate perspective

on the significance of this issue. Further details regarding this issue are documented in Section 4OA7, Licensee-Identified Violation.

.3 Routine Review and Screening of Identification and Resolution of Problems

As required by Inspection Procedure 71152, Identification and Resolution of Problems, and in order to help identify repetitive equipment failures, human performance issues or program issues for follow-up. The inspectors performed routine screening of issues entered into PBAPS's CAP. This review was accomplished by selectively reviewing copies of IRs, attending daily screening meetings, and accessing PBAPS's computerized database.

.4 Problem Identification and Resolution (PI&R) Semi-Annual Trend Review

a. Inspection Scope

The inspectors reviewed a list of over 4000 issue reports (IRs) that Exelon initiated at PBAPS from April 1, 2005 through November 1, 2005, to perform the semi-annual PI&R trend review. Approximately 35 of the IRs were reviewed in detail to verify whether the issues were adequately identified, appropriately evaluated and corrected. The inspectors evaluated the IRs against the requirements of LS-AA-125, Corrective Action Program (CAP) Procedure, and 10 CFR 50, Appendix B, Criterion XVI, Corrective Action. Issues numbered IR 325378 and IR 350646 contain information about the Unit 2 HPCI turbine bearing failure interim corrective actions and may indicate an adverse trend. These issues may be selected for an annual PI&R sample.

.5 Identification and Resolution of Problems (71121.01, 71121.02 71121.03, 71122.02)

a. Inspection Scope

The inspectors selectively reviewed self-assessments and audits since the previous inspection to determine if identified problems were entered into the corrective action program for resolution. The inspectors evaluated the database for repetitive deficiencies or significant individual deficiencies to determine if self-assessment activities was identifying and addressing the deficiencies. The review also included evaluation of data to determine if any problems involved performance indicator (PI) events with dose rates greater that 25 R/hr at 30 centimeters, greater than 500 R/hr at 1 meter or unintended exposures greater than 100 millirem total effective dose equivalent (TEDE), 5 rem shallow dose equivalent (SDE), or 1.5 rem lens dose equivalent (LDE).

The inspectors also reviewed the CAP to determine if PBAPS was including ALARA deficiencies and issues in its CAP. The review included self-assessments, audits and corrective action reports related to the ALARA program since the last inspection to determine if the follow-up activities were being conducted in an effective and timely manner commensurate with their importance to safety and risk. The inspectors also reviewed selected audits and surveillances (Nuclear Oversight Rapid Trending Report, dated September 26, 2005).

The CAP review also included a review of issue reports (IRs) since the last inspection which involved potential radiation worker or radiation protection personnel errors to determine if there was an observable pattern traceable to a similar cause. The inspectors reviewed IRs initiated in the occupational radiation safety area since the last inspection to determine if PBAPS was including ALARA deficiencies and issues in its corrective action program. The review included an evaluation of corrective actions, as appropriate. The review included a check of possible repetitive issues, such as radiation worker or radiation protection technician errors (375516, 375521, 375615, 375752, 375835, 376858, 3777352, 377366, 377680, 378224, 374965, 376326, 376326, 376862, 279624, 387627, 385918, 392426, 392853).

These reviews were conducted against the criteria contained in 10 CFR 20, Technical Specifications, and the station procedures.

b. Findings

No findings of significance were identified.

4OA3 Event Followup (71153)

The inspectors reviewed the following Licensee Event Reports (LERs) to verify the accuracy of the LERs, the appropriateness of the corrective actions, and to determine whether violations of requirements or generic issues existed.

