Peach Bottom 3
3Q/2004 Plant Inspection Findings

Initiating Events

Mitigating Systems

Significance: G Jun 30, 2004
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
Design Changes Made to the High Pressure Service Water Motor Operated Valve on the Residual Heat Removal Heat Exchanger Discharge Restricted HPSW Flow
A self-revealing non-cited violation (NCV) of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was identified. Specifically, design changes made to the high pressure service water (HPSW) motor-operated valve (MOV) on the residual heat removal (RHR) heat exchanger discharge restricted HPSW flow in the affected RHR loop. HPSW flow in this loop was reduced below the design basis flow. The HPSW design basis flow is used to verify RHR heat exchanger operability.

The finding is considered more than minor, in that, the issue was associated with the design control attribute of the mitigating systems cornerstone. The cornerstone objective was affected because improper control of the design change to MO-3-10-89D reduced HPSW flow through this loop below the design basis flow of 4500 gpm. The finding was evaluated using Appendix A of NRC IMC 0609, "Significance Determination of Reactor Inspection Findings for At-Power Situations." The inspectors concluded that this issue is of very low safety significance since the safety function was maintained.

The inspectors identified that a contributing cause of the finding was related to the problem identification and resolution cross-cutting area, in that Design Engineering personnel did not adequately resolve known problems with the HPSW MO-89 series valves.
Inspection Report# : 2004003*(pdf)*

Significance: G Mar 31, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
Maintenance Rule Bases Exceeded on the 2A Reactor Building Closed-Cooling Water Heat Exchanger and E-2 Emergency Diesel Generator
The NRC identified a non-cited violation (NCV) of 10 CFR 50.65, the Maintenance Rule, having very low safety significance (Green). As of December 14, 2003, the 2A reactor building closed cooling water (RBCCW) heat exchanger exceeded the unavailability criteria established by Exelon in its Maintenance Rule scoping document. The RBCCW system was not monitored against Exelon established criteria of two percent unavailability per 24 month period. Additionally, as of February 13, 2004, the E2 emergency diesel generator (EDG) exceeded the reliability criteria established by Exelon in its Maintenance Rule scoping document. The E2 EDG performance was not monitored against Exelon established criteria of one maintenance preventable functional failure (MPFF) per 24 month period. The events determined to be MPFFs on the E2 EDG occurred on March 21, 2003, and September 15, 2003.

The finding is more than minor because the E2 EDG was associated with the Equipment Performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The 2A RBCCW heat exchanger was associated with the Equipment Performance attribute of the Initiating Events cornerstone. Exelon's not analyzing the E2 EDG or the 2A RBCCW heat exchanger performance in accordance with the maintenance rule was determined to have very low safety significance (Green) using Phase 1 of the Significance Determination Process (SDP) for Reactor Inspector Findings for At-Power reactor situations.

Inspection Report# : 2004002*(pdf)*

Significance: G Dec 31, 2003
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
Inadequate Clearance Restoration Results in Automatic Start of All Four Emergency Diesel Generators
A self revealing non-cited violation (NCV) of Technical Specification 5.4.1 was identified. The NCV is of very low safety significance. The written clearance restoration instructions provided to maintenance technicians to restore Unit 3 reactor vessel water level instruments to service

following maintenance were inadequate. The inadequate instructions resulted in the unexpected generation of signals to actuate the Unit 3 emergency core cooling systems (ECCS) and to start the four EDGs. All four EDGs started but were not connected to the Unit 2 or 3 safety buses because normal power was available to these buses. None of the Unit 3 ECCS actuated because Unit-3 was in a refueling outage.

The finding is greater than minor because it is similar to Insignificant Procedure Error Example 5.a in Appendix E of IMC 0612, "Power Reactor Inspection Reports." The reactor vessel instrumentation system was being returned to service after maintenance with an inadequate work instruction and caused automatic start of all four EDGs. The finding is of very low safety significance on both Unit 2 and Unit 3. Unit 3 was assessed using IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process." The reactor coolant system level was maintained greater than 23 feet, the two sources of vessel level instrumentation used by plant operators to monitor reactor coolant system inventory were not affected, and the finding did not represent a loss of control. Unit-2 was assessed using IMC 0609, Appendix A "Significance Determination of Reactor Inspection Findings for At-Power Situations." The finding was not a design deficiency, did not represent an actual loss of safety function, and did not involve the loss of equipment designed to mitigate an external event.

