Oconee 3
4Q/2005 Plant Inspection Findings

Initiating Events

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: TBD 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)
Inspection Report# : 2005010*(pdf)*

Significance: G Jun 30, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Follow Procedure Requirements for Replacing the Seismic Trigger System Batteries
A NRC-identified non-cited violation of Technical Specification (TS) 5.4.1 was identified for failure to follow the procedure requirements in replacing the Seismic Trigger System batteries. The inspectors determined that the failure to follow procedure in replacing the batteries as required, the inadequate procedure for centering the masses, and use of an unapproved procedure to perform the calibrations collectively represented a performance deficiency because the licensee is required to follow procedures, have procedures with adequate acceptance criteria and to use approved procedures. The finding was considered to be more than minor in that it was concluded by the inspectors that failure to follow the procedure requirements of replacing the batteries could render the seismic switch and therefore, the seismic monitors inoperable, if the batteries failed after their expiration dates. Thus if left uncorrected the finding would become a more significant safety concern because this equipment is used to determine whether or not the units need to be shutdown following a seismic event. In addition, it was concluded that the finding affected the reliability of systems that respond to initiating events in that it could affect the post event operating procedures (Abnormal Operating Procedures (AOPs) and Emergency Operating Procedures (EOPs)) for responding to a seismic event. The finding was screened using the Phase 1 screening criteria specified under Seismic, Flooding and Severe Weather Screening Criteria. The inspectors concluded that whether or not the failure to replace the batteries represented a degradation of equipment, since the finding did not represent an actual loss of function, the issue would be screened as Green by Questions 2 and/or 3 of this section. (Section 1R22)
Inspection Report# : 2005003*(pdf)*

Mitigating Systems

Significance: G 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: G 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: TBD Dec 09, 2005
Identified By: NRC
Item Type: AV Apparent 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 UFSAR
Apparent Violation with two examples: (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# : 2005012*(pdf)*

Significance: G 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: G 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: G Jun 30, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
Inadequate corrective actions related to the identification of a failed KHU main step-up transformer cooling power contactor
A NRC-identified non-cited violation of 10 CFR 50 Appendix B, Criterion XVI, Corrective Action, was identified for inadequate corrective actions related to the timeliness of identification of a failed electrical contactor supplying one train of power to the Keowee Hydro Unit (KHU) main step-up transformer cooling systems, resulting in a reduction in reliability of the KHU overhead power path. The finding was considered to be more than minor because it affected the mitigating system cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events in that the reliability of the KHU overhead emergency power path was reduced for approximately a three week period. However, the cooling power to the transformer was maintained during this period; therefore, there was no actual loss of safety function for either the underground or overhead emergency power path. Consequently the finding was determined to be of very low safety significance. This finding also involved the cross-cutting aspect of problem identification and resolution. (Section 4OA2.3b.(5))
Inspection Report# : 2005003*(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: G Mar 31, 2005
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Inadequate Corrective Actions Following 3B RBCU Fan Failure Results in 2A RBCU Fan Failure
A self-revealing, non-cited violation of 10 CFR 50 Appendix B, Criterion XVI, Corrective Action, was identified for inadequate corrective actions following the 3B reactor building cooling unit (RBCU) fan blade failure, which led to the failure of a 2A RBCU fan blade. The finding was considered to be more than minor because it affected the barrier integrity cornerstone attribute of maintaining containment functionality, in that the failure to fully identify and correct the causes of the 3B RBCU fan blade failure resulted in a 2A RBCU fan blade failure less than eight months later. However, during an event requiring control of the containment environment with one RBCU inoperable, the two remaining RBCUs and two trains of reactor building spray would have been available to mitigate the consequences of the event; consequently, the finding was determined to be of very low safety significance using the SDP Phase 1 analysis. This finding also involved the cross-cutting aspect of problem identification and resolution. (Section 4OA2.2)
Inspection Report# : 2005002*(pdf)*

Emergency Preparedness

Significance: SL-IV Jun 30, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
Implementation of a Change to Emergency Action Level 4.7.U.2.1 which Decreased the Effectiveness of the Emergency Plan
A Severity Level IV non-cited violation (NCV) was identified for implementing a change which decreased the effectiveness of the emergency plan without prior NRC approval, contrary to the requirements of 10 CFR 50.54(q). The change involved Emergency Action Level 4.7.U.2.1 classification of "Natural Disasters, Hazards and Other Conditions Affecting Plant Safety." The finding was evaluated using the NRC's Enforcement Policy because licensee reductions in the effectiveness of its emergency plan impact the regulatory process. This finding has greater than minor significance in that the change extends the event time allowed prior to appropriate emergency classification of a natural disaster which could adversely affect the performance of both onsite and offsite emergency actions. The finding was determined to be a non-cited Severity Level IV violation because it involved licensee failure to meet an emergency planning requirement not directly related to assessment and notification. (Section 1EP4)
Inspection Report# : 2005003*(pdf)*

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 : March 03, 2006