Oconee 3
2Q/2006 Plant Inspection Findings

Initiating Events

Significance: G Jun 30, 2006
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Loss of Unit 3 Offsite Power During Mode 6.
self-revealing non-cited violation of Technical Specification (TS) 5.4.1 was identified for failure to adequately implement the procedure requirements for protected train equipment, resulting in the lockout of CT3 transformer and subsequent loss of Unit 3 power while in Mode 6.

The inspectors determined that the licensee's failure to adequately implement their procedure for protected train equipment was a performance deficiency. The finding was considered to be more than minor because it affected the initiating events cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions. The finding was determined to be of very low safety significance. This was based on the screening criteria found MC 609, Appendix G, Attachment 1, Checklist 4, Pressurized Water Reactor (PWR) Refueling Operation: Reactor Coolant System (RCS) Level > 23' or PWR Shutdown Operation with time to Boil > 2 Hours and Inventory in the Pressurizer. This finding did not meet the criteria in the checklist for requiring a phase 2 or 3 analysis, in that it did not increase the likelihood of a loss of RCS inventory, did not degrade the licensee's ability to terminate a leak path or add inventory, or degrade the licensee's ability to recover Decay Heat Removal (DHR) once it is lost.

This finding has a cross-cutting aspect in the area of human performance because the licensee's planned work activities did not effectively keep personnel apprised of the operational impact of the work due to the inadequate implementation of their protected train procedure. (Section 4OA3)

Inspection Report# : 2006003*(pdf)*

Significance: G Sep 30, 2005
Identified By: Self-Revealing
Item Type: FIN Finding
Inadequate Maintenance and Oversight Increased the Likelihood of a Unit 3 Reactor Trip with a Loss of Normal Heat Sink
A self-revealing finding was identified for inadequate maintenance and oversight of repair efforts on the actuator of 3DW-18 (the Unit 3 Upper Surge Tank (UST) Makeup Valve). Specifically, while attempting to repair an air leak on the actuator of 3DW-18, maintenance technicians removed the valve's bonnet and were ready to remove the valve's diaphragm with no hydraulic isolations made between the valve and the main condenser. Had the diaphragm been removed from 3DW-18, it is likely that Unit 3 would have tripped due to a loss of main condenser vacuum, as the top of the UST dome is vented to the main condenser. This event was considered to be a performance deficiency, as the licensee failed to provide adequate maintenance and oversight of the efforts to repair an air leak on the 3DW-18 actuator; thereby, increasing the likelihood of a unit trip with a loss of normal heat sink. This issue was considered to be more than minor because it affected the Initiating Events cornerstone objective of limiting the likelihood of events that upset plant stability. The finding is associated with the configuration control attribute, in that the inadequate maintenance and oversight of the repairs to the actuator of 3DW-18 increased the likelihood of a reactor trip with a loss of normal heat sink due to inadequate configuration control of a secondary plant system. The consequences of the finding were assessed through Phase 2 of the SDP, and although the likelihood of a unit trip was increased and would have resulted in a loss of the normal heat sink, the exposure time for this condition was less than 3 days and all other mitigation capabilities described on the Phase 2, SDP worksheet for transient (reactor trip) core damage sequences were maintained. Consequently, the finding was determined to be of very low safety significance. This finding involved the cross-cutting aspect of human performance. (Section 1R13)
Inspection Report# : 2005004*(pdf)*

Significance: G Sep 30, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Implement an Inspection Program for the Main Steam Lines
A NRC-identified non-cited violation of 10 CFR 50 Appendix B, Criterion X, Inspection, was identified for the failure to develop and implement an inspection program for monitoring the main steam line in the Unit 1, 2 and 3 East Penetration Rooms. The finding was considered to be a performance deficiency in that the licensee had committed to perform inspections of the steam lines to support the acceptability of Duke's design and analysis for the main steam lines, but the inspections were not being performed. The finding was considered to be more than minor because it impacted the Reactor Safety Initiating Events Cornerstone in that failure to perform the inspections could lead to failure to identify degrading main steam line conditions, which would cause an increase in the likelihood of an initiating event. The finding was screened as having very low safety significance under the Initiating Events Cornerstone, in that it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available. This finding involved the cross-cutting aspect of Problem Identification and Resolution. (Section 1R22.3)