Inspection Report# : 2003005*(pdf)*

Significance: G Dec 31, 2003
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
Inadequate Procedure Maintenance Guidance Results in Unit 2 High Pressure Coolant Injection System Check Valve Failure
A self-revealing non-cited violation (NCV) of Technical Specification 5.4.1 was identified for an inadequate high pressure coolant injection (HPCI) check valve maintenance procedure. The NCV is of very low safety significance. The deficiency resulted in the Unit 2 HPCI system suppression pool suction check valve not fully closing during surveillance testing on December 10, 2003. Since the check valve was not fully closed, approximately 16,000 gallons of water from the condensate storage tank was inadvertently transferred to the suppression pool. In addition, unplanned HPCI system unavailability was needed to facilitate repairs.

The finding is greater than minor because it is associated with the procedure quality attribute and adversely affects the mitigating systems cornerstone objective. The inconsistent valve performance did not ensure the availability or reliability of HPCI to respond to an initiating event. The finding is of very low safety significance because the finding was not a design deficiency, did not represent an actual loss of safety function, and did not involve the loss of equipment designed to mitigate an external event.

Inspection Report# : 2003005*(pdf)*

Significance: G Nov 18, 2003
Identified By: NRC
Item Type: VIO Violation
Failure to Adequately Maintain the E2 Emergency Diesel Generator
(By letter dated February 3, 2004, Final Significance Determination for a White Finding and Notice of Violation, EA 03-224.)

A self-revealing finding was identified for the failure to adequately maintain the E2 emergency diesel generator (EDG) between July 1992 and September 2003. This finding involved two apparent violations. An apparent violation of Technical Specifications was identified for the failure to maintain the maintenance procedure for installation of EDG adapter gaskets. The procedure did not incorporate certain vendor recommendations intended to provide proper sealing of the gaskets, leading to relaxation over several years that allowed combustion gases to enter the jacket coolant system. An apparent violation of 10 CFR 50 Appendix B, Criterion XVI, "Corrective Actions" was identified because Exelon did not correct a condition adverse to quality following two instances of low jacket water pressure observed on the E2 emergency diesel generator (EDG) in March and April 2003. Subsequently, the EDG failed due to a low jacket water pressure condition.

This finding is more than minor because it is associated with the equipment performance attribute of the Mitigating Systems cornerstone and adversely affects the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events. The finding was assessed using a Phase 3 evaluation. The finding is of low to moderate safety significance (WHITE) at Unit 2 based on delta core damage frequency (CDF) and delta large early release frequency (LERF). The finding is of very low safety significance (GREEN) at Unit 3 based on CDF and LERF. The difference between the two units is attributable to differences in electrical bus loads.

Inspection Report# : 2003013*(pdf)*
Inspection Report# : 2004010*(pdf)*

Significance: G Nov 18, 2003
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
Ineffective Instructions for Installation of SRV Packing
A non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified because maintenance procedures for the installation of the packing on the safety relief valve (SRV) air operator assembly were inadequate. The packing installation procedures did not assure that the packing was properly installed (packing nuts adequately tightened, etc.) on RV-3-02-071G resulting in hot steam leaking past the packing and damaging the air operator diaphragm during the September 15, 2003, event.

This finding is more than minor because it is associated with the Procedure Quality attribute of the Mitigation Systems cornerstone and adversely affects the cornerstone objective to ensure the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The finding was evaluated in accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," using a Phase 3 significance determination process (SDP) analysis. This issue is of very low safety significance, based on the Phase 3 analysis results, assuming that RV-3-02-071G would not have opened from the control room for one year. The Phase 3 evaluation, using the 3.01 SPAR model for Peach Bottom, indicated a negligible increase in risk of not being able to manually depressurize with the 10 remaining valves.

Inspection Report# : 2003013*(pdf)*

Significance: G Nov 18, 2003
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
EOP Support Procedures Not Adequately Established With Steps to Bypass Containment Isolations
A self-revealing non-cited violation of Technical Specification (TS) 5.4.1, "Administrative Controls - Procedures," was identified. The existing emergency operating support procedures did not have adequate instructions to be used when Group II / III isolation signals were present. This resulted in delaying restoration of torus level and reducing containment pressure for approximately 14 hours while a procedure was developed.

