Fort Calhoun
4Q/2007 Plant Inspection Findings

Initiating Events

Significance: G Sep 30, 2007
Identified By: NRC
Item Type: FIN Finding
Ineffective Corrective Actions for Hydrazine Spills
A Green self-revealing finding was identified for inadequate corrective actions, which resulted in a hydrazine spill. Specifically, corrective actions taken previously were ineffective at preventing hydrazine spills, a condition that had the potential to injure personnel, prevent personnel response to events, or adversely affect mitigating systems equipment (e.g., Diesel Driven Auxiliary Feedwater Pump FW-54.) This issue has been entered into the licensee's corrective action program as CR 200703745.

The finding was greater than minor because hydrazine spills could be reasonably viewed as a precursor to a significant event. During a previous event, the licensee attempted to neutralize the spill which resulted in a violent exothermic reaction and a toxic gas release to the Turbine Building. The finding, which is under the Initiating Events cornerstone, was of very low safety significance because it (1) did not result in exceeding the Technical Specification limit for RCS leakage; (2) did not contribute to both the likelihood or a reactor trip and that mitigation equipment would be unavailable; and (3) did not increase the likelihood of a fire or flood.

Inspection Report# : 2007004 *(pdf)*

Significance: G Sep 14, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Follow a Procedure That Would Identify Potential Missile Hazards (Section 4OA2.e(2)(a))
A noncited violation was identified for failure of operators to follow a procedure as required by Technical Specification 5.8.1.a. This failure resulted in the station not identifying that loose material had the potential to become airborne during high winds and potentially cause a loss of off-site power. This finding has a crosscutting aspect in the area of problem identification and resolution, specifically the corrective action program attribute (P.1(a)) in that the licensee failed to identify potential missile hazards despite numerous opportunities to do so.

This finding was determined to be greater than minor in that it affected the "Protection Against External Factors" attribute of the Initiating Events cornerstone. Further, this condition could also reasonably be viewed as a precursor to a significant event. The inspectors evaluated this finding using Manual Chapter 0609, Appendix A and determined that it was of very low safety significance (Green) because it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment would not be available. This condition has been entered into the licensee's corrective action program as Condition Reports 2007-3544 and 2007-3568.
Inspection Report# : 2007010 *(pdf)*

Significance: G Sep 14, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Inadequate Definition of a Missile Hazard Results in Loss of 161 KV Power
A self-revealing noncited violation of Technical Specification 5.8.1.a occurred for an inadequate procedure that narrowly defined the definition of a missile. This inadequacy resulted in the loss of 161 kilovolt power to the safety-related busses on August 20, 2007 during a high wind event when debris not meeting the definition of a missile struck a transformer relay cabinet. This finding has a crosscutting aspect in the area of human performance, specifically the resources attribute (H.2C) in that the licensee failed to conservatively describe in procedures what constitutes a

missile hazard.

This finding was determined to be greater than minor in that it affected the “Protection Against External Factors” attribute of the Initiating Events cornerstone. The inspectors evaluated this finding using Manual Chapter 0609, Appendix A. The initial screening determined a Phase 2 was required and since all safety-related equipment was operable and the safety-related busses remained energized, the Loss of Offsite Power Significance Determination Process worksheet was used to evaluate the risk. A “< 3 day” exposure results in an Initiating Event Likelihood of four for the Loss of Offsite Power Significance Determination Process worksheet. Evaluating all the sequences on the worksheet results in the lowest sequence being eight. This identifies that the significance of the finding was Green (very low safety significance) with respect to core damage frequency. This condition has been entered into the licensee’s corrective action program as Condition Report 2007-3361.
Inspection Report# : 2007010 *(pdf)*

Significance: G Mar 31, 2007
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Loss of Shut Down Cooling Due to Inadequate Procedure
A Green self-revealing noncited violation of 10 CFR Part 50, Appendix B, Criterion V, occurred when operating procedure OP-3A, “Plant Shutdown,” Revision 66, did not contain appropriate guidance to licensed operators to prevent the loss of shutdown cooling when reactor coolant pumps were secured. The procedure did not provide a caution statement, similar to one found in other procedures that would have alerted the operators that reduced spray flow exists when running less than four reactor coolant pumps.

