Salem 1
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
DEGRADED COMPONENT COOLING WATER VALVE IMPACT ON COMPONENT COOLING WATER HYDRAULIC ANALYSES
The team identified a finding of very low safety significance (Green) involving a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI,

Corrective Action. Specifically, the corrective actions for a degraded condition that impacted the existing design analysis for component cooling water flowrates to safety-related components under certain accident scenarios was inadequate. PSEG had failed to identify and evaluate the impact of a 700 gpm leak-by through a spent fuel pool heat exchanger valve which invalidated existing component cooling hydraulic model design analysis assumptions.

The finding was more than minor because the condition affected the design control performance attribute of the mitigating system cornerstone objective to ensure the capability of systems that respond to initiating events. 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 there was no loss of system safety function.
Inspection Report# : 2006006*(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: Self-Revealing
Item Type: FIN Finding
FAILURE OF NO. 11 SWITCHGEAR EXHAUST FAN DUE TO LACK OF PREVENTIVE MAINTENANCE PERFORMANCE
A self-revealing finding of very low safety significance (Green) was identified which was associated with the failure of the No. 11 Switchgear return exhaust fan breaker to close and start the fan in October 2005. Specifically, the failure occurred due to latch binding, caused by grease hardening, which was a result of the inadequate implementation of a station procedure which required the performance of preventive maintenance tasks on or before their suggested due dates.

The finding was more than minor because the failure of the 11 switchgear return exhaust fan breaker affected the Mitigating Systems cornerstone objective of ensuring the availability, reliability and capability 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. This screening determined that a Phase 2 evaluation was required, because the finding represented an actual loss of safety function of one non-Technical Specification Train of equipment designated as risk-significant per 10 CFR 50.65, for greater than 24 hours.

A Phase 3 evaluation was performed instead of a Phase 2 evaluation because the risk informed notebook directed performance of a Phase 3 evaluation for issues involving the Switchgear ventilation system. A Phase 3 Risk Assessment determined this finding to be of very low safety significance (Green). While a performance deficiency was identified with regard to the failure to properly implement a station procedure for a preventive maintenance task, there were no violations identified during the review of this issue.
Inspection Report# : 2006006*(pdf)*

Significance: G Dec 31, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
FREON LEAKS ON '11' CONTROL AREA CHILLER
The team identified a non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," because PSEG failed to properly evaluate and correct freon leaks on control area chillers. This condition resulted in a trip and unplanned unavailability of the '11' control area chiller. PSEG entered this issue into the corrective action program.

The finding was more than minor because it affected the equipment performance attribute of the Mitigating Systems cornerstone in that it reduced the availability of systems that respond to initiating events to prevent undesirable consequences. This issue also impacted the Initiating Events cornerstone because unavailability of one chiller train 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 2 SDP evaluation and determined the issue to be of very low safety significance (Green). The performance deficiency had a problem identification and resolution cross-cutting aspect, in that previous evaluations were narrow in scope and did not include periodic monitoring of freon inventory to preclude repeat trips.
Inspection Report# : 2005012*(pdf)*

Significance: G Dec 31, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE OF 12SW39 RENDERS 1B EMERGENCY DIESEL GENERATOR UNAVAILABLE
The team identified an NCV of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," for failure to properly evaluate and correct a known degraded condition on 12SW39, the service water stop valve to the 1B emergency diesel generator (EDG) jacket water and lube oil coolers. PSEG documented degraded operation of the 12SW39 valve in October 2004, when PSEG evaluated a similar failure of the 23SW39 valve to pass its surveillance stroke time test. On September 19, 2005, the 12SW39 valve failed to open causing the 1B EDG to be unavailable until operators opened the valve. PSEG entered this issue into the corrective action program.

The finding was more than minor because it affected the equipment performance attribute of the Mitigating Systems cornerstone objective to ensure the availability of systems that respond to initiating events to prevent undesirable consequences. Because the valve was demonstrated operable on September 18, 2005, the exposure time for the failure of 12SW39 was less than one day. 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 the issue to be of very low safety significance (Green). 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, and did not screen as potentially risk significant due to external events. The performance deficiency had a problem identification and resolution cross-cutting aspect, in that evaluation of the 12SW39 valve was incomplete and did not provide for adequate corrective actions.
Inspection Report# : 2005012*(pdf)*

Significance: G Dec 31, 2005
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
11 SAFETY INJECTION PUMP INOPERABLE DUE TO OPERATOR PROCEDURE ERROR
A self-revealing, non-cited violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified when the 11 safety injection pump discharge valve was discovered closed prior to a routine inservice pump test. The discharge valve was left closed five days earlier at the conclusion of a refueling outage surveillance test due to procedure implementation errors and inadequate operator fundamental standards.