This finding is more than minor because it is associated with the Procedure Quality attribute of the Mitigating Systems cornerstone and adversely affects the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. It delayed the operators in placing Unit 3 into a cold shutdown condition. This finding is of very low safety significance (Green) using Phase 1 of the Significance Determination Process for reactor inspection findings for At-Power reactor situations. The finding is of very low safety significance because the finding is not a design or qualification deficiency, does not result in a loss of safety function, and is not potentially risk significant due to seismic, flood, fire, or weather related initiating event.

Inspection Report# : 2003013*(pdf)*

Significance: N/A Jun 08, 2000
Identified By: NRC
Item Type: AV Apparent Violation
Assoc Circuit - Mechanical Damage from Fire Induced Cable Faults not evaluated.
PECO adopted a licensing position that mechanical damage to alternative shutdown equipment resulting from fire-induced cable faults, as described in Information Notice 92-18, was outside the scope of the licensing and design bases of the facility. As a result, PECO did not evaluate the control circuits of the alternative shutdown equipment to determine if it was susceptible to this problem. Since a detailed review of the alternative shutdown capability at PBAPS was not performed as part of the scope of this inspection, the risk associated with this issue was not established.

This issue is being treated as an apparent violation of Condition 2.C.4 of the operating licenses for both Unit 2 and Unit 3, which requires PECO to implement and maintain the fire protection program described in the NRC Safety Evaluation Reports. PECO has entered this issue into their corrective action program and has implemented reasonable compensatory measures pending final resolution of the issue. However, the issue of mechanical damage to safe shutdown equipment due to fire-induced cable faults is in contention between the NRC and the nuclear industry. As such, any further enforcement action will be deferred pending final resolution of this issue by the Nuclear Energy Institute and the NRC staff, in accordance with Enforcement Guidance Memorandum 98-02, Revision 2, issued February 2, 2000.
Inspection Report# : 2000003*(pdf)*

Significance: G Jun 08, 2000
Identified By: NRC
Item Type: AV Apparent Violation
Assoc Circuit - Reliance on signal spurious assumption of one per system per fire.
PECO's specification for performing circuit analyses of post-fire safe shutdown equipment stipulates that only one spurious actuation for each system affected by any one fire be analyzed. For the areas inspected, the team determined that PECO adequately protected against fire-induced spurious actuations. The team did not identify any additional spurious actuations which would have prevented achieving safe shutdown conditions in the post-fire operating environment.

The assumption that only a single spurious actuation need be considered for any one system for any one fire is an apparent violation of the requirements of Section III.G. and III.L. of Appendix R to 10 CFR 50. PECO entered this issue into their corrective action program and have implemented reasonable compensatory measures. However, the issue of multiple spurious actuations of equipment in a post-fire environment is in contention between the NRC and the nuclear industry. As such, any further enforcement action will be deferred pending final resolution of this issue by the Nuclear Energy Institute and the NRC staff, in accordance with Enforcement Guidance Memorandum 98-02, Revision 2, issued February 2, 2000.
Inspection Report# : 2000003*(pdf)*

Barrier Integrity

Emergency Preparedness

Occupational Radiation Safety

Significance: G Dec 31, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Properly Use Respiratory Protective Equipment in Accordance With 10 CFR 20.1703(a)
The inspector identified a non-cited violation of very low safety significance of 10 CFR20.1703(a). Exelon did not use continuous flow respirator protective equipment (Bullard Series 88 helmets) in accordance with the approval certification of the National Institute for Occupational Safety and Health (NIOSH). Specifically, on September 25 and 29, 2003, an NRC inspector identified that at least one worker on each day used Bullard Series 88 continuous flow airline respirators (NIOSH approval No. TC-19C-293) during blast cleaning of contaminated turbine components, and the respiratory protective equipment was used with breathing air provided at unapproved air pressure settings.

The finding was greater than minor in that it is associated with the occupational radiation safety cornerstone attribute of exposure control and did affect the cornerstone objective. Specifically, Exelon could not ensure adequate protection of worker health and safety from exposure to radiation from radioactive material if respiratory protection equipment is improperly used. The finding is suitable for SDP review in that there was a potential for a significantly greater unplanned, unintended dose if breathing air pressures outside the values specified by NIOSH were used. The finding is of very low safety significance in that, it did not involve an ALARA finding, result in an overexposure, result in a substantial potential for an overexposure, and did not compromise the ability to assess dose.
Inspection Report# : 2003005*(pdf)*

Public Radiation Safety

Physical Protection

Physical Protection information not publicly available.

Miscellaneous

Last modified : December 29, 2004