Oconee 3
3Q/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)*

Mitigating Systems

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 Dec 31, 2004
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Incorrect Wiring of the SSF Submersible Pump Motor Leads
A self-revealing non-cited violation of 10 CFR 50, Appendix B, Criterion XI, Test Control, was identified for the failure to establish and perform adequate testing to ensure that the standby shutdown facility (SSF) submersible pump would operate correctly to provide SSF equipment with a makeup source of water to the Unit 2 condenser circulating water (CCW) header when called upon. Specifically, the licensee's test program had failed to reveal that the pump's power leads had been reversed since November 19, 1992, despite the performance of twelve surveillances between November 19, 1992, and February 3, 2004. Failure to maintain the SSF submersible pump in a ready to operate condition was considered to be more than minor, in that, its incorrectly wired motor leads directly affected the cornerstone objective to ensure equipment reliability of a mitigating system (i.e., the SSF). A Phase 3 risk analysis determined that this issue was of very low risk significance. This was based primarily on the availability of an alternate source of water inventory to fill the Unit 2 CCW header (i.e.,via reverse, gravity supplied CCW flow from Lake Keowee through the unit's condensate coolers). (Section 4OA5.8)
Inspection Report# : 2004005*(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)*

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)*

Significance: G Dec 31, 2004
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Inadequate SFP Makeup Procedure Results in the Inadvertent Draining of Approximately 10,000 Gallons of Water from the Unit 3 SFP and the Declaration of a NOUE
A self-revealing non-cited violation of Technical Specification 5.4.1, Procedures, was identified for an inadequate Unit 3 spent fuel pool (SFP) makeup procedure, which resulted in the inadvertent draining of approximately 10,000 gallons of spent fuel pool inventory to the unit's borated water storage tank (BWST) and the declaration of a Notice of Unusual Event (NOUE). The finding was considered to be more than minor, because if left uncorrected, the inadvertent drain down of the SFP could have rendered the SFP cooling pumps inoperable. However, the inadvertent transfer of water from the SFP would have ceased when the suction of the SFP cooling pumps was uncovered, leaving approximately 20 feet of water over the top of the SFP racks to provide sufficient cooling to and shielding of the irradiated fuel assemblies in the Unit 3 SFP. Consequently, the finding was of very low safety significance. (Section 4OA3.1)
Inspection Report# : 2004005*(pdf)*

Emergency Preparedness

Occupational Radiation Safety

Public Radiation Safety

Physical Protection

Physical Protection information not publicly available.

Miscellaneous

Last modified : November 30, 2005