.1 (CLOSED) Licensee Event Report (LER) 05000278/2005002-00, Technical Specification Non-Compliance Due to a Unit 3 HPCI Suction Valve Logic Limit Switch Out of Adjustment

On March 21, 2005, during preparations for the performance of a routine surveillance test for the Unit 3 high pressure coolant injection (HPCI) system, Operations personnel discovered that the condensate storage tank (CST) suction motor operated (MO) valve (MO-17) did not automatically close when the suppression pool suction valves (MO-57 & MO-58) were opened for the surveillance test. PBAPS subsequently determined that a condition prohibited by Technical Specifications existed as a result of the Technical Specification 3.3.5.1 function of transferring the HPCI suction from the CST to suppression pool being inoperable for a time period longer than allowed in Technical Specification 3.3.5.1. Troubleshooting determined that the automatic closure of MO-17 failed to occur due to a limit switch being out of adjustment. The limit switch is within the motor operator housing for the MO-58 valve and provides an auto-close input into the MO-17 valve logic. The corrective action was for the MO-58 limit switch to be adjusted and the HPCI CST - suppression pool transfer logic was returned to an operable condition on March 22, 2005. This condition was initially reported pursuant to the requirements of 10CFR50.72(b)(3)(v)(D), any event or condition that at the time of discovery could have prevented the fulfillment of the safety function of structures or systems that are needed to mitigate the consequences of an accident. The inspectors reviewed the LER, which stated that PBAPS subsequently determined that the HPCI system was capable of performing its safety function due to the availability of an

adequate HPCI system suction supply. The inspectors noted that the supporting evaluation and informal calculation demonstrated that the CST level would drop below the point where the design calculation (M - 35) determined that vortexing would begin. The CST level would continue to drop into the CST discharge piping, that provides one suction supply to the HPCI pump, before the HPCI suction source would transition to the supply provided by the suppression pool in the torus. The engineering evaluation concluded that the HPCI pump and suction piping would not void. No findings of significance were identified. The licensee documented the event in Issue 315494. This LER is closed.

.2 (CLOSED) LER 05000278/2005004-00, Laboratory Analysis Identified Safety Relief Valve Set Point and Performance Deficiencies

Based on information received on October 2, 2005, from a laboratory performing safety relief valve (SRV) as-found testing, site engineering personnel determined that SRV set point and performance deficiencies existed with five SRVs that were installed during the 15th operating cycle for Unit 3. Four of the SRVs were determined to have their as-found set points in excess of the Technical Specification allowable $\pm$ 1 percent tolerance. In addition, one additional SRV was found to not properly re-close when tested. The cause of the four SRVs being outside of their allowable as-found set points is due to set point drift. Concerning the failure of the SRV to re-close, the preliminary laboratory failure analysis identified that the main valve disc had not properly re-seated when closing due to misalignment of the main valve disc spring. The valve was last refurbished in February 2001. The five SRVs were replaced with different SRVs for the 16th Unit 3 operating cycle. There were no actual safety consequences associated with this event. The licensee documented the event in Issue 381079. This LER was reviewed and no findings of significance were identified This LER is closed.

4OA6 Meetings, Including Exit

.1 Exit Meeting Summary

On January 13, 2006, the resident inspectors presented the inspection results to Mr. R. Braun and other PBAPS staff, who acknowledged the findings. The inspectors confirmed that proprietary information was not included in the inspection report.

4OA7 Licensee-Identified Violations

The following finding of very low safety significance (Green) was identified by Exelon and is a violation of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as a non-cited violation.

- 10 CFR 50.54(q) requires that licensee follow their emergency plans. Section 3 of the Peach Bottom Atomic Power Station Emergency Plan Annex, Classification of Emergencies, states that emergency action level values are based upon criteria established under Revision 2 to NUMARC/NESP-007, Methodology for Development of Emergency Action Levels. NUMARC/NESP-