This finding was determined to be greater than minor in that it affected the "Procedure Quality" attribute of the Initiating Events cornerstone. The inspectors attempted to evaluate this finding using Manual Chapter 0609, Appendix G, because the condition occurred during cold shutdown conditions. The reactor had been shut down for 79 days and one third of the fuel was replaced with new fuel bundles. The time to boil was three hours, therefore none of the checklists were applicable. Using Checklist 2 as a bounding evaluation, resulted in a Green finding. Since the finding was not suitable for analysis under the significance determination process, regional management and a Senior Reactor Analyst review determined that the finding was of very low safety significance (Green) because there was no affect on the reactor coolant system and no radionuclide release. This finding has been entered into the licensee’s corrective action program as Condition Report 200605629. This finding has a crosscutting aspect in the area of human performance associated with resources because procedure OP-3A, "Plant Shutdown, Revision 66" did not contain complete accurate and up to date information for the control of pressurizer spray while transiting to shutdown cooling.
Inspection Report# : 2007002 *(pdf)*

Mitigating Systems

Significance: G Aug 31, 2007
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Inadequate Emergency Diesel Generator Postmaintenance Test
A self-revealing Green non-cited violation of TS 5.8.1.a (Procedures) was identified for an inadequate postmaintenance testing procedure. Craftsmen had replaced the field flash relay auxiliary contacts (following a previous field flash failure on February 14, 2007) and had misaligned the contact assembly during installation. Postmaintenance testing was inadequate because it did not verify that the contacts properly repositioned to the closed position following the surveillance test. When the emergency diesel generator was started two days later, for a normal surveillance, the field did not flash because the contacts were stuck open.

This finding was greater than minor because the finding was associated with the mitigating systems cornerstone objective (procedure quality attribute) to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The exposure time for this performance deficiency was approximately 60 hours. Using the Manual Chapter 0609, Appendix A, “Determining the Significance of Reactor

Inspection Findings for At-Power Situations," Phase 1 screening worksheet, the inspectors determined that the finding was of very low safety significance (Green) because it was not: 1) a design or qualification deficiency; 2) a loss of system safety function; 3) an actual loss of safety function for greater than its technical specification allowed outage time; 4) a loss of safety function of a non-technical specification train; or 5) a seismic, flooding or severe weather related finding. The finding had crosscutting aspects in the human performance area, specifically the resource attribute (H.2(c)) in that a complete and accurate test instruction was not provided to test the 2CR auxiliary relay contacts.
Inspection Report# : 2007011 *(pdf)*

Significance: W Aug 31, 2007
Identified By: NRC
Item Type: VIO Violation
Inadequate Emergency Diesel Generator Corrective Measures
The inspectors identified an apparent violation of 10 CFR 50, Appendix B, Criterion XVI (Corrective Actions), with two examples, for the failure to: 1) treat the February 14, 2007 emergency diesel generator failure as a significant condition adverse to quality; and 2) promptly identify and correct a significant condition adverse to quality (high resistance on field flash circuit contacts) after determining that similar operating experience was applicable. In addition, a contributor to the inoperable emergency diesel generator included the failure to revisit the diesel generator operability evaluation in response to the applicable operating experience. Overall, the licensee responded to various problems in isolation and did not adopt a corrective action process that maintained emergency diesel generator reliability and availability.

