Catawba 1
3Q/2005 Plant Inspection Findings

Initiating Events

Significance: G Mar 31, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
Insufficient Fire Drill Oversight to Ensure Fire Brigade Performance Deficiencies are Identified
The inspectors identified a non-cited violation of Facility Operating Licenses NPF-35 (Unit 1) and NPF-52 (Unit 2), Condition 2.C.5, for the failure to implement the provisions of the approved fire protection program (Branch Technical Position CMEB 9.5-1) set forth in the Updated Final Safety Analysis Report (UFSAR) regarding fire brigade training and drills. Specifically, during the fire drill on February 10, 2005, the drill evaluator did not observe and assess the performance of the three teams attacking the simulated hydrogen fire on the Unit 2 main generator or operators in the main control room. As a result, some fire brigade member performance weaknesses were not noted during the drill, discussed during the post-drill critique or subsequently noted for development of appropriate corrective actions. The licensee recognized the drill team deficiency and implemented a change that required adequate team evaluators for future drills. This finding was determined to be greater than minor because it involved the degradation of a plant fire protection feature and has a credible impact on safety since fire brigade performance deficiencies may prevent a fire from being extinguished or allow a fire to propagate leading to a more significant event. The finding was determined to be of very low safety significance in accordance with Phase 1 of the Fire Protection Significance Determination Process because the fire brigade is only a single element of the defense-in-depth fire protection strategy and the noted deficiencies produced a minimal impact on the fire fighting capabilities of the fire brigade. This finding involved the cross-cutting aspect of Human Performance, since the single evaluator did not identify all of the drill deficiencies that occurred during the drill. (Section 1R05.2)
Inspection Report# : 2005002*(pdf)*

Significance: G Mar 31, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
Inadequate RCS Leakage Detection Instrumentation Surveillance Procedures
The inspectors identified a non-cited violation of Technical Specification 5.4.1.a, Written Procedures, because the licensee failed to establish and maintain an adequate surveillance procedure for containment atmosphere radioactivity monitor surveillance requirement (SR) 3.4.15.2 and SR 3.4.15.4, in that the associated alarm function was not set or tested to alarm at a value equivalent to 1 gallon per minute in one hour for a realistic current reactor coolant activity level. The inspectors identified a non-cited violation of Technical Specification 5.4.1.a, Written Procedures, because the licensee failed to establish and maintain an adequate surveillance procedure for containment atmosphere radioactivity monitor surveillance requirement (SR) 3.4.15.2 and SR 3.4.15.4, in that the associated alarm function was not set or tested to alarm at a value equivalent to 1 gallon per minute in one hour for a realistic current reactor coolant activity level.

The finding was determined to be greater than minor because the containment gaseous and particulate channel radiation monitors were not capable of performing the design bases function for an extended period of time. Additionally, the operability of the reactor coolant system (RCS) leakage detection instrumentation alarming functions was not verified for an extended period of time. The inoperability resulted in potential impact on reactor safety and adversely affected the availability and reliability of the barrier integrity equipment performance attribute of the initiating events cornerstone. The finding was determined to be of very low safety significance because other methods of reactor coolant system leak detection were available to the licensee and no actual leakage above 1gpm was indicated through the RCS water balance surveillance. The unavailability of the gaseous and particulate channel leak detection functions and the RCS leakage detection instrumentation alarm indications did not contribute to an increase in core damage sequences when evaluated using the significance determination phase 1 worksheets. This finding involved the cross-cutting aspect of problem identification and resolution. The licensee had evaluated the operability of the radiation monitors via the corrective action program and incorrectly determined that the radiation monitors were operable. (Section 1R15b.(2))

The finding was determined to be greater than minor because the containment gaseous and particulate channel radiation monitors were not capable of performing the design bases function for an extended period of time. Additionally, the operability of the reactor coolant system (RCS) leakage detection instrumentation alarming functions was not verified for an extended period of time. The inoperability resulted in potential impact on reactor safety and adversely affected the availability and reliability of the barrier integrity equipment performance attribute of the initiating events cornerstone. The finding was determined to be of very low safety significance because other methods of reactor coolant system leak detection were available to the licensee and no actual leakage above 1gpm was indicated through the RCS water balance surveillance. The unavailability of the gaseous and particulate channel leak detection functions and the RCS leakage detection instrumentation alarm indications did not contribute to an increase in core damage sequences when evaluated using the significance determination phase 1 worksheets. This finding involved the cross-cutting aspect of problem identification and resolution. The licensee had evaluated the operability of the radiation monitors via the corrective action program and incorrectly determined that the radiation monitors were operable. (Section