This finding is more than minor because it is 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 IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the inspectors conducted a Phase 1 SDP screening and determined that a Phase 2 SDP analysis was required because the finding represented an actual loss of safety function of a single train for greater than its Technical Specification (TS) allowed outage time. Using the Phase 2 SDP analysis, the inspectors determined that the risk significance of the finding based on internal initiating events that lead to core damage could have been of substantial safety significance. The inspectors referred the results to a senior reactor analyst (SRA) for further review and a more detailed Phase 3 SDP analysis. The SRA completed a Phase 3 analysis of the finding and determined the issue was of very low safety significance (Green). The Salem Standardized Plant Analysis Risk model Revision 3.21, indicated that the finding increased the chance of core damage, over the 132 hour exposure time, on the order of 1 in 200,000,000 or mid E-9. The performance deficiency has a human performance cross-cutting aspect.
Inspection Report# : 2005005*(pdf)*

Significance: G Dec 31, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
INADEQUATE RISK ASSESSMENT
The inspectors identified a non-cited violation of 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power

Plants," for a failure to incorporate an unavailable pressurizer power operated relief valve (1PR2) into a risk assessment during emergent maintenance activities. Operators inappropriately assumed that the weekly risk assessments were calculated such that equipment out of service for any portion of the week was calculated out of service for the entire week. The probabilistic risk assessment group changed its practice to only measure risk for actual scheduled maintenance durations.

This finding is more than minor because PSEG failed to adequately consider the unavailability of 1PR2, a risk significant SSC (included in Table 2 of the Salem Phase 2 SDP risk-informed notebook). The finding was evaluated in accordance with Appendix K of Inspection Manual Chapter 0609, "Maintenance Risk Assessment and Risk Management Significance Determination Process," and is determined to be of very low safety significance (Green). This determination is based on PSEG's incremental core damage probability calculated to be 1.7E-9 for the 3.2 hours that 1PR2 was out of service. The performance deficiency has a human performance cross-cutting aspect.
Inspection Report# : 2005005*(pdf)*

Significance: G Dec 31, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
EMERGENCY CORE COOLING SYSTEMS CONTAINMENT SUMP DEFICIENCIES
The inspectors identified a non-cited violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," because operators did not properly follow a surveillance test procedure for inspecting the emergency core cooling systems (ECCS) containment sump during a closeout inspection. Operators did not identify and document gaps in the sump screen during the inspection, as specified in the procedure.

This finding is more than minor because it affected the design control attribute of the mitigating systems cornerstone objective to ensure the reliability of systems that respond to initiating events to prevent undesirable consequences. 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 the issue to be of very low safety significance (Green). The inspectors reviewed a PSEG engineering evaluation for past operability, and concluded that potentially affected ECCS components and the containment spray system were likely capable of performing their intended safety functions. 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. The performance deficiency has a human performance cross-cutting aspect, because a design change package did not close some gaps and operators did not identify other sump gaps.
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)*

Barrier Integrity

Significance: G Mar 31, 2006
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
INADEQUATE MAINTENANCE PRACTICES RESULT IN UNAVAILABILITY OF THE 11 CONTAINMENT FAN COIL UNIT
A self-revealing, non-cited violation of 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” was identified. PSEG maintenance personnel omitted procedure steps to obtain motor vibration data at the conclusion of 11 containment fan coil unit (CFCU) preventive

maintenance, and the 11 CFCU motor outboard bearing subsequently failed. PSEG initiated actions to correct this post-maintenance testing problem.

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. The 11 CFCU was unavailable for about 92.5 hours. 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 had a human performance cross-cutting aspect.
Inspection Report# : 2006002*(pdf)*

Significance: G Dec 31, 2005
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
INADEQUATE MAINTENANCE PRACTICES RESULTED IN UNAVAILABILITY OF 13 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 13 containment fan coil unit (CFCU) malfunctioned. The malfunction was a result of previous inadequately performed maintenance. Maintenance technicians and operations and engineering personnel did not perform comprehensive troubleshooting efforts for an associated service water flow control valve, resulting in repeat malfunctions and extended unavailability of the 13 CFCU.

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)*

Emergency Preparedness

Occupational Radiation Safety

Significance: G Dec 31, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO SURVEY THE RESIDUAL HEAT REMOVAL PIT
The inspectors identified a non-cited violation of 10 CFR 20.1501, "Surveys and Monitoring," for deficient radiological area access control. An NRC inspector was exposed to unanticipated radiation levels of approximately 72 millirem per hour (mrem/hr) because PSEG radiation protection technicians were not directed to survey a residual heat removal (RHR) room after control room operators established the RHR system in a shutdown cooling lineup. Radiation levels in the area were as high as 150 mrem/hr.

The finding is more than minor because it is associated with the program and process attribute of the occupational radiation safety cornerstone and affected the objective to ensure the adequate protection of the worker health and safety from exposure to radiation from radioactive material during routine civilian nuclear reactor operation. Since this occurrence involved workers' unplanned, unintended dose or potential for such a dose that could have been significantly greater as a result of a single minor, reasonable alteration of circumstance, this finding was evaluated in accordance with IMC 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process." The inspectors determined that the finding was of very low safety significance (Green), because it did not involve (1) ALARA planning and controls, (2) an overexposure, (3) a substantial potential for an overexposure, or (4) an impaired ability to assess dose. The performance deficiency has a human performance cross-cutting aspect.
Inspection Report# : 2005005*(pdf)*

Public Radiation Safety

Physical Protection

Physical Protection information not publicly available.

Miscellaneous

Last modified : August 25, 2006