La Salle 2
4Q/2007 Plant Inspection Findings

Initiating Events

Significance: G Dec 14, 2007
Identified By: NRC
Item Type: FIN Finding
Failure to perform root cause for significant condition adverse to quality
The NRC identified a Green NCV of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," for the failure to perform an adequate RCA to determine the corrective actions necessary to prevent recurrence for a SCAQ. Specifically, the licensee did not evaluate whether there were any aspects under their control that may have identified or prevented the incorrect machining of the Unit 1 jet pump riser brace clamps. The modification was initiated and processed in accordance with the licensee's process, but the contractor had the primary responsibility for implementation. The licensee assigned the performance of the RCA to the contractor. The contractor identified that they had provided incorrect measurements. However, the licensee did not perform an evaluation of their involvement with the modification; specifically, they did not look at those aspects of the modification directly under their control. By not performing an independent evaluation, the licensee failed to identify the root cause of any weaknesses within their oversight of the work that may have identified the incorrect measurements. As such, they were not able to determine a corrective action to prevent recurrence of similar oversight of contractor activities. The performance deficiency has a cross-cutting aspect in the area of Problem Identification and Resolution, Corrective Action Program, because the licensee did not evaluate whether there were any aspects under their control that may have identified or prevented the incorrect machining of the clamps. [P.1(c)]

Inspection Report# : 2007006 *(pdf)*

Significance: G Mar 31, 2007
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Failure to Use Valve Alignment Checklist When Clearing Tag Out Results in Mispositioned Valve and Low Instrument Nitrogen System Header Pressure.
A self-revealing finding of very low safety significance was identified following the removal of a safety tag out and valve realignment for the 2A instrument nitrogen (IN) compressor. Specifically, operations personnel were restoring the system valve lineup following maintenance and placed one valve, 2IN073, into the closed position when it should have been left open, which resulted in an unplanned loss of IN system header pressure. A non-cited violation of Technical Specification 5.4.1.a was also identified for failure to follow the required steps for component restoration following the removal of a safety tag out as outlined in the licensee's procedures. The performance deficiency associated with this finding was the failure on the part of plant operators to follow the provisions of their procedure for equipment clearance orders and safety tagging. The finding was determined to be of more than minor significance in that it had a direct impact on the objective for the Initiating Events Cornerstone for Reactor Safety. Specifically, the inspectors determined that the licensee's failure to properly realign the Unit 2 IN system following maintenance created an unnecessary challenge to control room personnel, who were forced to use an abnormal operating procedure to maintain Unit 2 IN system header pressure to avoid unplanned and unintended valve actuations. Because the finding did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available, the inspectors concluded that the finding was of very low safety significance and within the licensee's response band. In addition, the inspectors also determined that the finding was related primarily to the cross-cutting area of Human Performance since personnel work practices did not support human performance in that the licensee failed to define and effectively communicate expectations regarding procedural compliance and personnel did not follow procedures. Corrective actions planned and completed by the licensee included coaching and counseling of the operators involved and a next shift communication message to all operators on the incident and preliminary cause.
Inspection Report# : 2007002 *(pdf)*

Significance: G Mar 31, 2007
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Failure to Adequately Plan and Proceduralize Reactor Vessel Nozzle Flushing Activities Results in Inadvertent ECCS Injection into the Reactor Vessel.
A self-revealing finding of very low safety significance was identified following the inadvertent initiation of the Division 1 emergency core cooling system (ECCS) on Unit 2 during reactor vessel nozzle flushing from the refuel floor for radiation dose reduction. Specifically, licensee work planning personnel did not recognize the potential adverse impact on ECCS instrumentation taps from using a high-pressure flushing wand to clean out reactor vessel nozzles, and failed to provide personnel performing the flushing activities with adequate procedural instructions. A non-cited violation of 10 CFR 50, Appendix B, Criterion V, was also identified for the failure to adequately prescribe documented instructions or procedures for the work activity that were appropriate to the circumstances. The performance deficiency associated with this finding involved inadequate work planning and written instructions for the reactor vessel nozzle flushing activities. The finding was determined to be of more than minor significance in that it had a direct impact on the objective for the Initiating Events Cornerstone for Reactor Safety. Because the finding involved adequate mitigation capability and was not an event that could be characterized as a loss of control, the inspectors concluded that it was of very low safety significance and within the licensee's response band. In addition, the inspectors determined that the finding was related primarily to the cross-cutting area of Human Performance since the licensee did not appropriately plan work activities consistent with nuclear safety and failed to incorporate risk insights in accordance with the work activity being performed. Corrective actions planned and completed by the licensee included halting all reactor vessel nozzle flushing operations until an initial investigation into the event was performed and conducting a full root cause analysis for the event.
Inspection Report# : 2007002 *(pdf)*

