Salem 2
2Q/2006 Plant Inspection Findings

Initiating Events

Significance: G Feb 17, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
INADEQUATE PROCEDURE FOR LOSS OF COMPONENT COOLING WATER
The team identified a finding of very low safety significance involving a non-cited violation of Technical Specification 6.8.1, Procedures, for an inadequate procedure to respond to a loss of component cooling water (CCW) event. The procedure was inadequate because it required operators to trip the reactor and immediately enter the emergency operating procedures (EOPs), but relied on an alarm response procedure to accomplish time critical and risk significant actions. The team identified that the execution of the alarm response procedure could be delayed during EOP implementation. As a consequence of relying on a lower tier procedure, the delayed actions significantly decreased margin with respect to reactor coolant pump (RCP) seal temperatures approaching operating limits during this postulated event.

This finding was more than minor because it was similar to Example 3.k in NRC Inspection Manual Chapter (IMC) 0612 Appendix E, Examples of Minor Issues. Specifically, PSEG’s human reliability analysis associated with a loss of CCW event, assumed operators could complete required risk significant, time critical actions in less than one minute, when in fact, the actions could have nominally taken 14 minutes. As a result of this procedure deficiency, there was a significant reduction in the time margin assumed in PSEG's analysis to perform risk significant manual actions (i.e., isolate letdown flow and transfer charging pump suction). This finding affected the Initiating Events Cornerstone objective to limit the likelihood of events that challenge critical safety functions, because it was associated with the cornerstone's attribute for procedure quality. The finding was of very low safety significance because it screened to Green in Phase 1 of the significance determination process (SDP) documented in IMC 0609, Appendix A, Significance Determination of Reactor Inspection Findings for At-Power Situations. Specifically, while the finding directly affected the likelihood of an RCP seal failure because PSEG's previous procedures had little margin for operator error or delay, it appeared that operators could have isolated letdown prior to reaching excessive RCP seal temperatures. Additionally, there was no affect on mitigating systems. A contributing cause of this finding was related to the cross-cutting area of problem identification and resolution.
Inspection Report# : 2006006*(pdf)*

Mitigating Systems

Significance: G Mar 31, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO COMPLY WITH STATION COLD SHUTDOWN REPAIR PROCEDURES
The team identified a non-cited violation (NCV) for failure to maintain equipment required for cold shutdown (CSD) repairs in the designated location. Specifically, procedure SC.MD-AB.ZZ-0001, Installation of Temporary 4KV Power Cables to CCW and RHR Motors, states that “All equipment required to install jumpers, cooling fans and make cable terminations are located in the Salem Safe Shutdown Equipment Storage Area.” Salem Safe Shutdown Equipment Storage Area is located in the Northwest area of the Hope Creek Unit 2 reactor building. An inventory of the designated area in response to inspector inquiries revealed that a significant number of CSD repair materials was found missing. The licensee generated a notification and restocked the missing repair materials.

The finding is more than minor because it is associated with the Mitigating Systems cornerstones attribute objective to ensure the availability of the post-fire cold shutdown system that responds to initiating events to prevent undesirable consequences. Under Manual Chapter 0609 Appendix F, Fire Protection, the finding was evaluated as representing a medium degradation. However, because the equipment involved only effects Cold Shutdown, the finding was determined to be of very low safety significance in accordance with the Fire Protection Significance Determination Process. The performance deficiency had a problem identification and resolution cross-cutting aspect because there was a previous case where cold shutdown repair equipment were found missing and where the corrective actions were ineffective to prevent recurrence.
Inspection Report# : 2006007*(pdf)*

Significance: G Feb 17, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
LACK OF SUPPORTING ANALYSES FOR TURBINE DRIVEN AUXILIARY FEEDWATER OPERATION UNDER STATION BLACKOUT CONDITION
The team identified a finding of very low safety significance involving a non-cited violation of 10 CFR 50, Appendix B, Criterion III, Design

Control. The team determined that analyses did not exist to verify the availability of the auxiliary feedwater (AFW) equipment and capability to operate during temperature conditions which would exist due to a postulated SBO event.

The finding was more than minor because it affected the design control attribute associated with the mitigating systems cornerstone as related to the availability, reliability, and capability of the AFW system. The team reviewed this finding using the Phase 1 SDP worksheet for mitigating systems and determined the finding was of very low safety significance (Green), because it did not represent a loss of system safety function.
Inspection Report# : 2006006*(pdf)*

Significance: G Feb 17, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
INADEQUATE SUPPORTING ANALYSES FOR AUXILIARY FEEDWATER PUMP LOW SUCTION TRIP SETPOINT
The team identified a finding of very low safety significance involving a non-cited violation of 10 CFR 50, Appendix B, Criterion III, Design Control. The technical basis of the AFW pump low suction pressure trip setpoint was not available, and the setpoint appeared to be inadequate to protect the pumps with respect to air entrainment under vortex conditions during a postulated extreme weather event which damages the AFW suction tank. This issue was applicable to all the AFW pumps for both units.

