

Oconee 1

3Q/2004 Plant Inspection Findings

Initiating Events

G**Significance:** Sep 25, 2004

Identified By: Self Disclosing

Item Type: NCV NonCited Violation

Failure to Correct the Degraded Condition of the 525 kV Switchyard Load Center Cabinets

A self-revealing non-cited violation of 10 CFR 50 Appendix B, Criterion XVI, was identified for a failure to correct water intrusion problems in the 525kV switchyard load centers, which resulted in the loss of the 525kV and 230kV switchyard auxiliary power supply during a period of heavy rain. The finding was considered to be more than minor because it affected the initiating events cornerstone objective by increasing the likelihood of events that upset plant stability, in that the loss of the auxiliary power to the switchyards would eventually lead to a loss of offsite power or a loss of the safety-related overhead power path from the Keowee hydroelectric units. In addition, the loss of offsite power could lead to a plant trip. In the SDP Phase 1 screening, the finding was determined to be of very low safety significance. Specifically, because the units were in the process of reducing power and would have been shut down before the switchyard batteries were actually depleted, the issue did not increase both the likelihood of a reactor trip and the likelihood that mitigation equipment (Keowee overhead path) would be lost. This finding has cross-cutting aspects related to PI&R (Section 1R12b.(2)).

Inspection Report# : [2004004\(pdf\)](#)**G****Significance:** Mar 27, 2004

Identified By: NRC

Item Type: NCV NonCited Violation

Inadequate Primary Sampling Procedure Results in Excessive RCS Leakage and a NOUE

A self-revealing non-cited violation of Technical Specification 5.4.1, was identified for an inadequate primary sampling procedure which resulted in a waterhammer-generated pressure wave that caused a thermal relief valve to lift. The relief valve stuck open, resulting in a 14 gpm unidentified reactor coolant system (RCS) leak and a Notice of Unusual Event. The licensee's investigation of the event concluded the relief valve stuck open due to foreign material trapped between its seat and disc.

The finding was considered to be more than minor because it affected the initiating event cornerstone, in that, the inadequate primary sampling system procedure increased the likelihood of a small loss of coolant accident (LOCA) occurring. However, the stuck open thermal relief valve was readily isolable by the excessive RCS leakage procedure, consequently, the finding screened out of the SDP Phase 1 analysis as being of very low safety significance.

Inspection Report# : [2004002\(pdf\)](#)**G****Significance:** Mar 27, 2004

Identified By: NRC

Item Type: NCV NonCited Violation

Inadequate Inservice Inspection Program for Inspections of Feedwater System and Steam Line System Supports

A non-cited violation of 10 CFR 50, Appendix B, Criterion X, Inspection, was identified by the inspectors for failure to establish an adequate inspection program for certain feedwater system and main steam system piping supports associated with high energy line break scenarios.

The lack of piping support inspections and resulting inability to identify adverse conditions related to the supports could affect the ability of the feedwater and/or the steam line systems to withstand various events such as seismic induced loading, which in turn could result in damage to other mitigation systems. This issue was considered to be more than minor because if left uncorrected it could prevent the detection of piping support defects which would increase the probability of an initiating event (feed line and steam line rupture). A Phase 1 evaluation was conducted using the initiating event screening criteria. Because the inadequate inspection of the supports had not caused an actual increase the likelihood of an initiating event, the issue was screened out as Green. The determination of no actual increase in the likelihood of an initiating event was based on no significant loading events, such as seismic events, having occurred.

Inspection Report# : [2004002\(pdf\)](#)

Mitigating Systems

G**Significance:** Sep 25, 2004

Identified By: Self Disclosing

Item Type: FIN Finding

Inadequate Maintenance and Oversight of the 5C Lee Combustion Turbine

A self-revealing finding was identified for the inadequate maintenance and oversight of the 5C Lee Combustion Turbine (LCT), which resulted in a condition that caused the turbine to trip off-line while being relied upon as the standby source of emergency power during the Keowee dual unit outage. The finding was considered to be more than minor because it affected the mitigating systems cornerstone objectives for ensuring availability, reliability and capability of systems that are in place to respond to initiating events, in that the 5C LCT was being operated as the standby source of emergency power during the initial Keowee dual unit outage when it failed. The issue was determined to be of very low safety significance based on the Phase 1 SDP screening results that the finding "did not" represent a loss of safety function of a non-Technical Specification train of equipment designated as risk significant, in that the 6C LCT and dedicated power path from Lee Station to Oconee remained operable and in service (Section 1R12b.(1)).

