Oconee 2
2Q/2004 Plant Inspection Findings

Initiating Events

Significance: G 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

Significance: G 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)*

Significance: G 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)*

Significance: G 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)*

Significance: G 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)*

Significance: G 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)*

Significance: G 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)*

Significance: G Sep 27, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to follow a procedure required by TS 5.4.1, resulting in multiple reactor operators/senior reactor operators failing to properly reactivate their licenses
The inspectors identified a non-cited violation of Technical Specification 5.4.1(a) for failure to follow Operation Management Procedure 1-12, "Maintenance of Licensed Operator, Shift Technical Advisor, and Non-licensed Operator Qualifications," resulting in multiple reactor operators/ senior reactor operators failing to properly reactivate their licenses. This finding is greater than minor because it affected the Mitigating System Cornerstone human performance attribute to ensure that licensed operators are available, reliable, and capable to respond to initiating events to prevent undesirable consequences. The finding was evaluated using the Operator Requalification Human Performance SDP and was determined to be of very low safety significance. Based on more than 20 percent of the reactivated operators failed to meet the requirements as defined in procedure OMP 1-12, the issue was a Green finding. (Section 4OA5.6)
Inspection Report# : 2003004*(pdf)*

Significance: W Aug 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 reseat 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)*

Barrier Integrity

Emergency Preparedness

Occupational Radiation Safety

Public Radiation Safety

Significance: G 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: G 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

Significance: N/A Jul 11, 2003
Identified By: NRC
Item Type: FIN Finding
Problem Identification and Resolution Inspection
The team identified that the licensee was effective at identifying problems and entering them into the corrective action program (CAP) for resolution. 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. The inspector's independent review did not identify significant adverse conditions which were not in the CAP for resolution. Evaluation and prioritization of problems was generally effective; although, one example was noted where an evaluation did not thoroughly examine the potential for generic implications. Corrective actions specified for problems were generally adequate; although, several examples were noted where corrective actions were not complete or not comprehensive. Audits and self-assessments continued to identify issues; however, some examples were noted where the issues were not correctly classified for resolution. Previous non-compliance issues documented as non-cited violations were properly tracked and resolved via the CAP. Personnel at the site felt free to raise safety concerns to management and to resolve issues via the CAP.
Inspection Report# : 2003009*(pdf)*

Last modified : September 08, 2004