Quad Cities 1
3Q/2004 Plant Inspection Findings

Initiating Events

Significance: G Dec 31, 2003
Identified By: NRC
Item Type: FIN Finding
FAILURE TO ENSURE TERMINAL CONNECTIONS TIGHTENED FOLLOWING WORK LEADS TO REACTOR RECIRCULATION RUNBACK
A self-revealing reactor recirculation runback occurred on October 7 due to a loose screw on terminal BB-13 in control room panel 901-18. The screw was likely loosened during modification work conducted in November 2002. The runback and associated control room operator actions resulted in lowering Unit 1 reactor power approximately 70 percent.

This finding was determined to be more than minor because it was a precursor to a significant event (the runback). The inspectors determined that this finding was of very low safety significance because the finding did not contribute to the likelihood of a primary or secondary loss of coolant accident initiator, the likelihood of a reactor trip and that mitigating equipment would not be available, or the increase in the likelihood of a fire or an internal or external flooding event.
Inspection Report# : 2003013*(pdf)*

Mitigating Systems

Significance: G Sep 30, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
INADEQUTE CORRECTIVE ACTIONS FOR JULY 2003 OUT OF TOLERANCE EVENT RESULTS IN REPEAT EVENT IN JULY 2004
The inspectors identified a finding of very low safety significance due to the failure to adequately correct a July 2003 main steam line high flow switch out of tolerance condition. The failure to correct this condition resulted in a July 2004 out of tolerance event on the Unit 1 main steam line high flow switches. Corrective actions included placing the switches on an increased calibration frequency, performing additional drift analysis procedures, and plans to replace the current switches with differential pressure transmitters during upcoming refueling outages.

This finding was considered to be more than minor because if left uncorrected the condition could have led to the setpoint for multiple main steam line high flow switches drifting above the analytical limit. This finding was determined to be of very low safety significance since the out of tolerance switches did not result in a loss of safety function for the containment isolation system. A Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XVI, was identified due to the licensee's failure to adequately address the cause of the July 2003 out of tolerance event. In addition, the corrective actions taken following the July 2003 event failed to preclude a repeat event in July 2004.
Inspection Report# : 2004009*(pdf)*

Significance: G Sep 30, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO PROMPTLY CORRECT DEFICIENCIES ASSOCIATED WITH A DEGRADED RESIDUAL HEAT REMOVAL SERVICE WATER VALVE
The inspectors identified a finding of very low safety significance involving a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action." As of September 17, 2004, the licensee had failed to promptly identify and correct the adverse effects of corrosive water on residual heat removal service water (RHRSW) valves. Specifically, in August 2002, an operating experience report from the Dresden Station described the failure of three RHRSW supply valves due to stem to disk separation because of corrosion. On December 7, 2002, Work Request 76586 was written to repair a potential disk to stem separation in safety-related 1A RHRSW supply to Train B control room heating, ventilation and air conditioning Valve 1-5799-385. However, as of September 17, 2004, the work request had not yet been completed, and the licensee had not examined any other RHRSW valves for corrosion. The licensee entered this issue into its corrective action program. Valve 1-5799-385 was partially repaired and labeled as "emergency use only" on October 6, 2004.

This issue was more than minor because it involved the equipment performance attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent

undesirable consequences. The issue was of very low safety significance since the degraded valve did not result in an loss of safety function for either the residual heat removal service water or the control room emergency ventilation system.
Inspection Report# : 2004009*(pdf)*

Significance: G May 28, 2004
Identified By: NRC
Item Type: FIN Finding
Failure to Provide Adequate Minimum Flow Protection for the RCIC Pump
Green. The inspectors identified a finding of very low safety significance involving inadequate design control of the reactor core isolation cooling system. Specifically, the design of the reactor core isolation cooling system and plant operating procedures did not provide adequate minimum flow protection for the reactor core isolation cooling pump. As a result, the reactor core isolation cooling flow could be reduced below the minimum flow requirements for the pump, potentially resulting in pump damage. This finding applies to both units.