Inspection Report# : [2004004\(pdf\)](#)

G

Significance: Sep 25, 2004

Identified By: Self Disclosing

Item Type: NCV NonCited Violation

Improperly Fabricated Connecting Rod in Keowee Output Breaker ACB-2

A self-revealing non-cited violation of 10 CFR 50 Appendix B, Criterion VIII, was identified for inadequate control of materials, parts, and components associated with an improperly fabricated connecting rod in Keowee hydroelectric unit (KHU) -2 output breaker ACB-2. This resulted in the connecting rod pulling apart and KHU-2 failing to load on July 29, 2004. The finding was considered to be more than minor because it affected the mitigating systems cornerstone objectives for ensuring availability, reliability and capability of systems that are in place to respond to events, in that following the rod failure in ACB-2, the Keowee overhead emergency power path became inoperable. Although the finding represented an actual loss of the safety function of a single train, it was determined to be of very low safety significance because it did not exceed the allowed Technical Specification outage time (Section 4OA2.2b.(2)).

Inspection Report# : [2004004\(pdf\)](#)

W

Significance: Jul 20, 2004

Identified By: NRC

Item Type: VIO Violation

Failure to Meet Licensing Basis for Staffing the SSF in the Event of a Confirmed Fire

A violation was identified for failure to comply with 10 CFR 50, Appendix R, Sections III.L.2.b and III.L.3, in that, for a severe fire in areas requiring the manning of the Standby Shutdown Facility (SSF) and activation of the SSF makeup pump, the licensees' method for implementing their alternative shutdown capability did not ensure that the reactor coolant makeup function would be capable of maintaining the reactor coolant level within the level indication of the pressurizer.

Inspection Report# : [2004013\(pdf\)](#)

G

Significance: Jun 26, 2004

Identified By: NRC

Item Type: NCV NonCited Violation

Inadequate QC Inspections of Reactor Building Emergency Sumps

A Green non-cited violation (NCV) was identified for inadequate quality control (QC) inspections of the Oconee Units 1, 2 and 3 Reactor Building Emergency Sumps (RBES), in that, gaps in the RBES structures were not discovered during previous QC inspections.

The finding was considered to be more than minor because it affected the mitigating systems cornerstone attribute of equipment performance reliability, in that, the inadequate QC inspections failed to identify RBES bypass flowpaths for debris to affect downstream emergency core cooling system components during RBES recirculation. However, because the gaps were small, the increase in the probability of debris bypassing the RBES screens was considered to be low. Consequently, the finding screened out of the Phase 1 SDP analysis as Green (very low safety significance).

Inspection Report# : [2004003\(pdf\)](#)

G

Significance: Jun 26, 2004

Identified By: Self Disclosing

Item Type: NCV NonCited Violation

Failure to Promptly Identify and Correct Seal Water Leakage Contamination of the SSF ASW Pump Inboard Bearing Lube Oil Water Contamination

A self-revealing, non-cited violation of 10 CFR 50 Appendix B, Criterion XVI, Corrective Action, was identified for the failure to correct identified packing leakage on the Standby Shutdown Facility (SSF) Auxiliary Service Water (ASW) pump, which resulted in the contamination of the pump's inboard bearing lubricating oil with water.

The finding was considered to be more than minor because it affected the mitigating systems cornerstone attribute of equipment performance reliability, in that, the failure to correct the pump's packing leakage and resultant water contamination of the inboard bearing lubricating oil

increased the likelihood of the SSF ASW pump failing to complete its mission time. As such, this finding was preliminarily identified as being "Greater Than Green" in NRC Choice Letter dated February 24, 2004 (i.e., Inspection Report 05000269,270,287/2004009). Subsequently, the licensee performed additional bearing testing commensurate with the as found contaminated oil conditions and bearing loading. The test results indicated that the water contamination of the inboard bearing lubricating oil would not likely have adversely affected the function of the SSF ASW pump during its mission time. The issue was subsequently reevaluated under the guidance of the reactor oversight process on June 30, 2004, and determined to be of very low safety significance (Green).