Inspection Report# : 2005004*(pdf)*

Significance: G Sep 09, 2005
Identified By: Self-Revealing
Item Type: FIN Finding
Inadequate Unit 3 Digital Control Rod Drive System Modification
A self-revealing finding was identified for using an undersized breaker in an inadequate modification of the Unit 3 control rod drive system which led to a reactor trip during routine maintenance of the alternate power supply breaker 2X2-5D. The finding is greater than minor because it affected the initiating events cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions, in that the undersized power supply breaker led to a reactor trip when the digital control rod drive control system (DCRDCS) was placed in a single power supply configuration during maintenance on the alternate feeder breaker for the Unit 3 DCRDCS. The finding was determined to be of very low safety significance, since it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available. The licensee modified the system with a suitable breaker. (Section 4OA3.2)
Inspection Report# : 2005010*(pdf)*

Significance: G Sep 09, 2005
Identified By: NRC
Item Type: FIN Finding
Inadequate corrective actions for an identified deficiency with the Unit 3 Digital Control Rod Drive System
The inspectors identified a finding for the failure to adequately assess and correct the adverse impact of an identified vulnerability with the digital control rod drive system to integrated control system (ICS) interface following a loss of all power to the rod control system. This condition led to the overcooling of the reactor coolant system and an engineered safeguards (ES) actuation following a reactor trip. The finding is greater than minor because it affected the initiating events cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions, in that the uncorrected DCRDCS to ICS interface resulted in the overcooling of the reactor coolant system (RCS) and subsequent ES actuation. ES actuation invokes a probability that operators will not manually terminate or reduce high pressure injection flow (HPI) prior to lifting a primary relief valve that, if failed to reseat, would lead to a primary system loss of coolant accident (LOCA). A phase 2 analysis was performed with results greater than green and a subsequent phase 3 analysis is in progress. This finding also involved the crosscutting aspect of problem identification and resolution. The licensee has modified the system to remove the vulnerability. (Section 4OA3.3) Subsequently, a regional Senior Reactor Analyst performed a Phase 3 risk evaluation of the issue and determined it to be of very low risk significance (Green). This was based primarily on the relatively low probability that the operators would fail to throttle or secure high pressure injection following the engineering safeguards actuation, coupled with a negligible human error rate dependance to subsequent operator actions for initiating high pressure recirculation. (IR 2006-02, Section 4OA5.1)
Inspection Report# : 2005010*(pdf)*
Inspection Report# : 2006002*(pdf)*

Mitigating Systems

Significance: G Jun 30, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to promptly correct long-standing east penetration room blowout panel-related deficiencies that preclued flood mitigation in the auxiliary buidling.
NCV of 10 CFR Part 50, Appendix B, Criteria XVI, Corrective Action, for failure to promptly identify and correct a significant condition adverse to quality. Specifically, as a result of inappropriate east penetration room blowout panel modifications (identified as a violation in 2002), in conjunction with the inappropriate addition of floor curbing and the inadequate strength of internal doors and block walls (all identified in DEC's corrective action program in 2001), Units 1, 2, and 3 continue to be operated outside their licensing basis with respect to HELB-related flood mitigation in the auxiliary building.
Inspection Report# : 2006014*(pdf)*

Significance: SL-IV Jun 30, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to report east penetration room blowout pante-related deficiencies would prevent fulfillment of the HPI system safety function.
NCV of 10 CFR 50.73, Part (v) was identified for the failure to report that east penetration room blowout panel-related deficiencies would prevent the fulfillment of the HPI system safety function to mitigate the consequences of a HELB (i.e., to shutdown the reactor and maintain it in a cold shutdown condition).
Inspection Report# : 2006014*(pdf)*