These concerns were greater than minor because they affected the mitigating systems cornerstone objective (equipment performance attribute), to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. For the preliminary significance determination, the inspectors used a 14 day exposure time, which was ½ the time period between the last successful surveillance and the February 14, 2007 failure. However, this exposure time could increase to 28 days if the NRC determines the failure was caused by contact binding, versus contamination. Using the NRC Inspection Manual Chapter 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," significance determination process, a Region IV senior reactor analyst determined that the finding was potentially Greater-than-Green. The finding had crosscutting aspects in the area of problem identification and resolution, operating experience component, in that the licensee failed to evaluate relevant operating experience in a timely manner.
Inspection Report# : 2007011 *(pdf)*

Significance: W Aug 31, 2007
Identified By: NRC
Item Type: VIO Violation
Failure to Provide Procedure for Safety Related Maintenance Activity
The inspectors identified an apparent violation of Technical Specification 5.8.1.a (Procedures) because craftsmen used an unapproved wet lubricant on the emergency diesel generator field flash relay auxiliary contact sliding mechanisms without a procedure that directed the action. The lubricant was the most likely contributor to oil and dust contamination on the auxiliary contact surfaces, which apparently caused the emergency diesel generator failure. In addition, a contributor to the violation included the failure to properly implement the Reliability Centered Maintenance Program.

This finding was greater than minor because it affected the mitigating systems cornerstone objective (procedure quality attribute), to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. For the preliminary significance determination, the inspectors used a 14 day exposure time, which was ½ the time period between the last successful surveillance and the February 14, 2007 failure. However, this exposure time could increase to 28 days if the NRC determines the failure was caused by contact binding, versus contamination. Using the NRC Inspection Manual Chapter 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," significance determination process, a Region IV senior reactor analyst determined that the finding was potentially Greater-than-Green. The finding had crosscutting aspects in the area of human performance, resources component, in that the licensee failed to provide a procedure to control a safety related maintenance activity.
Inspection Report# : 2007011 *(pdf)*

Significance: G Jul 25, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Inadequate Abnormal Operating Procedure for loss of Component Cooling Water
The team identified a noncited violation of Fort Calhoun Technical Specification 5.8, “Procedures,” for an inadequate Technical Specification required procedure. Specifically, Abnormal Operating Procedure 11, "Loss of Component Cooling Water,” could not be performed as written for establishing backup raw water to the containment fan coolers during post-accident conditions with a loss-of-component cooling water. The licensee has entered this finding into their corrective action program as Condition Report 2007-02268.

The finding is greater than minor because it is associated with the barrier integrity cornerstone attribute for operating post event procedure quality. Using the significance determination process of Manual Chapter 0609, Appendix A, for the containment barrier cornerstone, the finding did not represent an actual open pathway in the physical integrity of reactor containment or involve an actual reduction of defense-in-depth for the atmospheric pressure control of the reactor containment. The finding had a cross-cutting aspect in the area of human performance resources because the licensee did not ensure that procedures to assure nuclear safety, in this case establishing backup raw water to the containment fan coolers during post-accident conditions with a loss-of-component cooling water, were complete, accurate and up-to-date.
Inspection Report# : 2007007 *(pdf)*

Significance: G Jul 25, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Analyze Impact of Heat Loading in Safety Injection Pump Room from the Start of a Third High Head Safety Injection Pump
The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, for the failure to perform a complete and adequate analysis of safety injection pump room temperatures to support operation of two high pressure safety injection pumps in one room during a design basis accident. The licensee performed the design calculation based on a limiting case with only one high pressure safety injection pump operating. However, at the operators discretion, the second high pressure safety injection pump could be started. The starting of the second high pressure safety injection pump in safety injection pump room 21 would increase the room temperature to near equipment qualification temperature limits. The licensee has entered this finding into their corrective action program as Condition Report 2007-02441.

This finding is more than minor because the engineering calculation results did not include the operation of a second high pressure safety injection pump running which would increase the temperature in pump Room 21 to near equipment qualification temperature limits. This unanalyzed condition now raised reasonable doubt on the operability of a system or component (Manual Chapter 0612, Appendix E.3.J). Using the Manual Chapter 0609, Phase 1 screening worksheet, the issue screened as having very low safety significance because it was a design deficiency confirmed not to result in loss of operability in accordance with NRC Manual Chapter Part 9900, Technical Guidance, Operability Determination Process for Operability and Functional Assessment.
Inspection Report# : 2007007 *(pdf)*

