Catawba 1
1Q/2006 Plant Inspection Findings

Initiating Events

Mitigating Systems

Significance: G Mar 31, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
Inadequate Risk Assessment and Management Associated With Planned Nuclear Service Water System Maintenance
An NRC-identified non-cited violation was identified for the failure to adequately assess and manage the risk pertaining to a portion of the maintenance activities associated with the removal of the A train of nuclear service water (RN) from service for a planned 14-day outage as required by 10 CFR 50.65(a)(4).

The finding was more than minor because it was associated with the Equipment Performance attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring that the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences is maintained. The inspectors determined that the finding was of very low risk significance (Green), based on the resulting magnitude of the calculated Incremental Core Damage Probability (5.8E-7/day), the length of time that the two A train diesels were unavailable (<18 hours) and that no actual loss of safety function of the 2B DG occurred. This finding involved the cross-cutting aspect of human performance.
Inspection Report# : 2006002*(pdf)*

Significance: G Sep 30, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
Inadequate Control Of Purchased Equipment
The inspectors identified a non-cited violation (NCV) for the failure to assure that purchased equipment conformed to the procurement documents as required by 10 CFR Part 50, Appendix B, Criterion VII. This finding was greater than minor because it affected an objective and attribute of the Reactor Safety, Mitigating Systems Cornerstone to ensure the availability and reliability of systems that respond to initiating events to prevent undesirable consequences. The performance deficiency associated with this finding was the licensee's commercial grade dedication program did not verify manufacturing defects existed on previously dedicated commercial grade relays. The licensee was responsible to acquire the necessary information to assure the procured equipment maintained original design specifications and quality control. The finding was assessed using the SDP for Reactor Inspection Findings for At-Power Situations. The finding was evaluated using the SDP Phase 2 plant notebook and it was determined a Phase 3 evaluation was required, based on the increase in the probability failure rate of the relays which represented an increase in the likelihood of the loss of safety function of the nuclear service water (RN) system and its associated initiating event frequency. The regional SRA performed a Phase 3 SDP for the finding. Electrical schematics were reviewed to determine mode of failures caused by the relays. A time line was constructed to verify the time periods the various relays were in service. Conservative screening values were established for relay failure rates, based on number of demands experienced by the inservice relays. Fault trees were developed to estimate the relay failure impact on the Loss of Service Water initiating event frequency. Using these conservative values, the NRC's plant risk model was run to determine an upper limit for the risk due to the finding. The risk associated with the finding was determined to be GREEN. (Section 1R12)
Inspection Report# : 2005004*(pdf)*

Significance: G Sep 30, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Develop a Complex Lift Plan
The inspectors identified a NCV for the failure to follow the Duke Power Company Lifting Program procedure as required by 10 CFR 50, Appendix B, Section II, Quality Assurance Plan, when the inspectors determined that a complex lift was going to occur over the top of a safety-related structure with no developed or documented lift plan as required by the licensee lifting procedure. The finding is greater than minor because the finding could be viewed as a precursor to a significant event. Without a complex lift plan to ensure quality measures were taken and compensatory actions were considered, had the 23 ton steel structure fallen on the RN lake intake structure, a potential loss of RN may have occurred which would have required prompt action by the operators to transfer the assured water source to the standby nuclear service water pond. Damage to the RN pump structure could have adversely impacted reactor safety and affected the availability and reliability of a

mitigating system performance attribute of the reactor safety cornerstone. The finding was determined to be of very low safety significance, using the significance determination phase 1 worksheet, because the lack of a documented complex lift plan did not result in the loss of safety function of the RN system as the lift was deferred until a plan was developed. This finding involved the cross-cutting aspect of human performance since individuals did not follow or implement the requirements of the Duke Power Company Lifting Program procedure. (Section 1R13)
Inspection Report# : 2005004*(pdf)*

Significance: G Jun 30, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
Inadequate Post Maintenance Testing on 1RN-38B, 1B RN Pump Discharge Valve
The inspectors identified a non-cited violation of Technical Specification (TS) 5.4.1.a, written procedures, because the licensee failed to implement adequate post maintenance testing following maintenance in 1RN-38B, 1B Nuclear Service Water (RN) pump discharge valve, electric valve operator control circuit.

The finding was determined to be greater than minor because 1RN-38B, 1B RN pump discharge valve, was not capable of performing its intended function, which caused the 1B nuclear service water (RN) pump to be inoperable. The inoperability resulted in potential impact on reactor safety and adversely affected the availability and reliability of a mitigating system performance attribute of the reactor safety cornerstone. The finding was determined to be of very low safety significance, using the significance determination phase 1 worksheet, because the inoperability of 1RN-38B and the 1B RN pump did not result in the loss of safety function of the RN train in excess of its TS allowed outage time. This finding involved the cross-cutting aspect of human performance since individuals did not determine adequate post maintenance testing to verify that the valve could perform its intended function following the fuse replacement (Section 1R15b.1).