007 directs licensees to use site specific values for containment radiation levels to determine the loss or potential loss of the fuel cladding and the containment barriers, respectively. To do this, the licensee developed a family of curves that plotted drywell radiation levels as a function of time for various degrees of fuel (and cladding) damage. The licensee identified that the values developed (in the mid-1990s) for the drywell radiation monitor for these two fission product barriers were incorrect. Although the family of curves was appropriately calculated, an incorrect calculation input was used to obtain the drywell radiation monitor values which resulted in the drywell radiation monitor values being 10 times higher than they should have been. Upon discovery of this error, the licensee took immediate action to correct the drywell radiation monitor value and issued IR 376267, which initiated a root cause investigation, a technical evaluation, and other associated corrective actions which included a fleet-wide review of emergency action level (EAL) numerical values and their bases. The inspectors determined that the error associated with this EAL parameter to be of very low safety significance because it would not have delayed the declaration of any emergency because of redundant EALs, based upon core level, that would be exceeded prior to the drywell radiation monitor reaching its stated threshold. . For the minority of postulated events that would not be preceded by a low reactor vessel level condition, the inspectors credited existing emergency operating procedures to mitigate the event conditions.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Exelon Generation Company personnel

R. Braun, Site Vice President
J. Grimes, Plant Manager
N. Alexakos, Programs Manager
P. Cowan, Director of Licensing and Regulatory Affairs
C. Crabtree, Radiation Protection Supervisor
F. Crosse, Manager, Radwaste and Environmental
P. Davison, Engineering Director
D. Foss, Senior Regulatory Engineer, Regulatory Assurance
J. Geary, Instructor/Exam Development
C. Hardee, Manager, Radiological Engineering
K. Hudson, ISI/CISI/R&R/Snubber Program Manager
J. Leonard, Manager NDE Services
D. Lewis, Operations Director
M. Lyate, Radiation Support Manger
J. Mallon, Regulatory Assurance Manager
R. Norris, Manager, Radiation Protection
G. Stathes, Maintenance Director
A. Wasong, Training Director

NRC personnel

D. Lew, Deputy Director, DRP
J. Trapp, Branch Chief, DRP, Branch 4
A. Burritt, Senior Project Engineer, DRP, Branch 4.

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000278/2005005-01	NCV	Post-Maintenance Testing did not Identify Restricted HPSW Flow on Residual Heat Removal Heat Exchanger (Section 1R19)
05000278/2005005-02	NCV	Failure to Implement Radiation Protection Procedures for Drywell Initial Entry in Accordance with TS 5.4 (Section 2OS1)

Closed

05000278/2005002-00	LER	Technical Specification Non-Compliance Due to a Unit 3 HPCI Suction Valve Logic Limit Switch Out of Adjustment (Section 4OA3)
05000278/2005004-00	LER	Laboratory Analysis Identified Safety Relief Valve Set Point and Performance Deficiencies (Section 4OA3)

Discussed

NONE

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

AG-108, Preparation for Severe Weather
RT-O-040-620-2 Outbuilding HVAC and Outer Screen Inspection for Winter Operation
RT-O-040-630-2 Winterizing Procedure
OP-PB-108-111-1001 Preparation for Severe Weather
OP-AA-108-111-1001 Severe Weather and Natural Disaster Guidelines

Section 1R04: Equipment Alignment

AR 399646, HV-O-52F-10020A, Handle was not Attached to the Valve

Section 1R05: Fire Protection

PF-13C, Unit 3, Torus Room, Reactor Building 92'6" Elevation, Fire Zone 13C
PF-5C, Unit 2, Torus Room Reactor Building, 91'6" Elevation, Fire Zone 5C

Section 1R08: Inservice Inspection

NDT Examination Reports

UT Core Shroud H3 and H4 Welds
UT Examination Data Forms, Welds 19 and 20 #1 Coupling/19 and 20 #2 Coupling RHR Injection Testable Check Valve AO-3-10-046B Equalizing Line
UT Examination Data Forms, HPCI Welds FW 23-0-49 and FW 23-0-50
Radiographs, HPCI Welds FW 23-0-49 and FW 23-0-50

In Vessel Remote Visual Examination

EVT-1 Structural Weld Jet pump 2-AD-3B

Repair-Replacement

RHR Injection Testable Check Valve AO-3-10-046B Equalizing Line Socket-Welds
High Pressure Coolant Injection (HPCI) Motor Operated Valve MO-3-23-014