Significance: G Jul 25, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Translate Regulatory Requirements and Design Basis to Equipment Required to Support the Raw Water System - Unresolved Item 05000285/2005009-01
The team identified a non-cited violation of 10CFR Part 50, Appendix B, Criterion III, for the failure to translate the Fort Calhoun Station raw water strainer component’s design basis into specifications, procedures, and instructions. The licensee had classified the raw water strainer components as non-safety related parts. The raw water strainers are equipment necessary to ensure that nuclear safety functions provided by Safety Class 1, 2, or 3 equipment (raw water system) are capable of accomplishing those functions. The licensee has entered this finding into their corrective action program as Condition Report 2007-3046.

This finding is more than minor because it affected the mitigating system cornerstone objective (design control attribute) to ensure the reliability and capability of the raw water system to mitigate initiating events such that the raw water strainer function was necessary and relied upon for ensuring the nuclear safety functions that are provided by Safety Class 1, 2, or 3 equipment. Using Manual Chapter 0609, Phase 1 screening worksheet, the issue screened as having very low safety significance because it was a design or qualification deficiency confirmed not to result in a loss of operability per Part 9900, Technical Guidance, “Operability Determination Process for Operability and Functional Assessment.
Inspection Report# : 2007007 *(pdf)*

Significance: G Jul 25, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Inadequate Corrective Actions for the Turbine Driven Auxilary Feedwater Keep Warm Line Bypass Throttle Valves MS-366 and -368
The team identified a noncited violation of 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, for failure to promptly identify and correct conditions adverse to quality. Specifically, between November 11, 2005, to April 28, 2006, during quarterly surveillance tests of the steam bypass warmup valves for the Turbine Driven Auxiliary Feedwater pump, the licensee noted degrading conditions (change in flow coefficient, Cv) of the bypass warmup valves. The degraded bypass warmup valves allowed the throttle valve differential pressure to fall below established acceptance criteria. During a postulated steam line break, the deteriorated bypass warmup valves could pass more steam than designed. Passing more steam than previously analyzed through a pipe break would not meet the licensee’s commitment to maintain a mild environment to Room 19, where the auxiliary feedwater pumps are located, and, therefore would not ensure the operability of the safety-related equipment in the room. This issue was entered into the corrective action program as Condition Report 2007-2489.

The failure of allowing the turbine driven auxiliary feedwater pump bypass warmup valves to deteriorate to the point of allowing the throttle valve differential pressure to fall below established acceptance criteria, was a performance deficiency. This issue was more than minor because it affected the Mitigating Systems cornerstone objective of equipment reliability. The failure to ensure safety equipment performance is available and capable to respond to initiating events is a violation. Using the Manual Chapter 0609, Phase 1 screening worksheet, the issue screened as having very low safety significance, because it was a design deficiency in which there have been no actual loss-of-safety function, in accordance with NRC Manual Chapter Part 9900, Technical Guidance, Operability Determination Process for Operability and Functional Assessment. The finding had a cross-cutting aspect in the area of human performance decision making (H.1.b). The licensee had opportunities to identify and correct the degraded bypass warmup valve when it caused the throttle valve differential pressure to fall below established acceptance criteria, but had not perform a thorough review of the concern.
Inspection Report# : 2007007 *(pdf)*

Significance: G Jul 25, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Meet Single Failure Criteria Configuration for Component Isolation Valves
The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to meet the single valve failure requirements for the component cooling water surge tank. The component cooling water surge tank water and nitrogen supply lines were credited with only a single check valve for meeting single failure criteria requirements. Based on engineering review, this configuration was not considered acceptable. Manual Isolation Valves AC-1179 and NG-290 have now been administratively changed in accordance with the Safety Analysis for Operability from the normally open position to the normally closed position to meet the single failure criteria requirements for the component cooling water Surge Tank AC-2. Upstream Check Valves AC-391 and NG-113 were previously credited with meeting the single failure criteria. This issue was entered into the corrective action program as Condition Report 2007-2622.