Significance: G Jun 30, 2006

Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Perform Adequate Examinations of Letdown Filter Supports.
The inspectors identified a finding involving a non-cited violation of 10 CFR Part 50.55a(g)(4)ii for failure to perform a visual (VT-3) examination of the letdown filter housing supports as required by Section XI of the ASME Code. The examinations were performed with a remote camera and the required examination coverage was not obtained as required by Section XI of the ASME Code. The limited remote VT-3 examinations found no indications that the structural integrity of the supports was unacceptable for service. The licensee entered this issue into the Corrective Action Program.

This finding was of more than minor significance because the incomplete examination of the letdown filter housing supports, if left uncorrected, could become a more significant structural support concern. In addition, a failure to examine the letdown filter supports as required by the ASME Code is related to the “Equipment Performance” attribute of the “Initiating Events” cornerstone and affects the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown, as well as power operations. This finding was evaluated using Phase 1 of the NRC IMC 0609, “Significance Determination Process (SDP).” This finding was of very low safety significance because the worst case degradation of the letdown filter supports would result in a detectable and isolable RCS leak that would not impair the mitigating function of the high pressure injection (HPI) system. (Section 1R08)

Inspection Report# : 2006003*(pdf)*

Significance: G Jun 30, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
Failure To Maintain Containment Electrical Penetration Enclosures.
The inspectors identified a non-cited violation of 10 CFR 50 Appendix B, Criterion XVI for failure to identify a condition adverse to quality in that East and West Penetration Room containment electrical penetration enclosures had not been maintained, such that a number of enclosures allowed the introduction of dirt and debris inconsistent with conditions under which these penetrations were environmentally qualified.

The finding was considered to be a performance deficiency in that the licensee failed to maintain the containment electrical penetration covers such that debris was allowed to accumulate in a number of enclosures; thereby, jeopardizing the environmental qualification of safety-related circuits. This finding was considered to be more than minor because it affected the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events in that, the degraded penetration covers, if left uncorrected could allow the environmental qualification of safety-related circuits to degrade such that they would fail following a high energy line break (HELB) in the east penetration rooms. Using the phase 1 screening worksheet of Manual Chapter 0609, Appendix A, the finding was determined to be of very low safety significance, as it did not result in a loss of operability of any equipment needed to mitigate the effects of a HELB.

This finding has a cross-cutting aspect in the area of problem identification and resolution, as the licensee did not appropriately identify the degraded penetration covers consistent with their corrective action program. (Section 4OA5.1)

Inspection Report# : 2006003*(pdf)*

Significance: G Mar 17, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
Non-Conservative EOP Procedure Setpoint for Operator Action to accomplish BWST to RBES swap over on Low BWST Level
The team identified a Green, non-cited violation (NCV) of Technical Specification 5.4.1.b for a non-conservative operator action setpoint in the Emergency Operating Procedures. Specifically, the 6 foot level setpoint for operator action to complete the BWST to Reactor Building Emergency Sump (RBES) swap over by closing the BWST suction valves did not include enough margin to preclude degradation or damage to the pumps due to vortex formation in the BWST in all cases. When the NRC notified the licensee of this condition, the licensee entered it into the corrective action program. This finding is greater than minor because it is associated with the procedure quality attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring reliable, available, and capable systems that respond to initiating events to prevent undesirable consequences. This finding is of very low safety significance because no actual loss of safety function occurred and operators have been trained to identify loss of pump suction. This finding has been entered into the licensee’s corrective action program as PIP O-06-01374. (Section 1R21.2.1.1)
Inspection Report# : 2006006*(pdf)*

Significance: G Feb 15, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Promptly Identify and Correct a Long-Standing Discrepancy Between the Unit 3 Control Room and its Tornado Licensing Basis