1R15b.(2))
Inspection Report# : 2005002*(pdf)*

Mitigating Systems

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)*

Significance: SL-IV Mar 31, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
Inadequate 10 CFR 50.59 Documentation
The inspectors identified a non-cited violation for making a change to the facility (implemented as a change to the UFSAR in 1995) that involved an Unreviewed Safety Question (USQ), for which no written evaluation provided an adequate bases for the determination that the change did not require a license amendment pursuant to 10 CFR 50.90. Specifically, the UFSAR change reflected an increased length of time for incore instrumentation room sump instrumentation, as well as gaseous and particulate radiation monitors, to detect a 1 gpm leak. This increased the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety evaluation report for the reactor coolant system loss of coolant accident (LOCA) leak rate predictions, because the ability to detect a 1 gpm leak within one hour was relied on and credited in the leak-before-break design analysis. The significance of the violation was evaluated under the 10 CFR 50.59 Rule that was in effect at the time of the change, as well as the current 10 CFR 50.59 Rule. The current 10 CFR50.59 Rule requires, in part that "records must include a written evaluation which provides the bases for the determination that the change does not require a license amendment". This information (i.e., the ability to detect a 1gpm leak within one hour) was relied on in part, by NRC for approval of the leak-before-break analysis. Since, the NRC Enforcement Manual states that violations which existed under the old and new rule should be categorized using the current enforcement guidance, this finding was assessed as a SL IV violation. The significance of this violation was not formally evaluated under the Reactor Oversight Process per the Enforcement Policy, because the Agency views 10 CFR 50.59 issues as potentially impeding the regulatory process (i.e., it precluded NRC review of a change to the facility). The finding was not suitable for evaluation using the SDP. Given that the change to the incore instrumentation room sump instrumentation sensitivity capabilities and the gaseous and particulate radiation monitor sensitivities increased the length of time to detect a 1 gpm leak, and the fact that a diverse means of detecting a 1 gpm leak within one hour existed in accordance with Technical Specification (TS) requirements, the delta core damage frequency for the applicable core damage accident sequences stemming from LOCA initiating events were determined to be of very low safety significance. (Section 1R15b.(1))
Inspection Report# : 2005002*(pdf)*

Significance: G Dec 31, 2004
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Failure to Maintain Two ECCS Trains Operable Due to Gas Accumulation In the Charging Pump Suction Piping
A self-revealing non-cited violation was identified for gas intrusion that resulted in a failure to maintain the 1A and 1B centrifugal charging pumps and 1A safety injection pump in an operable condition, in accordance with Technical Specification 3.5.2, Emergency Core Cooling Systems (ECCS). The licensee had several opportunities to evaluate industry events (some having elements identical to this Catawba gas intrusion event) to address the pressurizer as a gas source and evaluate system integration that could lead to inoperability of ECCS equipment. This finding was greater than minor because it affected an objective and attribute of the Reactor Safety Mitigating Systems Cornerstone, in that gas accumulation in the centrifugal charging pump suction piping rendered ECCS systems unavailable and unreliable. Due to the short exposure time and the assumption that the 1A safety injection pump was only affected during high pressure recirculation, the finding was determined to be of very low safety significance. (Section 4OA3.1)

Inspection Report# : 2004006*(pdf)*

Barrier Integrity

Significance: G Mar 31, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Identify Containment Isolation Valve Inoperability During MOV Thrust Testing
The inspectors identified a non-cited violation for the failure to implement Operations Management Procedure (OMP) 1-8, "Authority and Responsibility of On-Shift Operations Personnel," when licensed operators in the work control center and main control room did not identify that testing performed on a TS containment isolation valve rendered it inoperable and as a result, required actions were not reviewed for implementation. The inspectors determined that this violation was greater than minor because it affected an objective and attribute of the Reactor Safety Barrier Integrity cornerstone associated with the reactor containment integrity in that one of two in-series containment isolation valves was rendered inoperable during planned maintenance activities and not identified by operations personnel so the required TS action statement was not reviewed for implementation. The finding was assessed using the SDP for Reactor Inspection Findings for At-Power Situations. The finding was evaluated using the Phase 1 SDP analysis and determined to be of very low safety significance based on the short length of time the containment isolation valve was de-energized in the non-closed position. This finding involved a human performance cross cutting issue when the licensed operators did not adequately fulfill their duties and responsibilities to recognize and understand plant conditions to implement TS requirements properly. (Section 4OA2.2)
Inspection Report# : 2005002*(pdf)*

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)*

Significance: SL-IV Jan 24, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Update the FSAR Involving Dose Calculations
A violation of 10 CFR 50.71(e) was identified involving DEC's failure to update the FSAR to reflect correct design basis accident dose calculations. Because of its low safety significance and because the issue was entered into their corrective action program (Problem Investigation Process reports G-04-0334 and C-04-4116), the NRC is treating this Severity Level IV violation as a non-cited violation, consistent with Section VI.A of the NRC Enforcement Policy.

Inspection Report# : 2005006*(pdf)*

Last modified : November 30, 2005