The failure to comply with ANSI 51.1, "Nuclear Safety Criteria for the Design of Pressurized Water Powerplants," with respect to single failure criteria (double isolation) for the demineralized water and Nitrogen makeup lines to the Component Cooling Water (CCW) surge tank is a performance deficiency. This finding is more than minor because it affected the mitigating system cornerstone objective (design control attribute) to ensure the reliability and capability

of the equipment needed to mitigate initiating events. Using the Phase 1 worksheet in Manual Chapter 0609, "Significance Determination Process," this finding is determined to be of every low safety significance because there was no actual loss of a safety function.
Inspection Report# : 2007007 *(pdf)*

Significance: G Mar 31, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Promptly Identify and Correct an Inoperable component Cooling Water flow Element
A Green self-revealing noncited violation was identified for the licensee's failure to promptly identify and correct a repetitively inoperable component cooling flow element. The initial failure occurred in 1999 and had failed three times within the past two years. The failure to recognize and fix this condition led to the flow element repeatedly being out of service and unable to perform its function during a potential design basis accident.

This finding was determined to be greater than minor because the condition had an impact on availability/reliability of the component and thus affected the "Equipment Performance" attribute under the Mitigating Systems cornerstone. The inspectors evaluated this finding using Manual Chapter 0609, Appendix A, and determined that it was of very low safety significance (Green). This conclusion was reached because the finding was not a design or qualification deficiency, the finding did not represent a loss of safety function, was not an actual loss of safety function of a single train for greater than its Technical Specification Allowed Outage time, did not represent an actual loss of safety function for non-Technical Specification equipment, and was not potentially significant due to external events such as flooding, seismic occurrences, etc. This violation was entered into the licensee's corrective action program as Condition Report 200605986. This finding has a crosscutting aspect in the area of problem identification and resolution associated with corrective action because the licensee failed to identify and correct the condition despite numerous opportunities to do so. This crosscutting aspect is indicative of current performance because the most recent failure of the flow element occurred in December 2006.
Inspection Report# : 2007002 *(pdf)*

Significance: W Dec 21, 2006
Identified By: NRC
Item Type: VIO Violation
Containment Spray Train 'B' Inoperable in Excess of Technical Specifications due to Failure to Perform Adequate Maintenance and Testing
A violation of 10 CFR Part 50, Appendix B, Criterion V, was identified for the OPPD's failure to perform adequate maintenance and testing on containment spray header isolation Valve HCV-345. This issue was self revealed on September 13, 2006, when reactor coolant water issued from the containment spray headers indicating that either Valve HCV-344 or Valve HCV-345 was not properly seated. The failure to perform adequate maintenance and testing for this component resulted in one train of containment spray being inoperable from May 11, 2005 to September 9, 2006, a period of 454-days. This exceeded Technical Specification 2.4(2) allowed outage time of 24 hours when the reactor is critical.

The issue was more than minor because it affected the equipment performance attribute of the Mitigating System Cornerstone due to the impact on availability and reliability of the containment spray system. The finding was characterized under the significance determination process as having low to moderate safety significance because one train of containment spray was unavailable to respond to a loss-of-coolant accident and would have been unable to perform its mitigating system function. This condition was entered into the OPPD's corrective action program as Condition Report 200604627. The finding has a crosscutting aspect in the area of human performance, specifically resources, in that complete and accurate procedures and work packages were not provided.
Inspection Report# : 2006018 *(pdf)*