Specified in the Updated Final Safety Analysis Report
NCV for failure to promptly identify and correct a long-standing discrepancy between the Unit 3 control room and its tornado licensing basis specified in the Updated Final Safety Analysis Report: (1) a failure to take adequate corrective actions to bring the Unit 3 control room (i.e., north control room wall) within its licensing basis to withstand the effects (wind force, missiles, and differential pressure) of differing tornado intensities; (2) inadequate corrective actions involving the inappropriate use of 50.59 to remove the Unit 3 control room tornado missile requirements from the UFSAR.
Inspection Report# : 2006009*(pdf)*

Significance: Dec 31, 2005
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Inadequate Procedures for Testing the SSF Diesel Generator With the CCW Supply Secured
A Green self-revealing non-cited violation was identified for failure to have adequate procedures for testing the Standby Shutdown Facility (SSF) diesel generator as required by Technical Specification (TS) 5.4.1. The licensee's existing test procedures did not establish the appropriate plant conditions with the Unit 2 condenser cooling water (CCW) system shut down such that the water supply to the SSF auxiliary service water (ASW) and station ASW heated above 90 degrees F rendering both unavailable for all three units. The licensee entered this finding into their corrective action program under Problem Investigation Process report (PIP) O-05-7479. This finding was considered to be of more than minor significance because it affected the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences, as the elevated temperature of the SSF ASW and station ASW supply resulted in the unavailability of these systems. This issue was determined to be of very low safety significance based on the screening criteria found in MC 0609, Appendix A, Phase 1 SDP worksheet. More specifically, the total additional unavailability of the SSF (one day) as result of overheating the supply did not exceed the TS allowed outage time. (Section 1R12)
Inspection Report# : 2005005*(pdf)*

Significance: Dec 31, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Identify Unmitigated/Unprotected Feedwater Line Terminal Ends
A NRC-identified non-cited violation of 10 CFR 50 Appendix B, Criterion XVI was identified for the failure to identify a condition adverse to quality, in that feedwater terminal ends had not been identified; and therefore, actions to mitigate the affects from a terminal end line break had not been implemented. The licensee entered this finding into their corrective action program under PIP O-06-00138. This finding was considered to be more than minor because an unprotected terminal end line break would impact the Reactor Safety Cornerstone for Mitigating Systems associated with the availability, reliability and function of systems needed to respond to a high energy line break (HELB). This issue was determined to be of very low safety significance based on a very low initiating event frequency being calculated as a result of the limited number of welds and feet of pipe under consideration. In addition, the large early release frequency impact was below the threshold, because of the size of break required to damage the containment penetration was an even lower probability event. This finding involved the crosscutting aspect of Problem Identification and Resolution. (Section 4OA5.2)
Inspection Report# : 2005005*(pdf)*

Significance: Sep 30, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
Performing Licensed Duties While Medically Unqualified
A NRC-identified non-cited violation of 10 CFR 50.74 was identified for failure to make a notification of a change in operator or senior operator status regarding information for one licensed operator concerning his medical qualification. Specifically, the operator failed to meet the American Nuclear Standards Institute /American Nuclear Society (ANSI/ANS-3.4, "Medical Certification and Monitoring of Personnel Requiring Operator Licenses for Nuclear Power Plants," 1983 Standard for a blood pressure (BP) limitation. This impacted the NRC's ability to perform its regulatory function, in that the NRC was not able to make a licensing decision with regards to a potential restriction to ensure compliance with ANSI/ANS-3.4. Consequently, an operator stood several watches in a Technical Specification license position with his BP greater than the ANSI/ANS limits. This finding is of very low safety significance because there was no evidence that the operator endangered plant operations as a result of hypertension while performing licensed duties since the original issuance of his license. However, the regulatory significance was important because pertinent information was not provided to the NRC when the operator knowingly discontinued taking his medication. Subsequently, this impacted a licensing decision for the individual. (Section 1R11.2)
Inspection Report# : 2005004*(pdf)*