Significance: G Mar 31, 2007
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Inadequate Procedural Instructions to Place Shutdown Cooling in Service Results in Inadvertent ECCS Injection into the Reactor Vessel.
A self-revealing finding of very low safety significance was identified following the inadvertent initiation of Unit 2 Division 2 ECCS, which occurred when operators started shutdown cooling (SDC) while reactor coolant system was pressurized. Specifically, adequate procedural instructions were not provided and as such, control room personnel did not recognize the potential consequences associated with initiating SDC with a pressurized reactor coolant system. A non-cited violation of 10 CFR 50, Appendix B, Criterion V, was also identified for the failure to adequately prescribe documented instructions or procedures for the work activity that were appropriate to the circumstances. The performance deficiency associated with this finding involved Unit 2 control room personnel not properly or thoroughly reviewing actions associated with starting SDC with the reactor vessel water system solid and pressurized prior to their performance. The finding was determined to be of more than minor significance in that it had a direct impact on the objective for the Initiating Events Cornerstone for Reactor Safety. Because the finding involved adequate mitigation capability and was not an event that could be characterized as a loss of control, the inspectors concluded that it was of very low safety significance and within the licensee's response band. In addition, the inspectors determined that the finding was related primarily to the cross-cutting area of Human Performance since the control room personnel did not use conservative assumptions in decision-making and as such, did not identify the possible unintended consequences of their actions. Corrective actions planned and completed by the licensee included performing an initial investigation into the event, performing an engineering analysis of system impact and conducting a full root cause analysis for the event.
Inspection Report# : 2007002 *(pdf)*

Mitigating Systems

Significance: G Dec 14, 2007
Identified By: NRC

Item Type: FIN Finding
Failure to correct a signifcant condition adverse to quality
The NRC identified a Green NCV of 10 CFR 50, Appendix B, Criterion XVI, “Corrective Action,” for the failure to correct a SCAQ in a timely manner. Specifically, the licensee had not repaired or replaced all of the affected CSCS valves that are susceptible to separation of the valve disc from the valve stem. The first failure was in September 1996. The cause was determined to be vibration accelerated corrosion and erosion of the valves internal carbon steel components. There were at least four additional failures between 2002 and 2006. Corrective actions included the refurbishment or replacement of the 88 susceptible valves, as appropriate. As of this inspection, ten valves have not been refurbished or replaced. The valves are associated with safety-related and important-to-safety systems. The performance deficiency has a cross-cutting aspect in the area of Problem Identification and Resolution, Corrective Action Program, because the licensee did not take the appropriate corrective actions to address a safety issue in a timely manner, commensurate with the safety significance. [P.1(d)]
Inspection Report# : 2007006 *(pdf)*

Significance: G Sep 30, 2007
Identified By: Self-Revealing
Item Type: FIN Finding
Instrument Maintenance Technicians Cause Pressure Spike when Valving in Flow Transmitter and Render RHR Train Inoperable
A self-revealing finding of very low safety significance was identified following the inadvertent actuation of the 2B RHR pump minimum flow valve (2E12-F064B) during a Unit 2 RHR pump 2B/2C flow indication calibration. Specifically, licensee instrument technicians returning the 2E12-N015B flow instrument to service created a pressure spike when valving in the instrument following calibration; the pressure spike was sufficient to cause the 2B RHR pump injection flow high alarm (2H13-P601-B307) setpoint to be reached, and the 2B RHR pump minimum flow valve automatically repositioned shut as a result. No violations of NRC requirements or regulations were identified by the inspectors.

The performance deficiency associated with this finding involved the failure of licensee instrument maintenance technicians to exercise due caution when restoring the 2E12-N015B transmitter (2B RHR pump flow indication) to service following calibration, such that a pressure spike was created that caused the 2B RHR pump minimum flow valve to automatically reposition shut. This error resulted in the unnecessary and unintentional actuation of a safety-related component, and rendered the 2B RHR train inoperable per Technical Specifications. The finding was determined to be of more than minor significance in that it had a direct impact on the objective for the Mitigating Systems Cornerstone for Reactor Safety. Because the finding did not represent the actual loss of a safety function for any single train or system, and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event, the inspectors concluded that it was of very low safety significance and within the licensee’s response band. In addition, the inspectors determined that the finding was related primarily to the cross-cutting area of Human Performance since the licensee’s instrument maintenance technicians did not appropriately utilize applicable human error prevention techniques, such as self checking, etc., when restoring the 2E12-N015B flow transmitter to service (Aspect H.4(a)). Corrective actions planned and completed by the licensee included the performance of a quick human performance investigation and conducting a detailed apparent cause evaluation analysis for the event.
Inspection Report# : 2007004 *(pdf)*