This finding was more than minor since it could have affected the mitigating system cornerstone objective of ensuring the availability of systems required to respond to initiating events. This finding was of low safety significance because it did not represent an actual degradation of the reactor core isolation cooling system. The licensee initiated appropriate corrective actions, including implementing a procedure change and obtaining formal minimum flow information from the pump vendor, to ensure continued operability. No violation of NRC requirements occurred.
Inspection Report# : 2004004*(pdf)*

Significance: G Mar 31, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
AUTOMATIC DEPRESSURIZATION SYSTEM VALVE 1-0203-3B WAS INOPERABLE WHEN REQUIRED TO BE OPERABLE
Technical Specification 3.4.3.A requires that with one relief valve inoperable, restore the valve to operable status within 14 days or be in mode 3 within 12 hours and in mode 4 within 36 hours. In addition, Technical Specification 3.5.1.G requires that with one automatic depressurization system valve inoperable, restore the valve to operable status within 14 days or be in mode 3 within 12 hours and reduce reactor dome pressure to 150 psig or below within 36 hours. Contrary to the above, the licensee discovered on November 15, 2003, that automatic depressurization system valve 1-0203-3B was inoperable when required to be operable from July 23 until November 11, 2003.
Inspection Report# : 2004002*(pdf)*

Significance: G Dec 31, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO DEMONSTRATE PERFORMANCE OR CONDITION OF REACTOR BUILDING FLOOR DRAIN SUMP HIGH LEVEL ALARMS WERE EFFECTIVELY CONTROLLED THROUGH PERFORMANCE OF PREVENTIVE MAINTENANCE
The inspectors identified a Green finding involving a Non-Cited Violation for the failure to demonstrate effective control of the condition of the reactor building floor drain sump high level alarms through the performance of preventive maintenance. As a result, the licensee had not set goals or monitored the performance of the alarms as required by 10 CFR Part 50.65(a)(1).

This finding was determined to be more than minor because if left uncorrected the failure to perform appropriate preventive maintenance would become a more significant safety concern. Due to the nature of this finding, it was unable to be assessed using the Significance Determination Process. However, the details of this finding were reviewed by Region III management, maintenance rule personnel in the Office of Nuclear Reactor Regulation, and Office of Enforcement personnel and determined to be of very low risk significance.
Inspection Report# : 2003013*(pdf)*

Barrier Integrity

Significance: G Sep 30, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
INADEQUATE CHANNEL CHECK PROCEDURE FOR DRYWELL RADIATION MONITORS
A finding of very low safety significance was self-revealed in January 2004 when the Unit 2 drywell radiation monitor failed downscale due to an un-soldered wire connection. The finding was considered a violation of regulatory requirements due to having a channel check procedure which failed to provide appropriate acceptance criteria to determine whether the radiation monitors remained operable. Corrective actions included validating that additional drywell radiation monitors had soldered wire connections where needed, training personnel to verify the proper operation of the drywell radiation monitors, and revising the appropriate procedures with appropriate quantitative and qualitative acceptance criteria.

This finding was more than minor because it was associated with the containment procedure attribute of the barrier integrity cornerstone and impacted the objective of providing reasonable assurance that the physical design barriers protect the public from radionuclide releases caused by accidents and events. The finding was of very low safety significance because it did not contribute to: (1) a degradation of the radiological barrier function provided for the control room, the auxiliary building, the spent fuel pool, or the standby gas treatment system; (2) a degradation of the barrier function of the control room against smoke or a toxic atmosphere; or (3) an actual open pathway in the physical integrity of reactor containment. The finding was determined to be a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion V due to the failure to have a channel check procedure which contained appropriate acceptance criteria.
Inspection Report# : 2004009*(pdf)*

Emergency Preparedness

Occupational Radiation Safety

Public Radiation Safety

Physical Protection

Physical Protection information not publicly available.

Miscellaneous

Last modified : December 29, 2004