The finding was more than minor because it affected the design control attribute associated with the mitigating systems cornerstone as related to the availability, reliability, and capability of the AFW system. The team reviewed this finding using the Phase 1 SDP worksheet for mitigating systems and determined the finding was of very low safety significance (Green), because it was a design deficiency confirmed not to result in loss of operability. Based on PSEG’s evaluation and credit for operator actions to mitigate the condition, the deficiency would not have resulted in the AFW system becoming inoperable given the failure of the AFW suction tank due to an extreme weather event.
Inspection Report# : 2006006*(pdf)*

Significance: G Feb 17, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
RESIDUAL HEAT REMOVAL ROOM INTERNAL FLOOD PROTECTION
The team identified a finding of very low safety significance involving a non-cited violation of 10 CFR 50, Appendix B, Criterion III, Design Control. The Unit 2 design did not ensure that an internal auxiliary building flood, due to a postulated moderate energy line break, could not affect both residual heat removal (RHR) pump rooms as specified in Updated Final Safety Analysis Report (UFSAR) section 3.6.5.12.5. This issue did not apply to Salem Unit 1.

The finding was more than minor because it affected the mitigating systems cornerstone as related to the availability, reliability, and capability of the RHR system. The team reviewed this finding using the Phase 1 SDP worksheet for mitigating systems and determined the finding was of very low safety significance (Green), because it was a design deficiency confirmed not to result in loss of operability. The performance deficiency had a PI&R cross-cutting aspect.
Inspection Report# : 2006006*(pdf)*

Significance: G Dec 31, 2005
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
22 CONTROL AREA CHILLER INOPERABLE DUE TO INADEQUATE MAINTENANCE PROCEDURE
A self-revealing, non-cited violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified when the 22 control area chiller tripped due to its associated condenser service water outlet valve (22SW102) failing closed. The 22SW102 valve was identified one month earlier as having significant wear conditions during a preventive maintenance activity. The conditions were not corrected and the valve was returned to service without further evaluation. The wear conditions were an indication of the 22SW102 ultimate failure condition.

This finding is more than minor because it is associated with the equipment performance attribute, and it affected the mitigating systems cornerstone objective to ensure the availability and reliability of systems that respond to initiating events to prevent undesirable consequences. The chilled water system is listed as a mitigating system in Table 2 of the Risk Informed Inspection Notebook for Salem Generating Station, Revision 2, and provides support and cooling for the control area ventilation system and the emergency control air compressors. This issue also impacted the initiating events cornerstone because unavailability of one train of a chiller increased the likelihood of loss of control area ventilation and loss of control air events. In accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the inspectors conducted a Phase 1 SDP screening and determined a more detailed Phase 2 evaluation was required to assess the safety significance because the finding affected two cornerstones (initiating events and mitigating systems). Using the Phase 2 SDP analysis, the inspectors determined that the finding was of very low safety significance (Green). The performance deficiency has a problem identification and resolution cross-cutting aspect.
Inspection Report# : 2005005*(pdf)*

Significance: G Sep 30, 2005
Identified By: Self-Revealing

Item Type: FIN Finding

UNAVAILABILITY OF STATION BLACK-OUT AIR COMPRESSOR DUE TO INCOMPLETE PREVENTATIVE MAINTENANCE

A self-revealing finding was identified for failure to implement corrective actions to create a preventive maintenance task to clean lube oil coolers on the station black-out air compressor (SBOAC). As a result, the SBOAC tripped due to a high air outlet temperature condition during a monthly performance test on August 14, 2005. PSEG entered the failure to perform necessary preventive maintenance into their corrective action program for resolution. The finding was not a violation of NRC requirements because it pertained to non-safety related equipment. The cause of the finding is related to the cross-cutting element of problem identification and resolution.

Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC's regulatory function and was not the result of any willful violation of NRC requirements. This finding was more than minor because it was associated with the equipment performance attribute, and it affected the mitigating systems cornerstone objective to ensure the availability of systems that respond to initiating events to prevent undesirable consequences. In accordance with Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the inspectors conducted a Phase 1 Significance Determination Process (SDP) screening and determined that the safety function of the SBOAC, which is risk significant per 10 CFR 50.65, was lost for greater than 24 hours. This required that a Phase 2 SDP analysis be performed. Because the Salem Risk-Informed Inspection Notebook did not consistently describe the SBOAC, the regional Senior Reactor Analyst conducted a Phase 3 SDP analysis and determined the issue to be of very low safety significance.