Significance: Sep 30, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Develop and Implement a Cleanliness Inspection Program for the Containment Electrical Penetrations
A NRC-identified non-cited violation of 10 CFR 50 Appendix B, Criterion X, Inspection, was identified for the failure to develop and implement an inspection program for inspection and cleaning of the containment electrical penetrations located in the East and West Penetration Rooms of Units 1, 2, and 3. The finding was considered to be a performance deficiency in that the licensee had failed to develop an inspection program for their

containment electrical penetrations to ensure cleanliness of the electrical connections. The inspectors concluded that if left uncorrected (no inspection) debris and rust accumulation could lead to failure of the electrical circuits during a high energy line break as a result of grounds and shorts. Therefore, failure to perform cleanliness inspections was considered to be more than minor because it could impact the Reactor Safety Mitigating Systems Cornerstone objective for reliability of a mitigating system/train (i.e., circuits needed to mitigate a high energy line break. The finding was screened as very low safety significance in the Phase 1 review under the Mitigating Systems Cornerstone, in that failure to perform an electrical penetration inspection was not considered to be a design deficiency, was not considered to represent a loss of safety system function, was not considered to represent an actual loss of safety function of a single train, and did not involve seismic, flooding or severe weather. This finding involved the cross-cutting aspect of Problem Identification and Resolution. (Section 1R22.2)
Inspection Report# : 2005004*(pdf)*

Significance: G Sep 09, 2005
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Failure to Have a Written Procedure for the Restoration of ES Components
A self-revealing NCV was identified for failure to have a restoration procedure for ES components as required by Technical Specification (TS) 5.4.1. Following a valid actuation signal, the licensee did not restore ES channels 1 and 2 to operable status for more than seven hours due to a lack of specific procedural direction.
The finding is greater than minor because it adversely impacted the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences and affected the human performance attribute of the same cornerstone. Specifically, the failure to have a restoration procedure resulted in the operators using their knowledge and skill of the craft to restore ES alignment and resulted in the Safety Injection actuation circuitry being inoperable for more than seven hours. The licensee has entered this finding into their corrective action program and was reviewing a new draft procedure, "ES Recovery," at the time of the inspection. (Section 4OA3.4.b(1))
Inspection Report# : 2005010*(pdf)*

Significance: G Sep 09, 2005
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Failure to maintain Enclosure 5.1, ES Actuation
A self-revealing NCV was identified for not maintaining Emergency Operating Procedure (EOP) Enclosure 5.1, "ES Actuation," in accordance with TS 5.4.1. Enclosure 5.1 contained unnecessary steps to open BS-1 and BS-2 which allowed borated water storage tank (BWST) water to drain to the reactor building normal sump and compelled the operators to take actions outside their written EOP guidance to secure the loss of water to the sump. The finding is considered to be of more than minor significance because it adversely impacts the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences and affected the procedure quality and human performance attributes of the same cornerstone. Specifically the failure to remove the steps to open the valves from the procedure resulted in operators having to take actions outside the procedure to stop the loss of BWST water. (Section 4OA3.4.b(2))
Inspection Report# : 2005010*(pdf)*

Significance: N/A Jan 23, 2004
Identified By: NRC
Item Type: VIO Violation
Failure to Obtain Prior NRC Approval to a Change to the Facility Involving Unreviewed Safety Questions on High Energy Line Break Analysis
The inspectors identified an apparent violation of 10 CFR 50.59 (a)(1) (1999 version of 10 CFR) which states, in part, that the licensee may make changes in the facility as described in the safety analysis report without prior Commission approval, provided the proposed change does not involve an unreviewed safety question (USQ). 10 CFR 50.59 (a)(2) states, in part, that a proposed change involves an USQ if the probability of occurrence or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased, or if it may create an accident different from any previously evaluated.