Action Request Reports

AR 00177764, UT Torus Inspection
AR 00179788, U3 ASME Class 1 Hydro Test
AR 00188632, ISI/CISI Program Oversight Assessment Items
AR 00254250, Hanger 10DDN-H91 Integral Attachment Has an Indication

Section 1R11: Licensed Operator Requalification Program

Requalification Program Procedures

TQ-AA-109, Exelon Nuclear Duel Handling Supervisor Training, Rev. 3
TQ-AA-106-303, Exelon Nuclear Licensed Operator Training Job Performance Measure Development Job Aid, Rev. 2
TQ-AA-106-304, Exelon Nuclear Licensed Operator Requal Training Exam Development Job Aid, Rev. 5

JPMs

Refuel Platform C.O.L., Main Hoist Grapple Checks
Response to an Unanticipated Spent Fuel Pool Hi Radiation Alarm During Fuel Handling in the Spent Fuel Pool
Actual Dummy Fuel Movement in the Spent Fuel Pool
Control Rod Removal Using Combined Grapple
Fuel Handling Director Shift Turnover Checklist

Biennial Written Exams 2004

2004 NRC Written Examination (PBAPS)

Other

Senior Reactor Operator - Limited Requalification (LSROR) Training
2005 LSRO Operating Exam Sample Plan

Section 1R12: Maintenance Effectiveness

AR 252364, Mechanical Vacuum Pump Pre-outage Work Not Completed
ER-AA-310-1004, Maintenance Rule - Performance Monitoring

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Regulatory Guide 1.160, Monitoring the Effectiveness of Maintenance at Nuclear Power Plants
Regulatory Guide 1.182, Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants
AR 434998, Control Rod Did Not Move As Expected
A 1544131
ECR 05-159, Install Line Stop Hardware to Replace ESW 518 Valve
HV-AA-1272, Technical Task Risk/Rigor Assessment, Pre-job Brief, Independent Third Party Review and Post-job Brief

Section 1R15: Operability Evaluations

CC-AA-309-101, Engineering Technical Evaluations
LS-AA-105, Operability Determinations
OP-AA-108-111, Adverse Condition Monitoring and Contingency Planning
OP-AA-106-101-1006, Rev. 2, Operational and Technical Decision Making Process

Section 1R16: Operator Work-Arounds

Drawing —303, P & I Diagram - Main Steam, Bypass and Crossaround
Drawing —304, P & I Diagram - Turbine & Extraction Steam
Operator Workaround Board Meeting Minutes for 7/19/2005 meeting
IR 351613
AR A1487260, Auxiliary Steam Supply to Seal Steam Header
TC #05-0140, ARC-20C208R, Steam Seal Header Low Pressure
Design Baseline Document (DBD) P-T-09, "Internal Hazards"
DBD P-S-34, Radwaste System
T-103 Secondary Containment EOP
Alarm Response Card 224 C-5, C RHR Pump Room Flood

Section 1R19: Post-Maintenance Testing

ECR 04-352 Modify Orifice Plates Downstream of MO-3-10-89D
AR 388447, Low Flow Through HPSW Side of 3D RHR Heat Exchanger During RT-O-032-300-3
AR 227081 Low Flow From 3 DP 042 During ST
A 1454398 Unit 3 HPSW Piping Replacement
A 1472299 HPSW MO-3-10-89B Leak Thru

Section 1EP4: Emergency Action Level (EAL) Revision Review and Emergency Plan (E-Plan) Changes

EP-AA-1000, Standardized Radiological Emergency Plan, Rev. 16
EP-AA-1008, Radiological Emergency Plan Annex for Peach Bottom
EP-AA-120-1001, 10 CFR 50.54(q) Change Evaluation
Exelon Standard Emergency Plan and Implementing Procedures

