Columbia Generating Station

Initiating Events

Significance: G Dec 28, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
Man-lift inappropriately stored in control room
The inspectors identified that the licensee failed to properly store a man-lift, located in the control room, in accordance with plant procedures. The man-lift could have tipped against control room panels containing sensitive plant system control circuits during a seismic event (or other disturbance) resulting in a reactor scram. A violation of Technical Specification 5.4.1.a was identified that is being treated as a noncited violation in accordance with Section VI.A.1 of the NRC Enforcement Policy. The inspectors determined that the issue was greater than minor in significance because it affected the reactor safety, initiating events cornerstone objective. The inspectors utilized the NRC's significance determination process Manual Chapter 0609, Appendix A worksheet and determined that the issue was of very low safety significance (Green). The issue screened out as Green because the problem did not: 1) contribute to the likelihood of a primary or secondary system loss of coolant accident initiator; 2) contribute to both the likelihood of a rector trip and the failure of mitigation equipment; or 3) increase the likelihood of a fire or internal/external flood.
Inspection Report# : 2002004*(pdf)*

Mitigating Systems

Significance: G Sep 21, 2002
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
Failure to Properly Design a Fire Protection Flood Barrier
The licensee did not properly design fire protection systems, including flood barriers, to ensure that water from the systems did not affect safety-related equipment (a self disclosing issue). A spill of 15 to 20 gallons of water on the cable spreading room floor leaked through the floor to safety-related components below. The inspectors also identified that the licensee had missed multiple opportunities to identify and correct the deficiencies earlier. A violation of 10 CFR 50.48a was identified that is being treated as a noncited violation in accordance with Section V1.A.1 of the NRC Enforcement Policy. The inspectors determined that the significance was more than minor because the problem affected the reactor safety cornerstone, mitigating systems objective. Specifically, leakage through the cable spreading room floor following the actuation, rupture or inadvertent operation of the fire protection sprinkler system could adversely impact safety-related switchgear associated with Division I and II systems. The inspectors utilized the NRC's significance determination process (Manual Chapter 0609, SDP Phase 1 Worksheet for IE [initiating event], MS [mitigating system], and B [barrier] Cornerstone, dated March 3, 2002) and determined that the issue was of very low safety significance. The finding was determined to involve a design deficiency confirmed not to result in loss of function per Generic Letter 91-18, Revision 1 (Section 4OA5).
Inspection Report# : 2002003*(pdf)*

Significance: G Mar 26, 2002
Identified By: Licensee
Item Type: NCV NonCited Violation
Standby gas treatment charcoal adsorber deluge valve isolated for an extended period due to a human performance error
Technical Specification 5.4.1.d required, in part, that written procedures for the fire protection program be implemented. Fire Protection Procedure, 15.1.19, "Fire Protection System Flow Path Valve Exercise," Revision 12, required FP-V-72, standby gas charcoal adsorber deluge isolation valve, be locked open. Contrary to the Technical Specification and the fire protection program, this valve was locked in the closed position between January 12 and March 13, 2002, because of human performance error. An operator failed to correctly reposition the valve during a previous surveillance. This issue had more than minor significance because the mispositioned valve resulted in loss of fire suppression capability to one standby gas charcoal absorber. The inspectors determined the issue had very low safety significance (Green) because the charcoal absorber deluge system only provided defense-in-depth fire suppression capability and the standby gas treatment system was not required for postfire plant safe shutdown, as described in FSAR Appendix F, fire protection evaluation. The licensee placed this issue into the corrective action program as Problem Evaluation Request 202-0783.
Inspection Report# : 2001009*(pdf)*