Significance: G Sep 28, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Periodically Test Keylock Switches
The inspectors identified a finding of very low safety significance and an associated NCV of the LaSalle County Station Facility Operating License associated with the Fire Protection Program for failure to ensure that all necessary testing was identified and performed. Specifically, the licensee failed to periodically test remote-local keylock control switches on the switchgear for the emergency buses which are required to implement a safe shutdown for a plant fire in accordance with the licensee’s Safe Shutdown Analysis described in Appendix H, Section H.4 of the Fire Protection Report. This issue was entered into the licensee’s corrective action program, and as a compensatory measure, the licensee implemented procedure changes to the safe shutdown procedures that gave direction to manually close a breaker if the breaker failed to close using the remote-local keylock switch. The licensee also successfully tested a portion of the remote-local switches and initiated efforts to determine a schedule for testing of

the remaining keylock switches.

The finding was more than minor because the licensee did not ensure the operability and functional performance of the remote-local keylock control switches to perform satisfactorily in service. The finding was of very low safety significance based on the results of a Phase 1 screening completed in accordance with IMC 0609, Appendix F, "Fire Protection Significant Determination Process."
Inspection Report# : 2007009 *(pdf)*

Significance: G Sep 28, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Translate Backwash Valve Settings into Procedures
The inspectors identified an NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," in that, the design bases for the manual backwash valve position values for the Diesel Generator Cooling Water (DGCW) backwash strainers were not

correctly translated into procedures and instructions. Specifically, the manual backwash valve positions derived from flow test surveillance procedures based on hydraulic calculation models were not translated into operations procedures for manual operation of the DGCW strainer backwash valves. This issue was entered into the licensee's corrective action program, and the licensee updated the applicable operating procedure to reflect the correct manual settings for the DGCW strainer backwash valves.

This issue was more than minor because the DGCW backwash valves could be manually opened more than required during a loss of power event, and thus divert some cooling flow from post accident required equipment. The finding was of very low safety significance based on a Phase 1 screening in accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," because on re-evaluation, the design function was maintained.
Inspection Report# : 2007009 *(pdf)*

Significance: G Sep 28, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Lack of Station Blackout Analysis for RCIC
A finding of very low safety significance was identified by the inspectors associated with a Non-Cited Violation of 10 CFR 50.63, "Loss of All Alternating Current Power." Specifically, the licensee did not have an appropriate analysis to determine the capability of coping with a station blackout, in that, it had no analysis that verified the proper operation of the reactor core isolation cooling (RCIC) turbine at the elevated suppression pool temperatures encountered during a station blackout event. This issue was entered into the licensee's corrective action program. The licensee obtained additional information and performed a preliminary analysis which showed that the RCIC turbine would operate as required.

This finding was more than minor because the licensee did not have an analysis that demonstrated the availability and reliability of the RCIC turbine at the elevated suppression pool temperatures encountered during a station blackout event. The issue was of very low safety significance based on a Phase 1 screening in accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situation," because the licensee obtained additional data from the RCIC turbine manufacturer and performed a functionality analysis which demonstrated the pump turbine could operate at heightened suppression pool temperatures.
Inspection Report# : 2007009 *(pdf)*

Significance: SL-IV Sep 28, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Lake Level Instrumentation Removed from Service without 10 CFR 50.59 Evaluation
The inspectors identified an NCV of 10 CFR Part 50.59, "Changes, Tests, and Experiments," which had very low safety significance. Specifically, the licensee failed to complete a 50.59 evaluation for removing main control room lake level instrumentation from service. Although the UFSAR stated that the lake level was continuously monitored in

the main control room, the level instrument had not functioned reliably for several years and was removed from the plant maintenance schedule in December 2005. At the time of the inspection, control room monitoring of the lake level was not available. The licensee entered the issue into their corrective action program and initiated more frequent operator rounds as a compensatory measure.

The finding was more than minor because the inspectors could not reasonably determine that this change would not have ultimately required prior approval from the NRC. This finding was categorized as Severity Level IV because the underlying technical issue for the finding was determined to be of very low safety significance based on a Phase 1 screening in accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situation."
Inspection Report# : 2007009 *(pdf)*

Significance: G Jun 30, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Licensee Relied on Operator Manual Actions for Post-fire SSD.
The inspectors identified a non-cited violation (NCV) of the LaSalle County Station Operating License for the failure to establish the required physical protection or separation of cables to ensure that one redundant train of systems necessary to achieve and maintain hot shutdown condition was free of fire damage. The licensee instead relied on operator manual actions for post-fire Safe Shutdown (SSD) in the event of a fire in non-alternate shutdown areas. The manual actions were not identified in the SSD procedures. Since the inspection in 2005, the licensee implemented appropriate procedure changes and incorporated the required manual actions.