Barrier Integrity

Emergency Preparedness

Occupational Radiation Safety

Significance: G Jun 30, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Stop Work and Notify RP Upon Receiving a Dose Rate Alarm
A self-revealing noncited violation of Technical Specification 5.8.1(a) was identified because a worker failed to stop work and notify radiation protection upon receiving a dose rate alarm per procedural requirements. On March 24, 2007, a chemist received an electronic alarming dosimeter dose rate alarm while performing an instrument calibration. The individual received a peak dose rate of 186 millirem per hour and the dose alarm set point was 120 millirem per hour. The chemist did not self-check subsequent actions when problems arose with the calibration source and did not notify radiation protection of the alarm until after exiting from the Radiologically Controlled Area. The worker was coached and received remedial radiation worker training. This finding was entered into the licensee's corrective action program.
The failure to follow a station procedure is a performance deficiency. This finding is greater than minor because it is associated with one of the cornerstone attributes (exposure control) and affected the Occupational Radiation Safety cornerstone objective, in that the failure to follow the station procedure resulted in additional personnel exposure. The inspectors used the Occupational Radiation Safety Significance Determination Process and determined that this finding was of very low safety significance because it did not involve: (1) an ALARA finding, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose. In addition, this finding had a human performance crosscutting aspect associated with work practices because the chemist did not use human error prevention techniques, such as self-checking (H.4(a))
Inspection Report# : 2007003 *(pdf)*

Significance: SL-IV May 16, 2007
Identified By: NRC
Item Type: VIO Violation
Failure to Follow Radiation Work Permit Instructions
On at least three occasions between November 26, 2005, and March 27, 2006, a security officer deliberately failed to proceed to the radiation controlled area reader and log in following the instructions on the keypad, and confirm that the electronic alarming dosimeter is on and reading zero prior to assuming his post on the roof of the radioactive waste building which was posted as a radiation area inside the radiation controlled area. In addition, the individual enlisted the aid of two other security officers to return his radiation controlled area to access control in order to expedite his departure at the end of the shift. These occurrences are violations of Technical Specification 5.8.1.a which states, in part, that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, February 1978, Appendix A. Regulatory Guide 1.33, Appendix A, Section 7.e.(1), recommends procedures for access control to radiation areas including a radiation work permit system.

The failure to follow radiation work permit instructions is a performance deficiency. Because there are willful aspects of the violation, it is subject to traditional enforcement. NRC management determined this to be a severity level IV violation.
Inspection Report# : 2007009 *(pdf)*

Significance: G Mar 31, 2007
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Failure to Obtain a High Radiation Area Access Authorizations and Associated Radiological Briefing
The inspectors reviewed two examples of a self-revealing, noncited violation of Technical Specification 5.11.1 in which workers failed to obtain high radiation area radiological briefing before entering the area. The first example occurred on October 25, 2006, when a worker received an electronic alarming dosimeter dose alarm while performing

duties as a fire watch on one of the steam generator platforms, which was posted as a high radiation area. The second example occurred on October 29, 2006, when a worker received an electronic alarming dosimeter dose alarm while pulling electrical cable inside the bioshield, which was posted as a high radiation area. For both issues, the licensee restricted access to the radiologically controlled area pending discussion with the individuals and their supervisors. This issue was also included as preshift briefings and management meetings to heighten the awareness of changing radiological conditions and for workers to be more mindful of the radiation work permit requirements.

This finding is greater than minor because it is associated with one of the cornerstone attributes, exposure/contamination control, and affects the Occupational Radiation Safety cornerstone objective in that the failure to obtain high radiation area access authorization and the associated radiological briefings could have resulted in additional personnel exposure. Using the Occupational Radiation Safety Significance Determination Process, the inspectors determined that this finding was of very low safety significance because it did not involve: (1) an as low as reasonably achievable planning or work control issue; (2) an overexposure; (3) a substantial potential for overexposure; or (4) an impaired ability to assess dose. This finding also has a crosscutting aspect in the area of human performance work control because neither the individuals nor their supervisors appropriately coordinated work activities and evaluated the impact of changes to work assignments. These issues have been entered into the licensee’s corrective action program as Condition Report 200604937 and Condition Report 200605033.
Inspection Report# : 2007002 *(pdf)*

Public Radiation Safety

Physical Protection

Although the NRC is actively overseeing the Security cornerstone, the Commission has decided that certain findings pertaining to security cornerstone will not be publicly available to ensure that potentially useful information is not provided to a possible adversary. Therefore, the cover letters to security inspection reports may be viewed.

Miscellaneous

Last modified : February 04, 2008