Significance: W Feb 13, 2002
Identified By: NRC
Item Type: VIO Violation
Inadequate design controls over breaker modification and three examples of inadequate corrective actions followed breaker malfunctions
The inspectors identified a finding with an associated violation involving 10 CFR Part 50, Appendix B, Criterion III (Design Control), and 10 CFR Part 50, Appendix B, Criterion XVI (Corrective Actions). The finding involved the degraded performance of 22 breakers that were installed in the plant during May and June 2001. Sixteen of these breakers had active safety functions for Division I and II components. The licensee failed to properly verify the design adequacy of replacement Westinghouse DHP-VR 350 breakers equipped with a SURE CLOSE mechanism to operate in the existing configuration. The design change did not address the substantially reduced closing force available to operate the mechanism operated cell switch between the previous and current designs. Subsequently, the plant experienced four breaker malfunctions involving the Division II standby service water system and emergency diesel generator from June 2001 through February 2002. On June 29, 2001, and November 19, 2001, the Division II standby service water system pump breaker failed, in that the mechanism operated cell switch failed to reposition. In addition, on January 17, 2002, the Division II emergency diesel generator breaker mechanism operated cell switch malfunctioned. For these three malfunctions or failures, the licensee did not identify the cause, the generic aspects of the problem, or take effective actions to prevent repetition. This issue was considered to be more than minor because it impacted the operability of two safety-related systems with the potential to impact others. Using the NRC's Phase 3 Significance Determination Process, this finding was determined to have low to moderate safety significance based on the overall change in core damage frequency and large early release frequency. The licensee captured these problems in Problem Evaluation Request 202-0456 and 202-0927 (Sections 02.03 and 02.04, respectively). The condition that resulted in the breaker failures, the malfunction and the overall degradation of the 22 breakers has been corrected. The U.S. Nuclear Regulatory Commission (NRC) performed a supplemental inspection (IR 50-397/2002006) to assess the licensee's evaluation associated with the failure to meet the requirements of 10 CFR 50, Appendix B, Criterion III (Design Control) and Criterion XVI (Corrective Actions). Specifically, the licensee failed to properly verify the design adequacy of replacement Westinghouse DHP-VR 350 breakers to operate in the existing cubicles, without vendor recommended maintenance, and the licensee failed to specify effective corrective measures in response to initial breaker failures - which permitted even more failures to occur and further delayed correcting the condition for several additional months. The NRC documented this issue in NRC Special Inspection Report 50-397/02-05. The licensee has appropriately identified the root cause and contributing factors that led to the White finding. The licensee's identification of the extent of condition was sufficiently broad to capture all of the known affected breakers and to also pursue potential maintenance related problems with breakers of other designs, that were not directly impacted by this particular problem. The team determined that the licensee's evaluation was thorough and that it provided adequate assurance that all problematic areas were properly addressed.
Inspection Report# : 2002005*(pdf)*
Inspection Report# : 2002006*(pdf)*

Barrier Integrity

Significance: G Dec 10, 2002
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
Recurrence of containment isolation valve failures due to inadequate corrective actions
The inspection team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, for the failure to take effective corrective actions to preclude containment isolation valve failures caused by system debris, a known and preventable problem. The original problem surfaced in 1996, but the licensee failed to follow through on planned corrective measures and two additional valve failures were experienced in the past 18 months. In addition, the licensee did not identify a current operability concern until prompted by the NRC and the licensee's first two attempts at addressing operability were inadequate, in part, because they were based on inaccurate information. In the past 18 months, the licensee experienced two containment isolation valve failures due to system debris - a known and preventable problem. Valves FDR-V-3 and FDR-V-4 are both 3-inch ball valves located in the drywell unidentified leakage rate instrument line. Based on the above, the team determined that the issue was more than minor in significance because the problem affected the reactor safety, barrier integrity cornerstone objective. The team utilized the NRC's significance determination process Manual Chapter 0609, Appendix A worksheet and determined that the issue was of very low safety significance (Green). The issue screened out as Green because the problem did not result in an actual open pathway in the physical integrity of the reactor containment or an actual reduction of the atmospheric pressure control function of the reactor containment.
Inspection Report# : 2002006*(pdf)*