Inspection Report# : 2005003*(pdf)*

Significance: G Jun 30, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Adequately Evaluate Potential RHR System Differential Pressure During Postulated Accident Conditions In Generic Letter 89-10 MOV Testing Program
A non-cited violation was identified for inadequate design control as required by 10 CFR 50, Appendix B, Criterion III, in that, the licensee found that they had incorrectly assumed that the Unit 1 and Unit 2 containment sump suction valves needed to function under a maximum 20 pound per square inch pressure differential (psid) and then implemented periodic testing under their Generic Letter 89-10 Motor Operated Valve (MOV) testing program to ensure the valves would open against this psid. Subsequent licensee analysis determined that the valves could experience up to 364 psid during specific accident conditions. Because this violation appeared to be of greater significance than the licensee's initial characterization of the issue, this finding is being treated as an NRC-identified violation in accordance with NRC Enforcement Guidance. This finding involved the cross-cutting aspect of human performance since individuals did not determine the proper design parameters and conditions for all required accident scenarios.

This finding was greater than minor because it affected an objective and attribute of the Reactor Safety Mitigating Systems Cornerstone for availability and reliability, in that excessive psid across the containment sump suction valves could prevent the valves from opening and providing a required injection supply source to the emergency core cooling system pumps. The finding was assessed using the significance determination process for Reactor Inspection Findings for At-Power Situations. The evaluation determined that the finding exceeded the threshold that required evaluation under Phase 3 of the significance determination process. The Phase 3 analysis conducted by the Regional Senior Reactor Analyst, determined the finding to be of very low safety significance because the dominant factor in the analysis was that the need for sump recirculation would have to coincide with a degraded grid condition and such an initiating event frequency was sufficiently low enough to conclude the deficiency was Green. (Section 1R15b.2).

Inspection Report# : 2005003*(pdf)*

Barrier Integrity

Emergency Preparedness

Occupational Radiation Safety

Significance: G Sep 30, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to conduct adequate airborne radionuclide surveys for workers making 'at power' lower containment entries
The inspector identified a NCV of 10 CFR 20.1501(a) for failure to conduct adequate airborne radionuclide concentration surveys prior to personnel making Unit 1 (U1) or Unit 2 (U2) lower containment 'at power' entries. Specifically, the licensee failed to assure grab samples collected using the U1 and U2 Containment Air Release and Addition System effluent monitor system (EMF) -38,-39, -40 skid supply line were representative of lower containment airborne conditions. This finding is greater than minor because the failure to conduct adequate surveys of lower containment airborne radionuclide concentrations decreased the effectiveness of radiological controls for workers entering potential airborne radiation areas. The finding was associated with radiation protection program and process attributes of the Occupational Radiation Safety Cornerstone. The finding is of very low safety significance because workers who may have entered lower containment airborne areas were provided with appropriate external radiation monitoring devices, were screened for internally deposited radionuclides upon exiting the radiologically controlled area, and the assigned doses resulting from external radiation sources and from internally deposited radioactive materials were within regulatory limits. This finding has a Problem Identification and Resolution cross-cutting aspect due to the February 2005 evaluation for the ventilation alignment issue not being thorough nor comprehensive. The licensee has entered this finding in its corrective action program as PIP C-05-05169 and was evaluating corrective actions to take (Section 2OS1).
Inspection Report# : 2005004*(pdf)*

Public Radiation Safety

Physical Protection

Physical Protection information not publicly available.

Miscellaneous

Significance: SL-III Jan 24, 2005
Identified By: NRC
Item Type: VIO Violation
Failure to Provide Complete and Accurate Information Involving MOX Amendment Fuel Assemblies and Related Dose Calculations
10 CFR 50.9(a) states, in part, that information provided to the Commission by an applicant for a license or by a licensee shall be complete and accurate in all material respects. Contrary to the above, on February 27, 2003, November 3, 2003, and March 16, 2004, the licensee submitted incomplete and inaccurate information regarding a proposed amendment to the facility operating license, to allow the irradiation of four mixed oxide (MOX) lead test assemblies (LTAs). Specifically:

(1) The proposed license amendment of February 27, 2003, failed to indicate that the reactor core would also include eight next generation fuel LTAs as part of the complete core loading of 193 fuel assemblies. This information was material to the NRC in that, as part of the license amendment review, substantial further inquiry by the NRC was necessary to review the thermal-hydraulic conditions and mechanical design arising from the proposed reactor core composition.

(2) The above submittals included radiation dose evaluations that were not based on the current plant design basis accident radiation doses. This information was material to the NRC, in that as part of the license amendment review, substantial further inquiry by the NRC was necessary to review the radiation doses arising from the proposed reactor core composition.

This is a Severity Level III Violation (Supplement VII) with no civil penalty assessed.

Inspection Report# : 2005006*(pdf)*

Last modified : May 25, 2006