Peach Bottom Annex Emergency Plan

Section 2OS1: Access Controls to Radiologically Significant Areas

Unit 3 drywell atmosphere air sample data
Unit 3 reactor coolant chemistry data for shutdown
Plant source term analysis data
Various radiation monitor calibration and operability check data
Whole body counter calibration data
Various radiological survey records for ongoing outage work activities including records for initial Unit 3 drywell entry
Various radiation work permits for Unit 3 outage work activities and associated ALARA plans.
Various personnel whole body count data results
Procedure RP-AA-300, Rev. 1, Radiological Survey Program

Section 2OS3: Radiation Monitoring Instrumentation and Protective Equipment

Procedure HP-C-470, Rev.0, Calibration of the APTEC PMW Personnel Monitor
Procedure HP-C-406, Rev.0, Calibration of Eberline Model PM-7, Personnel Monitor
Procedure SI 2R-636-8103-A1C2, Rev 4, 5, (Drywell High Range)
Procedure ST-C-095-868-2, Rev.6, (Drywell High Range)
Procedure SI3R-636-9103-D1C2, Rev.3,4 (Drywell High Range)
Procedure ST-C-095-868-3, Rev.5 (Drywell High Range)

Section 4OA1: Performance Indicator (PI) Verification

EP-AA-125-1001, EP PI Guidance
EP-AA-125-1002, ERO Performance, PI Guidance
EP-AA-125-1003, ERO Readiness, PI Guidance
EP-AA-125-1004, Emergency Response Facilities and Equipment PI Guidance

Section 4OA2: Identification and Resolution of Problems

Procedure ST-C-095-865-2, Rev.1, Determination of Annual Dose Equivalent from All Uranium Fuel Cycle Sources
Procedure LS-AA-2140, Rev. 4 Monthly Data Elements for NRC Occupational Exposure Control Effectiveness

Condition Reports	Work Orders
AR 00290719	R0046824
AR 00294486	R0781698
AR 00294487	R0781699
AR 00387422	R0791236
AR 00387475	R0884859
	R0884885

* Indicates this was generated as a result of this inspection.

Section 4OA7: Licensee Identified Violations

Root Cause Investigation Report, Action Tracking Item Number 376267
Evaluation of the Ability to Properly Classify Events due to Errors in Fission Product Barrier Matrix - Peach Bottom Atomic Power Station
Emergency Plan: Peach Bottom Atomic Power Station Annex, Section 3 Classification of Emergencies
Emergency Plan: Peach Bottom Atomic Power Station Annex, EAL Technical Basis

LIST OF ACRONYMS

ALARA	as low as is reasonably achievable
AR	action report
ASME	American Society of Mechanical Engineers
BWRVIP	Boiling Water Reactor Vessel Internals Project
CAP	corrective action program
CFR	Code of Federal Regulations
CIV	containment isolation valve
CR	condition report
EAL	emergency action level
ECCS	emergency core cooling system
ECR	engineering change request
EDG	emergency diesel generator
HP	Health Physics
HPCI	high pressure coolant injection
HPSW	high pressure service water
HRA	high radiation area
HX	heat exchanger
IMC	Inspection Manual Chapter
IR	issue report
ISI	Inservice Inspection
JPM	job performance measure
LER	licensee event report
LHRA	locked high radiation area
LSRO	limited senior reactor operator
MO	motor operated
NCV	non-cited violation
NDE	nondestructive examination
NRC	Nuclear Regulatory Commission
OE	operating experience
PSA	probabilistic safety assessment
PBAPS	Peach Bottom Atomic Power Station
PSIG	pounds per square inch gauge
R/hr	rem per hour
RBCCW	reactor building closed-cooling water
RCA	radiologically controlled area

RCIC	reactor core isolation cooling
RCS	reactor coolant system
RHR	residual heat removal
RMS	radiation monitoring system
RPS	reactor protection system
RT	routine test
RTP	rated thermal power
RWP	radiation work permit
SDP	significance determination process
SSCs	structures, systems, or components
SRVs	safety relief valves
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
UT	ultrasonic testing
VIP	vessel internals projects
WO	work order