Inspection Report# : [2004003\(pdf\)](#)

G

Significance: Feb 27, 2004

Identified By: NRC

Item Type: NCV NonCited Violation

Inadequate Maintenance of Fire Safe Shutdown Procedures

The inspector and analyst identified an NCV of 10 CFR 50, Appendix R, Section III.L.3 and Technical Specification 5.4.1. During a severe fire in the control room, the procedures implemented for control room evacuation and Safe Shutdown Facility activation were inadequate, in that, operator action to close valve FDW-315, steam generator (S/G) emergency feedwater (EFDW) control valve, was not directed as required to prevent an overcooling event due to spurious actuation of an EFDW pump. The finding is greater than minor because it is associated with procedure quality and degraded the reactor safety mitigating system cornerstone objective. The finding is of very low significance because the fire ignition frequency of the affected cables is low, thereby reducing the likelihood of an EFDW pump start and the need to close valve FDW-315. (Section 4OA5.01)

Inspection Report# : [2004010\(pdf\)](#)

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# : [2004007\(pdf\)](#)

G

Significance: Dec 27, 2003

Identified By: NRC

Item Type: FIN Finding

Failure to implement the Standby Shutdown Facility (SSF) diesel generator manufacturer's recommended preventive maintenance schedule for replacement of grommets

The inspectors identified a finding for failure to implement the Standby Shutdown Facility (SSF) diesel generator manufacturer's recommended preventive maintenance schedule for replacement of grommets every six years. Consequently, at ten years some of the grommets were found to be "at or near failure." This finding is more than minor because a failure of the grommets could lead to diesel coolant leaks and loss of cooling to the diesel; thereby, affecting the reactor safety mitigating system cornerstone objective to ensure the availability, reliability, and capability of a system that responds to initiating events to prevent core damage. A Phase III evaluation, which credits the replenishment of SSF diesel generator cooling and recovery of offsite power, indicated that the performance deficiency was of very low safety significance. (Section 4OA5.4)

Inspection Report# : [2003005\(pdf\)](#)

G**Significance:** Dec 27, 2003

Identified By: NRC

Item Type: NCV NonCited Violation

Failure to Maintain Flood Protection Barriers

A non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion V, was identified by the inspectors for failure to follow instructions in that a flood protection barrier was found improperly removed. The finding was considered to be more than minor because the missing flood barrier affected the mitigating systems cornerstone in that safety related equipment was no longer protected from external factors such as flooding. The Phase 1 screening concluded, that for accident scenarios involving breaks of smaller non-seismic piping in the auxiliary building, the low pressure injection safety function could be adversely affected. Auxiliary building flooding has been previously analyzed in a Phase 3 analysis. This analysis concluded that performance deficiencies related to mitigation of small piping breaks, such as those for which the flood protection barrier was intended to mitigate, would result in a "Green" finding because they would not affect the component cooling system (i.e., Reactor Coolant Pump seal cooling.) (Section 1R06.1)

Inspection Report# : [2003005\(pdf\)](#)**G****Significance:** Dec 27, 2003

Identified By: NRC

Item Type: NCV NonCited Violation

Design Calculation Contains Inaccurate Post LOCA Room Temperatures and a Lack of Assurance that Safety-Related Pumps were Capable of Operating in this Temperature Environment

A NCV of 10 CFR 50 Appendix B, Criterion III, was identified by the inspectors for failure to properly translate design basis parameters for emergency core cooling systems (ECCS) into applicable specifications, drawings, procedures, and instructions. Specifically, design calculation OSC-6667 documented that post LOCA temperatures in the low pressure injection (LPI) and high pressure injection (HPI) pump rooms could reach ambient temperatures as high as 257 degrees; however, safety-related pumps and motors in those rooms (i.e., LPI, HPI, and reactor building spray pumps and motors) were not environmentally qualified for this type of environment. The finding was considered to be more than minor because it potentially affected the mitigating systems cornerstone, in that it affected the environmental qualification of safety-related equipment needed to mitigate a loss of coolant accident. The finding was determined to be of very low safety significance (Green) due to the fact that the re-calculated ambient temperatures were lower than 257 degrees and that actual testing indicated that the pumps and motors could operate successfully at the predicted ambient temperatures without adverse consequences. Therefore, there was no loss of function, and the issue was screened out in Phase 1 of the SDP as Green. (Section 4OA5.5)

Inspection Report# : [2003005\(pdf\)](#)**W****Significance:** Oct 07, 2003

Identified By: NRC

Item Type: VIO Violation

Failure to Promptly Identify and Correct Insufficient SSF Pressurizer Heater Capacity