Inspection Report# : 2005004*(pdf)*

Significance: Sep 30, 2005

Identified By: Self-Revealing

Item Type: NCV NonCited Violation

2A EMERGENCY DIESEL GENERATOR INOPERABLE DUE TO OPERATOR PROCEDURE ERROR

A self-revealing non-cited violation was identified for PSEG's failure to comply with 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." Operators performed surveillance procedure steps out of sequence, inadvertently tripping the 2A emergency diesel generator on undervoltage on August 18, 2005. PSEG entered the failure to implement a surveillance procedure into their corrective action program for resolution. The cause of the finding is related to the cross-cutting element of human performance.

Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC's regulatory function and was not the result of any willful violation of NRC requirements. This finding was more than minor because it was associated with the human performance attribute, and it affected the mitigating systems cornerstone objective to ensure the availability of systems that respond to initiating events to prevent undesirable consequences. In accordance with Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the inspectors conducted a Phase 1 Significance Determination Process screening and determined the issue to be of very low safety significance. The finding was not a design or qualification deficiency, did not represent a loss of system safety function, did not represent an actual loss of safety function of a single train for greater than its Technical Specification allowed outage time, did not represent an actual loss of safety function of one or more non-Technical Specification trains of equipment designated as risk significant per 10 CFR 50.65 for greater than 24 hours, and did not screen as potentially risk significant due to external events.

Inspection Report# : 2005004*(pdf)*

Barrier Integrity

Significance: Dec 31, 2005

Identified By: Self-Revealing

Item Type: NCV NonCited Violation

POOR MAINTENANCE RESULTS IN UNAVAILABILITY OF 25 CONTAINMENT FAN COIL UNIT

A self-revealing, non-cited violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified when the 25 containment fan coil unit (CFCU) malfunctioned. The malfunction was a result of previous inadequately performed maintenance. Maintenance technicians did not follow work instructions and incorrectly installed an air booster relay diaphragm to an associated air-operated valve, which resulted in equipment unavailability.

The finding is more than minor because it affected the human performance attribute of the barrier integrity cornerstone objective to provide reasonable assurance that containment barriers protect the public from radionuclide releases caused by accidents or events. In accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the inspectors were directed to IMC 0609, Appendix H, "Containment Integrity Significance Determination Process," because the finding represented an actual loss of defense-in-depth of a system that controls containment pressure. The finding was determined to be of very low safety significance (Green) because the Salem Units include a large, dry containment and containment fan coil unit failures do not significantly contribute to large early release frequency. The performance deficiency has a human performance cross-cutting aspect.

Inspection Report# : 2005005*(pdf)*

Significance: Dec 31, 2005

Identified By: NRC
Item Type: NCV NonCited Violation
INADEQUATE CONTAINMENT CLOSURE PROCEDURE REQUIREMENTS
The inspectors identified a non-cited violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for deficient containment closure controls during the Spring 2005 Unit 1 refueling outage. PSEG did not ensure that one of the containment equipment hatches could be closed, either inside or outside of containment, for a postulated event involving core boiling or fission product release. Installation of either hatch required a heavy lift crane. The inside crane would be affected by high temperatures and high humidity on a loss of decay heat removal with the reactor coolant system vented, and the outside crane was unavailable for several hours during high wind conditions.

The finding is more than minor because it affected the procedure quality attribute of the barrier integrity cornerstone objective to provide reasonable assurance that containment barriers protect the public from radionuclide releases caused by accidents or events. Based upon the finding representing a potential open pathway in the physical integrity of reactor containment while the unit was shutdown, IMC 0609, Appendix H, "Containment Integrity Significance Determination Process," was used to determine the significance of the finding. Appendix H, Table 6.3 was used for the Phase 1 screen. Based upon Salem Unit 2 being a pressurized water reactor with a large, dry containment and the finding impacting an intact containment penetration, the finding required a Phase 2 analysis. The Phase 2 risk approximation determined the finding to be of low to moderate safety significance. Consistent with IMC 0609 guidance, a Senior Reactor Analyst performed a Phase 3 risk assessment to more accurately identify the risk significance and determined the issue to be of very low safety significance (Green).
Inspection Report# : 2005005*(pdf)*

Emergency Preparedness

Occupational Radiation Safety

Public Radiation Safety

Physical Protection

Physical Protection information not publicly available.

Miscellaneous

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