The finding was more than minor because it affected the attribute of protection against external factors (i.e., fire) and it impacted the objective of the mitigating systems cornerstone. The failure to ensure that one redundant train of systems necessary to achieve and maintain hot shutdown condition free of fire damage and failure to provide adequate instructions for manual actions in shutdown procedures could have adversely impacted the operators's ability to promptly take appropriate actions and could have complicated safe shutdown in the event of a fire. The finding was of very low safety significance (Green) based on a Phase 1 SDP evaluation completed in accordance with IMC 0609, Appendix F, "Fire Protection Significance Determination Process."
Inspection Report# : 2007003 *(pdf)*

Significance: G Mar 31, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Incomplete Residual Heat Removal Heat Exchanger Vessel Weld Examinations.
The inspectors identified a finding of very low safety significance and an associated non-cited violation of 10 CFR 50.55a(g)4 for the licensee's failure to perform examinations of the ASME Code Section XI required weld volume for the Unit 1 and 2 'B' residual heat removal (RHR) heat exchanger shell welds. Specifically, the licensee completed only ? of the Code required weld examination volume for four shell welds on each heat exchanger vessel. The performance deficiency associated with this finding was the failure of the licensee to complete a full volumetric examination of the 1B and 2B RHR heat exchanger shell welds. This finding was of more than minor significance because it directly affected the Mitigating System Cornerstone objective of equipment performance (reliability). Because the finding did not represent a design or qualification deficiency that resulted in the loss of operability the inspectors concluded that it was of very low safety significance and within the licensee's response band. In addition, the inspectors also determined that the finding was related primarily to the cross-cutting area of Human Performance, since the licensee failed to ensure supervisory and management oversight of work activities, including contractors, such that nuclear safety was supported. Corrective actions planned and completed by the licensee included repeating the 'B' RHR heat exchanger shell weld examinations to ensure the required Code volume was covered.
Inspection Report# : 2007002 *(pdf)*

Barrier Integrity

Significance: May 18, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Inadequate Procedure for Removal of Drywell Head Bolts
Green. The inspectors identified a finding having very low safety significance and an associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," involving an inadequate maintenance procedure used to remove drywell head bolts. Specifically, in maintenance procedure MA-AB-756-600 "Reactor Disassembly," the licensee failed to provide instructions to remove only "every other bolt" to ensure that the drywell head assembly configuration remained within the analyzed configuration for operating Modes 1 through 3. As a corrective action, the licensee intended to provide additional procedure instructions to restrict bolt removal to every other bolt, or delete the procedure option for early bolt removal with the plant in Modes 1 through 3.

The finding was determined to be greater than minor because absent NRC intervention the inadequate procedure could lead to a more significant problem. Specifically, procedure MA-AB-756-600 would have allowed removal of bolts from adjacent locations on the drywell head assembly which could affect the structural and/or leakage integrity of the containment. The finding was of very low safety significance based on a Phase 1 screening in accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," because it did not represent an actual open pathway for containment, and did not involve a reduction in defense in depth for the atmospheric control or hydrogen control function of containment. The primary cause of this finding was related to the cross-cutting area of human performance because the licensee did not provide complete, accurate, and up to date design documentation to plant personnel. (Section 1R17)

Inspection Report# : 2007007 *(pdf)*

Significance: May 18, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Lack of Calibrated Air Wrench ofor Drywell Head Assembly Bolt Installation
Green. The inspectors identified a finding having very low safety significance and an associated NCV of 10 CFR Part 50, Appendix B, Criterion XII, "Control of Measuring and Test Equipment," involving lack of calibrated tools used to establish torque for the drywell head assembly bolts. Specifically, for five air hammer wrenches used to install drywell head assembly bolts on Unit 1 and Unit 2, the licensee failed to ensure these tools were properly calibrated to confirm the accuracy of the torque applied. The licensee entered this issue into the corrective action program, performed an operability evaluation, and concluded that sufficient torque had been applied to the drywell head bolts. The licensee operability conclusion was based upon the vendor advertised torque wrench specifications, torque margins available in the design analysis, and periodic air hammer wrench maintenance.