Contrary to 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, as of March 2002, the licensee failed to promptly identify and correct a condition adverse to quality involving pressurizer ambient heat losses that exceeded the capacity of those pressurizer heaters powered from the Standby Shutdown Facility (SSF). Evidence of this condition, which may have existed from the time the SSF was put into service in 1986 until the condition was discovered in March 2002, included pressurizer insulation problems (since pre-operational testing) and numerous Problem Investigation Process reports since 1996 identifying pressurizer heater capacity concerns. As a result of the failure to promptly identify and correct this condition, an insufficient number of pressurizer heaters powered from the SSF has been available to assure natural circulation during certain postulated SSF events. This issue has a low to moderate safety significance because of the importance of the SSF powered pressurizer heaters to maintain a pressurizer steam bubble during events where the SSF is used to achieve safe shutdown. Specifically, without a steam bubble to maintain primary system pressure, reactor coolant system (RCS) subcooling would be jeopardized, and single phase RCS natural circulation would be interrupted due to voiding in the hot leg. Decay heat would then challenge the pressurizer safety relief valves, and a failure of one of these valves to reseal would lead to core damage since the SSF standby makeup pump is of insufficient capacity to recover the resultant loss in RCS inventory.

Inspection Report# : [2003012\(pdf\)](#)Inspection Report# : [2004011\(pdf\)](#)

Barrier Integrity

G**Significance:** Dec 27, 2003

Identified By: Self Disclosing

Item Type: NCV NonCited Violation

Inadequate maintenance procedure for inspection of RCS FME barrier

A self-revealing NCV of 10 CFR 50 Appendix B, Criterion V, was identified for an inadequate maintenance procedure for inspection of the foreign material exclusion (FME) barrier in the 1B hot leg during steam generator replacement. This allowed the introduction of sealant

material in the reactor coolant system (RCS) piping. The finding was considered to be more than minor because it potentially affected the barrier integrity cornerstone, as foreign material, if left uncorrected, could have an adverse impact on fuel cladding integrity during operation. Additionally, inadequate inspection activities, if not corrected could have adverse consequences on future activities affecting quality. The finding was determined to be of very low safety significance (Green) due to the fact that the foreign material was successfully removed and that the SDP phase one screening for findings that potentially affected the fuel barrier screen as Green. (Section 4OA5.3)
Inspection Report# : [2003005\(pdf\)](#)

Emergency Preparedness

Occupational Radiation Safety

Public Radiation Safety

Significance:  Dec 27, 2003

Identified By: Self Disclosing

Item Type: NCV NonCited Violation

Failure to Comply with 10 CFR 61.56(b)(2) Waste Characteristic Requirements Involving Liquid Content of Waste Shipped to a Licensed Burial Site for Disposal

A self-revealing NCV of 10 CFR 61.56(b)(2) was identified because the licensee transported a cask shipment for disposal at Chem-Nuclear Systems, Barnwell, South Carolina which contained liquid above regulatory limits. This finding is greater than minor because it was associated with the low level burial attribute of the Public Radiation Safety Cornerstone and adversely affected the cornerstone objective to ensure adequate protection of the public health and safety from exposure to radioactive materials released into the public domain. The finding is of very low safety significance because the shipping cask was discovered to have minimal liquid exceeding the regulatory limit of one percent of the waste shipment total volume transported to the burial site for disposal and the liquid was discovered prior to waste disposal. (Section 2PS2b.(1))

Inspection Report# : [2003005\(pdf\)](#)

Significance:  Dec 27, 2003

Identified By: Self Disclosing

Item Type: NCV NonCited Violation

Failure to Comply with 10 CFR 61.55 (a)(2)ii requirements for Classifying Waste Shipped to a Licensed Burial Site for Disposal

A self-revealing NCV of 10 CFR 61.55(a)(2)(ii) was identified because the licensee transported a cask shipment for disposal at Chem-Nuclear Systems, Barnwell, South Carolina with the incorrect waste classification. The cask was originally shipped to Chem-Nuclear Systems, Barnwell, South Carolina, as Class A stable waste and later determined by the licensee to be Class B stable waste. This finding is more than minor because it was associated with the low level burial attribute of the Public Radiation Safety Cornerstone and adversely affected the cornerstone objective to ensure adequate protection of the public health and safety from exposure to radioactive materials released into the public domain. The finding is of very low safety significance because the shipping container was discovered by the licensee to have been under-classified prior to its final disposal and the burial site representatives were properly notified of the classification error. (Section 2PS2b. (2))

Inspection Report# : [2003005\(pdf\)](#)

Physical Protection

[Physical Protection](#) information not publicly available.

Miscellaneous

Last modified : December 29, 2004