On May 17, 2001, the licensee made a change to the facility, as described in the Updated Final Safety Analysis Report, Section 3.6.1.3, associated with the High Energy Line Break (HELB) analysis, which involved unreviewed safety questions, and failed to obtain prior NRC approval. The UFSAR Section was changed to increase the maximum initiation time following HELB of Emergency Feedwater from 15 to 30 minutes and of High Pressure Injection from 1 hour to 8 hours (based on referenced reports and analysis). The analysis discussed an increased cycling of pressurizer Safety Relief Valves on steam and water, boiler condenser mode of decay heat removal, and an unapproved computer code for application to HELB, but failed to recognize that such changes may increase the probability of occurrence or the consequences of a malfunction of equipment important to safety or may create an accident different from any previously evaluated. In addition, the change resulted in more than a minimal increase in risk.

Based on the results of the inspection, a pre-decisional enforcement conference was held on March 2, 2004, in the NRC's Region II Office in Atlanta, Georgia, with the licensee staff to discuss the apparent violation, its significance, root causes, and corrective actions. Based on the information developed during the inspection, and the information presented at the conference, the NRC determined that a violation of NRC requirements occurred. On April 8, 2004, the NRC issued a Notice of Violation (NOV) and proposed imposition of a $60,000 Civil Penalty (ADAMS accession number ML040990355). The violation involves a failure to adhere to the requirements of 10 CFR 50.59, in that Duke Energy Corporation made changes to the Oconee facility as described in Section 3.6.1.3 of the UFSAR and referenced analyses that involved unreviewed safety questions (USQs) without obtaining prior NRC approval.

Inspection Report# : 2004005*(pdf)*
Inspection Report# : 2004007*(pdf)*
Inspection Report# : 2005002*(pdf)*
Inspection Report# : 2005005*(pdf)*

Barrier Integrity

Significance: SL-IV Mar 31, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Provide Complete and Accurate Information to the NRC
An in-office review of the results of NRC Office of Investigations (OI) Report No.: 2-2005-016, identified a non-cited violation of 10 CFR 50.9 for failure to provide complete and accurate information in Licensee Event Report (LER) 05000287/2001-001, regarding the condition of the Unit 3 Reactor Pressure Vessel Head (RPVH), resulting from boric acid leakage. The LER stated that boric acid leakage caused no detectable corrosion to the vessel head, when in fact some minor corrosion was identified. The licensee corrected the incomplete and inaccurate information with a revision to LER 05000287/ 2001-001, dated August 18, 2005. Because this issue potentially affected the NRC's ability to perform its regulatory function, it was evaluated using the traditional enforcement process. The failure to provide accurate and complete information precluded the NRC from being able to pursue or consider further inquiry or inspection activity in regards to RPVH corrosion, the significance of which was not known at the time. NRC review determined that there was no evidence that the licensee's actions were willful. Additionally the NRC determined that the corrosion was not structurally significant and would not have resulted in a regulatory action or substantial further inquiry. (Section 4OA5.3)
Inspection Report# : 2006002*(pdf)*

