Salem 2
4Q/2006 Plant Inspection Findings

Initiating Events

Significance: G Dec 31, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO INSTITUTE EFFECTIVE CORRECTIVE ACTIONS FOR REACTOR COOLANT SYSTEM TUBING LEAKS
The inspectors identified a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Actions," in that corrective actions established in 1998 to identify, clean, and inspect Unit 2 reactor coolant system (RCS) instrument tubing were not implemented. Because these corrective actions were not implemented, three through-wall cracks were identified in RCS instrument tubing in October 2006.

This finding is more than minor because it was associated with the equipment performance attribute of the Initiating Events cornerstone and affected the objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shut down as well as power operations. The inspectors determined that the finding was of very low safety significance (Green) using a Phase 1 screening in Appendix A of Inspection Manual Chapter 0609, "Determining the Significance of Reactor Inspection Findings for At-Power Situations." Assuming worst case degradation, the finding would not result in exceeding the Technical Specification limit for identified RCS leakage and would not have likely affected other mitigation systems resulting in a total loss of their safety function. The finding has a cross-cutting aspect in area of problem identification and resolution, because PSEG did not take appropriate corrective actions, in 1998 and 2005, to address these safety issues in a timely manner, commensurate with their safety significance and complexity.
Inspection Report# : 2006005 *(pdf)*

Significance: G Sep 30, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
22 SERVICE WATER STRAINER TRIP
The inspectors identified a non-cited violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for failure to accomplish maintenance in accordance with procedures. PSEG maintenance personnel omitted procedure steps to adequately tighten or properly lock a locknut on the 22 service water strainer during preventive maintenance. Consequently, the 22 service water strainer motor tripped due to increased strainer basket internal interference after it was returned to service.

The finding is more than minor because it is associated with the equipment performance attribute of the Initiating Events cornerstone, and it affected the cornerstone objective. Unavailability of the 22 SWS and SWP increased the likelihood of a loss of service water. This finding also impacted the Mitigating Systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, incorrectly performed maintenance degraded both availability and reliability of the 22 SWS and SWP. 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 more detailed Phase 2 evaluation was required to assess the safety significance because the performance deficiency affected two cornerstones. However, the Risk-Informed Inspection Notebook for Salem Nuclear Generating Station does not evaluate loss of service water initiating events. Therefore, an NRC Region 1 Senior Reactor Analyst (SRA) conducted a Phase 3 analysis and determined the finding was of very low safety significance (Green). The performance deficiency has a cross-cutting aspect in the area of human performance related to the work practices component, because PSEG did not effectively communicate expectations regarding procedure compliance and personnel did not follow procedures.
Inspection Report# : 2006004 *(pdf)*

Significance: G Sep 30, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
REACTOR COOLANT SYSTEM TUBING STRUCTURAL INTEGRITY
The inspectors identified a non-cited violation for PSEG’s failure to follow Salem Technical Specification 3.4.11.1.b., Structural Integrity. PSEG discovered a leak on the instrument tubing for reactor coolant system loop flow transmitter 2FT416 and did not properly classify and evaluate the leak for operability or structural integrity, or alternatively isolate the affected tubing.

The finding is more than minor because it affects the Initiating Events cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown and at power. The inspectors determined that the finding was of very low safety significance (Green) using a Phase 1 screening in Appendix A of Inspection Manual Chapter 0609, “Determining the Significance of Reactor Inspection Findings for At-Power Situations.” It is expected that a tubing crack would result in an increase in reactor coolant system (RCS) leakage, and operators would take action prior to exceeding Technical Specification limits for RCS leakage. Therefore, assuming worst case degradation, the finding would not result in exceeding the Technical Specification limit for identified RCS leakage and would not have likely affected other mitigation systems resulting in a total loss of their safety function. The performance deficiency has a cross-cutting aspect in the area of problem identification and resolution, related to the corrective action program component, because PSEG did not thoroughly evaluate the condition.
Inspection Report# : 2006004 *(pdf)*

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 Dec 31, 2006

Identified By: NRC
Item Type: NCV NonCited Violation
INADEQUATE PROCEDURE IMPLEMENTATION FOR SCAFFOLD CONSTRUCTION
The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” because PSEG did not adequately implement procedural controls for scaffold construction in safety-related areas. This performance deficiency had the potential to adversely impact the upper bearing cooling supply to five of the six Unit 2 service water (SW) pumps and three of the six Unit 1 SW pumps. Once identified, PSEG corrected the scaffold deficiencies.

The issue screened as more than minor based on NRC Inspection Manual Chapter (IMC) 0612, Appendix E, “Examples of Minor Issues and Cross-Cutting Aspects,” Example 4.a, because the inspectors identified multiple examples where there was not an engineering seismic impact evaluation to demonstrate no adverse effect on safety-related SW equipment. The finding was determined to be of very low safety significance (Green) because the performance deficiency was not a design deficiency or qualification deficiency; did not represent an actual loss of safety function of a system; did not represent an actual loss of safety function of a single train for greater than the Technical Specification allowed outage time; did not represent an actual loss of safety function of one or more non-Technical Specification trains of equipment; and did not screen as potentially risk significant due to seismic, flooding or a severe weather initiating event. This finding has a cross-cutting aspect in the area of human performance because PSEG personnel did not follow procedures.
Inspection Report# : 2006005 *(pdf)*

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)*

Barrier Integrity

Significance: G Dec 31, 2006
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
INCORRECTLY POSITIONED FUEL ASSEMBLY
A self-revealing non-cited violation of Salem Technical Specification 6.8.1.b, "Procedures and Programs" was identified when PSEG discovered that an irradiated fuel assembly was incorrectly positioned into the spent fuel pool (SFP) and subsequently transferred without authorization during the reactor core offload of Salem Unit 2's fifteenth refueling outage. Contrary to procedural requirements, PSEG did not ensure that the SFP crane operator used a working copy of the applicable transfer sheets, fuel handling technicians did not properly document a fuel movement irregularity and then transferred a fuel assembly within the SFP without fully apprising the fuel handling senior reactor operator (SRO) or reactor engineer (RE) of the circumstances and, finally, PSEG did not ensure that spent fuel manipulations in the SFP were supervised by a qualified SRO or RE.

This finding is more than minor because it affected the configuration control attribute of the barrier integrity cornerstone. Specifically, mispositioned fuel in the SFP increases the likelihood of an unanalyzed condition in the SFP and a potential impact on the fuel cladding barrier. An increased likelihood of an unanalyzed condition existed because SFP activities were conducted such that more than one fuel assembly could have been incorrectly positioned. This finding was evaluated by the significance determination process of Inspection Manual Chapter (IMC) 0609, Appendix M, “Significance Determination Process Using Qualitative Criteria” because neither IMC 0609, Appendix A, “Determining the Significance of Reactor Inspection Findings for At-Power Situations”; nor IMC 0609, Appendix G, “Shutdown Operations Significance Determination Process,” apply to the spent fuel pool. NRC management determined the finding was of very low safety significance because the deficiency did not cause actual degradation of plant systems, structures or components. Specifically, PSEG analysis demonstrated that the incorrectly positioned fuel assembly was in an acceptably safe location for each move. This finding has a cross-cutting aspect in the area of human performance because PSEG did not ensure supervisory and management oversight of work activities, including contractors, such that nuclear safety was supported.
Inspection Report# : 2006005 *(pdf)*

Emergency Preparedness

Occupational Radiation Safety

Public Radiation Safety

Physical Protection

Physical Protection information not publicly available.

Miscellaneous

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