Emergency Preparedness

Occupational Radiation Safety

Significance: G Mar 26, 2002
Identified By: Licensee
Item Type: NCV NonCited Violation
Failure to follow ALARA planning procedures
Technical Specification 5.4.1(a) requires procedures to implement the ALARA program. Procedure SWP-MAI-01, "Work Management-Planning, Scheduling and Work Activities," Revision 10, implements ALARA planning. Section 3.7.1 of the above procedure states, in part, if the job scope is changed, ensure all appropriate ndividuals/ organizations who initially reviewed the work order task review it again and concur with the proposed amendment. On September 17, 2001, the licensee documented a task pertaining to motor operator refurbishment work on Reactor Feedwater Valves MO-65A and B that was worked outside the original work order task ALARA plan. Radiation protection ALARA personnel were not informed of the change to the work order task ALARA plan. The licensee originally estimated the exposure as 348 Person-Millirem, but the actual exposure was 1800 Person-Millirem. The failure to contact ALARA personnel to obtain their concurrence is a Technical Specification 5.4.1(a) violation. This event is described in the licensee's corrective action program, reference PER 201-1983. This violation is being treated as a noncited violation. The safety significance of this violation was determined to be very low (Green) by the Occupational Radiation Safety Significance Determination Process because there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised.
Inspection Report# : 2001009*(pdf)*

Public Radiation Safety

Physical Protection

Significance: N/A Mar 21, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
Security officer announced the arrival and presence of an NRC inspector
On March 21, 2002, the inspectors identified a violation of 10 CFR 50.70, "Inspections," after a security officer announced to other on-duty officers that the NRC resident inspector had arrived on-site. The security officer stated that gatehouse officers have provided central alarm and secondary alarm station security operators advance notification of NRC inspection activities. The officer stated he was unaware of the prohibition restricting the announcement of NRC inspection personnel. The issue was more than minor due to the potential for impacting the NRC's ability to perform its regulatory function. This Severity Level IV violation is being treated as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as Problem Evaluation Request 202-0867 (NCV 50-397/01009-03).
Inspection Report# : 2001009*(pdf)*

Significance: G Jan 23, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
Security Officer Inattentive at the Secondary Alarm Station
On January 23, 2002, the NRC inspector identified an inattentive security officer while on post at the plant secondary alarm station. This condition was a violation of the Facility Operating License, NPF-21, Section 2.E, and CGS physical security plan. The physical security plan required the secondary alarm station be continuously monitored. The officer's inattentiveness resulted in a temporary loss of the security monitoring function. This issue was more than minor because it involved a failure of the licensee to meet the requirements of 10 CFR 73.55(f) and associated licensee security plan. The issue was of very low safety significance (Green) because it did not involve an actual facility intrusion and there have not been greater than two similar findings in the previous four quarters. This violation is in the licensee's corrective action program as Problem Evaluation Request 202-0230 (NCV 50-397/01009-02).
Inspection Report# : 2001009*(pdf)*

Miscellaneous

Significance: N/A Dec 10, 2002
Identified By: NRC
Item Type: FIN Finding
PI&R Inspection
NRC had documented a substantive human performance issue in NRC Inspection Report 50-397/01-04. The issue involved several plant events that were caused by poor human performance. The team reviewed corrective actions associated with that finding, which included: 1) increased contractor training and oversight during outages; 2) increased support to operators during outages; 3) increased resources towards job planning prior to outages; 4) staff coaching sessions; and 5) the use of a human performance simulator. The team also reviewed human performance data and statistics for the past two years, which showed marked improvement for the non-outage period since April 2002. Based on a review of the licensee's records and interviews with plant personnel and managers, the licensee has taken reasonable actions to preclude significant problematic human performance trends. No findings of significance were identified.
Inspection Report# : 2002006*(pdf)*

Last modified : March 25, 2003