Emergency Preparedness

Occupational Radiation Safety

Public Radiation Safety

Significance: G Dec 31, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Ensure Adequacy of Measurements of Particulate Effluents Released from Unit Vent
An NRC-identified NCV of 10 CFR 20.1302(a) was identified for failure to ensure surveys of particulate radioactive materials in effluents released to unrestricted areas by the unit vents were adequate to demonstrate compliance with dose limits for individual members of the public. The failure to conduct appropriate evaluations to assure representative sample collection from the Unit 1, 2, and 3 unit vent exhaust streams when sampled through the tee connections on the sample line to 1,2,3-RIA-43 and the elbow connections on the associated Selected Licensee Commitment required unit vent particulate sampler lines could result in inaccurate measurement of airborne particulate radionuclides in effluent samples, potentially leading to effluent releases exceeding allowed concentrations or dose limits to members of the public. This finding was entered into the licensee's corrective action program as PIPs O-04-7084 and O-05-4874. The licensee has approved and scheduled installation of a design modification for the monitors that will remove the non-conforming bends and replace them with bends of radius greater than or equal to five times the size of the diameter of the sample lines. This finding is greater than minor because it is associated with the program and process attribute of the Public Radiation Safety Cornerstone and affects the cornerstone objective of assuring adequate protection of public health and safety from exposure to radioactive materials released into the public domain as a result of routine civilian nuclear reactor operation. This finding which involved radioactive material control was assessed using the Public Radiation Safety SDP. Since the finding did not result in the failure to assess dose, due to the licensee having other means by which dose from particulate releases could be assessed, and because the licensee did not exceed the limits in 10 CFR 50 Appendix I or 10 CFR 20.1301(d), it was determined to be of very low safety significance. The cause of the finding is related to the cross-cutting element of Problem Identification and Resolution. (Section 2PS1)
Inspection Report# : 2005005*(pdf)*

Significance: G Dec 31, 2005
Identified By: Self-Revealing
Item Type: NCV NonCited Violation

Failure to Adequately Survey and Maintain Control of Licensed Material (Section 2PS3)
A self-revealing non-cited violation of 10 CFR 20.1501(a) and 10 CFR 20.1802 was identified for an inadequate survey of contaminated equipment and failure to control and maintain constant surveillance of licensed material when free-released contaminated equipment was subsequently shipped to two locations as "clean" material without appropriate radiological controls. One of the locations was a non-licensed individual possessing neither the training nor equipment necessary to identify and control the contaminated material. The licensee entered the finding into the corrective action program as PIP O-04-8873. The corrective actions associated with this PIP included sending a radiological response team to one of the locations to identify, contain, and decontaminate any contaminated equipment and performing a detailed root cause analysis of the event. The finding is greater than minor because it is associated with the human performance attribute of the Public Radiation Safety Cornerstone and affects the cornerstone objective of assuring adequate protection of public health and safety from exposure to radioactive materials released into the public domain as a result of routine civilian nuclear reactor operation. The failure to conduct adequate surveys resulted in the free-release of contaminated equipment, potentially leading to exceeding the dose limits to members of the public through loss of control of licensed material. This finding which involved radioactive material control was assessed using the Public Radiation Safety SDP. Since the finding neither resulted in an exposure to the public in excess of five millirem nor involved greater than five occurrences, it was determined to be of very low safety significance. The cause of this finding is related to the cross-cutting element of Human Performance. (Section 2PS3)
Inspection Report# : 2005005*(pdf)*

Physical Protection

Physical Protection information not publicly available.

Miscellaneous

Significance: N/A Jul 01, 2005
Identified By: NRC
Item Type: FIN Finding
Biennial PI&R Inspection
The inspectors concluded that, in general, problems were properly identified, evaluated, and corrected. The licensee was effective at identifying problems and entering them into the corrective action program (CAP) for resolution; however, several minor plant material condition deficiencies were identified during plant system walkdowns that had gone undetected by licensee personnel. The licensee maintained a low threshold for identifying problems as evidenced by the continued large number of Problem Investigation Process reports (PIPs) entered annually into the CAP. Generally, the licensee properly prioritized issues and examined issues; although several minor problems were noted where lower significance issues were mis-categorized or the investigations lacked thoroughness. Formal root cause evaluations for significant problems were thorough and detailed. Corrective actions specified for problems were generally adequate; although, several minor problems were noted where corrective actions were not complete or not comprehensive. Audits and self-assessments were effective in identifying deficiencies in the CAP. Personnel at the site felt free to raise safety concerns to management and to resolve issues via the CAP.
Inspection Report# : 2005008*(pdf)*

Last modified : August 25, 2006