Quad Cities 2
4Q/2004 Plant Inspection Findings

Initiating Events

Significance: G Jun 30, 2004
Identified By: NRC
Item Type: FIN Finding
FAILURE TO APPROPRIATELY IMPLEMENT TURBINE THRUST BEARING WEAR DETECTOR CALIBRATION AND SURVEILLANCE TESTING PROCEDURES
A finding of very low safety significance was self-revealed when the Unit 2 main turbine and reactor automatically tripped during thrust bearing wear detector testing. The turbine trip was a result of the licensee's failure to implement the thrust bearing wear detector test program as described in the vendor manual. The inspectors determined that the licensee had modified their test program to gain efficiencies in plant operation, work control, and radiation protection. However, the licensee did not recognize that the increased efficiencies also increased the likelihood of a plant transient during thrust bearing wear detector testing.

This finding was more than minor because it was viewed as a precursor to a significant event (a transient). This finding was of very low safety significance because Unit 2 responded to the turbine trip and reactor trip as designed and all mitigating systems equipment was available following the reactor trip. The finding was not considered a violation of regulatory requirements since the main turbine thrust bearing wear detector was a non-safety related component.
Inspection Report# : 2004005*(pdf)*

Mitigating Systems

Significance: G Dec 31, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE OF UNIT 2 TARGET ROCK VALVE TO MEET TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS OF TECHNICAL SPECIFICATION 3.4.3
The inspectors identified a finding of very low safety significance involving a Non-Cited Violation of Technical Specification 3.4.3 due to the Unit 2 target rock valve being unable to actuate within plus or minus one percent of its nameplate value during as-found testing conducted in April 2004.

This issue was determined to be more than minor because if left uncorrected, this condition could put the licensee at risk for exceeding their vessel overpressure limits following an accident or an anticipated transient without scram. This issue was of very low safety significance because the actuation of the valve at the higher setpoint would not have resulted in exceeding the pressure limits assumed in the licensee's current analyses. Corrective actions for this issue included installing a new valve, performing additional testing to better understand the degradation mechanism, operating the Quad Cities units at pre-extended power uprate power levels, developing a modification to install better materials in the bellows cap area, and continuing the ongoing vibration assessments.

Inspection Report# : 2004010*(pdf)*

Significance: G Dec 31, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
HISTORICAL FAILURE OF MAIN STEAM SAFETY VALVES TO MEET TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS
A finding of very low safety significance and a Non-Cited Violation of Technical Specification 3.4.3 were identified by the inspectors in November 2004 due to the licensee's repeated inability to demonstrate that the main steam safety valves would actuate within plus or minus one percent of the nameplate value when required.

This issue was determined to be more than minor because it led to continued degradation of the main steam safety valves and put the licensee at risk for exceeding their vessel overpressure limits following an accident or an anticipated transient without scram. This finding was of very low safety significance because an adequate number of safety valves and relief valves were available to prevent an overpressure condition from occurring. Corrective actions for this issue included installing new main steam safety valves, submitting a license amendment to change the

main steam safety valve operating tolerances, and revising a previously issued Licensee Event Report to report the previous failures.
Inspection Report# : 2004010*(pdf)*

Significance: G Dec 31, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO PROVIDE ADEQUATE CAPABILITY TO ISOLATE THE SAFETY RELATED TORUS FROM THE NON-SEISMIC PORTIONS OF THE REACTOR CORE ISOLATION COOLING SYSTEM
The inspectors identified a finding and a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," in May 2004 when they discovered that the design of the reactor core isolation cooling system did not provide adequate capability to isolate the safety-related torus from the non-seismic reactor core isolation cooling system under all conditions. As a result, torus water could potentially drain into the reactor building following a seismic event and a failure of the reactor core isolation cooling piping. The loss of torus inventory could potentially affect the safety-related water supply for the emergency core cooling systems.

This finding was more than minor since it could have affected the mitigating cornerstone objective of ensuring the availability of systems required to respond to initiating events. This finding was of low safety significance because a subsequent evaluation demonstrated that the reactor core isolation cooling piping would not have failed during a seismic event. The licensee initiated a procedure change to remotely bypass the valve control logic such that the reactor core isolation cooling system remained operable and the operators could close the valve when required for containment isolation. The licensee also initiated engineering changes to revise the valve control logic as a permanent resolution to the issue.

Inspection Report# : 2004010*(pdf)*

Significance: G Sep 30, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
VIOLATION OF TS 3.3.6.1 DUE TO MAIN STEAM LINE HIGH FLOW SWITCHES BEING FOUND OUT OF TOLERANCE
A finding of very low safety significance was identified when the setpoint for two of the Unit 2 main steam line high flow switches were found to be higher than allowed by Technical Specification 3.3.6.1 in July 2003. As corrective actions, the licensee recalibrated the switches and performed a root cause analysis.

This finding was more than minor because if left uncorrected the switches could have continued to drift to a level above the analytical limit. Had this occurred, the licensee would have been operating in a condition not previously reviewed by the NRC. 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. However, this finding was a Non-Cited Violation of Technical Specification 3.3.6.1 as the out of tolerance switches resulted in the failure to ensure that two trip systems per channel per steam line were operable during Mode 1 operations.
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)*

Barrier Integrity

Significance: G Dec 31, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE OF SAFETY VALVE DISCHARGE LINE FLANGES TO MEET CODE REQUIREMENTS
The inspectors identified a finding of very low safety significance and a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," in August 2004 due to the licensee's failure to adequately translate code design requirements into an operability evaluation for the main steam safety relief valve discharge line flanges.

This issue was more than minor because if left uncorrected the failure to perform adequate operability evaluations could become a more significant safety concern. This issue was of very low safety significance because it did not involve the degradation of a radiological barrier, a barrier used to protect the control room from smoke or toxic gases, and did not result in an actual open pathway in the physical integrity of the reactor containment. As part of the corrective actions for this issue, the licensee implemented compensatory actions to ensure continued operability of the installed flanges and initiated plans to modify the operable but degraded flanges to meet their design requirements.
Inspection Report# : 2004010*(pdf)*

Significance: G Dec 31, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
CONTROL ROOM EMERGENCY VENTILATION TEST FAILURE DUE TO INADEQUATE PROCEDURE AND DEFICIENT MODIFICATION TO HATCH COVERS
A finding of very low safety significance and a Non-Cited Violation of Technical Specification 3.7.4.A were identified by operations personnel in October 2004 due to the licensee's failure to demonstrate that the control room emergency ventilation system was capable of maintaining the control room emergency zone differential pressure at greater than 1/8 of an inch at a flow rate of 2000 standard cubic feet per minute since 1998.

This issue was determined to be more than minor because if left uncorrected, the condition of the control room emergency ventilation system would have continued to degrade without being identified by the licensee. This issue was of very low safety significance since the finding only represented a degradation of the radiological barrier provided for the control room. Corrective actions for this issue including providing additional sealing material to the cable tunnel hatch covers and revising the control room emergency ventilation surveillance procedures to ensure that the Technical Specifications continue to be met.
Inspection Report# : 2004010*(pdf)*

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 : March 09, 2005