The finding was determined to be greater than minor because absent NRC intervention the lack of calibration testing for these wrenches could lead to a more significant problem. Specifically, the drywell head assembly bolts may not receive sufficient torque to establish a preload which assures containment leakage and structural integrity. The finding was of very low safety significance based on a Phase 1 screening in accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," because it did not represent an actual open pathway for containment, and did not involve a reduction in defense in depth for the atmospheric control or hydrogen control function of containment. This finding had a cross-cutting aspect in the area of human performance because the licensee did not provide adequate and available facilities and equipment (e.g. calibrated equipment) for personnel reassembling the drywell head. (Section 1R17)

Inspection Report# : 2007007 *(pdf)*

Significance: Mar 31, 2007
Identified By: NRC
Item Type: FIN Finding
De-Tensioning Drywell Head in Mode 3 Has Unanticipated Impact on Technical Specifications.
A finding of very low safety significance was identified by the inspectors during review of the licensee's activities associated with de-tensioning the drywell head in preparation for scheduled reactor refueling operations. Specifically,

the inspectors identified that the licensee had not performed a current Technical Specification required Type ‘B' local leak rate test (LLRT) with half of the drywell head closure bolts de-tensioned, such that when they performed the de-tensioning activity in Mode 3 the surveillance requirement was no longer met. Because the licensee took action in response to the inspectors' questions and completed a Type ‘B' LLRT on the drywell head with half of the closure bolts de-tensioned within the allowed outage time provided in the Technical Specifications, no violation of regulatory requirements was identified in conjunction with the finding. The performance deficiency associated with this finding was the licensee's failure to recognize the impact on the Technical Specifications from this activity until questioned by the inspectors. The finding was determined to be of more than minor significance in that if left uncorrected it would have represented a more significant safety concern. Specifically, absent NRC intervention, the licensee would have not performed a Type ‘B' LLRT within the Technical Specification action statement time limit and a Technical Specification violation would have resulted. Because the finding involved adequate mitigation capability, did not impact primary containment availability, and was not an event that could be characterized as a loss of control, the inspectors concluded that it was of very low safety significance and within the licensee’s response band. In addition, the inspectors determined that the finding was related primarily to the cross-cutting area of Human Performance since licensee personnel did not use conservative assumptions in decision-making and as such, did not identify the possible unintended consequences their actions. Corrective actions planned and completed by the licensee included the performance of an apparent cause evaluation, and actions for the licensee outage organization to flag any departures from normal practices and discuss these items at weekly pre-outage planning meetings. Other corrective actions included the performance of a 10 CFR 50.59 screening and/or evaluation to support the change to the reactor vessel disassembly procedure allowing the partial de-tensioning of the drywell head in Mode 3, and an action to evaluate potential changes to procedure LTS-100-15.
Inspection Report# : 2007002 *(pdf)*

Emergency Preparedness

Occupational Radiation Safety

Public Radiation Safety

Significance: G Sep 30, 2007
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
External Radiation Levels on Package Exceeds 200 mrem/hr on Contact
A self-revealing NCV of 10 CFR 71.5 was identified when a package of licensed material offered for shipment exceeded the external radiation limit contained in 49 CFR 173. The shipment was surveyed upon receipt at the final destination by individuals qualified in radioactive materials package receipt and the radiation levels at the package surface were in excess of 200 millirem (mrem)/hr. As a result of this event, the licensee changed the shipping procedure to require that all items placed in the package be surveyed prior to closure, survey and shipment.

The cause of the error was a failure to assure that all package contents were properly surveyed and secured so they could not shift and create a change in radiation field during transport. The finding, under the Public Radiation Safety Cornerstone, does not involve the application of traditional enforcement. The finding was more than minor as it involves an occurrence in the licensee’s radioactive material transportation program that is contrary to NRC or Department of Transportation (DOT) regulations and is a key attribute under the objective of the radiation safety cornerstone to ensure adequate protection of public health and safety from exposure to radioactive materials released into the public domain, as a result of routine civilian nuclear reactor operation. Although the limits for external radiation levels on a package were exceeded, the finding is of very low safety significance because the area of the package having the higher external radiation levels would note have been accessible to a member of the public. The inspectors determined that the finding had a cross-cutting aspect associated with problem identification and resolution,

in that the licensee did not implement and institutionalize operating experience through changes to procedures (Aspect P.2(b)).
Inspection Report# : 2007004 *(pdf)*

Physical Protection

Although the NRC is actively overseeing the Security cornerstone, the Commission has decided that certain findings pertaining to security cornerstone will not be publicly available to ensure that potentially useful information is not provided to a possible adversary. Therefore, the cover letters to security inspection reports may be viewed.

Miscellaneous

Last modified : February 04, 2008