Point Beach 1
4Q/2004 Plant Inspection Findings

Initiating Events

Significance: G Jun 30, 2004
Identified By: NRC
Item Type: FIN Finding
Potential Loss of Hot Leg Vent Path During Nozzle Dam Installation
The inspectors identified a finding associated with installing steam generator nozzle dams and establishing a hot leg vent path during a portion of the Unit 1 cycle 28 refueling outage (U1R28). The primary cause of this finding was related to the cross-cutting area of human performance, involving the decision by several licensed and experienced personnel to allow nozzle dam installation to commence prior to establishment of a vent path through the pressurizer manway.

The finding is considered more than minor because it affected: (1) the Reactor Safety Initiating Events Cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown operations, and (2) the human performance attribute of the Initiating Events Cornerstone. The finding was considered to be of very low safety significance and did not require quantitative assessment since: (1) conditions meeting a loss of control were not met in that no inadvertent change in reactor coolant system temperature or change in reactor vessel level actually occurred, and (2) the licensee had maintained adequate mitigation capability for the existing plant conditions. No violation of regulatory requirements occurred because: (1) the actual sequence of events showed that all four nozzle dams had not been completely installed while the pressurizer manway was still in place, and (2) an engineering analysis showed that an adequate hot leg vent path was available while one of the ‘A' steam generator hot leg nozzle dam side pieces was not installed. The licensee has entered this finding into its corrective action program.
Inspection Report# : 2004003*(pdf)*

Significance: G Jun 30, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
Loss of Transient Combustible Control in the Containment and Turbine Buildings During a Unit 1 Refueling Outage
The inspectors identified a Non-Cited Violation of 10 CFR 50.48(a)(2)(i) having very low safety significance when transient combustibles were stored in the Unit 1 containment building and the turbine building without required administrative controls. The finding also affected the cross-cutting area of human performance in that the licensee failed to identify the transient combustible materials during tours required by the Fire Protection Evaluation Report.

The inspectors concluded that the finding is more than minor because it affected the Reactor Safety Initiating Events Cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown, specifically protection against external factors (fire). The inspectors determined that the finding was of very low safety significance (Green), since the issue was assigned a low degradation rating and the quantity of transient combustibles had been bounded by the analysis contained in the Fire Hazards Analysis Report. The licensee has entered this finding into its corrective action program.
Inspection Report# : 2004003*(pdf)*

Mitigating Systems

Significance: G Dec 31, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
10 CFR 50, Appendix B, Criterion XVI, "Corrective Action." Failure to Take Corrective Actions for a Condition Adverse to Quality
A finding of very low safety significance was identified by the inspectors for a violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," for the failure to take actions for a condition adverse to quality. Specifically, in September 2003 a condition report was written to address the susceptibility of fouling of a small mesh strainer installed in a fire protection line which provided emergency cooling to the turbine driven auxiliary feedwater pumps and turbine bearing coolers. The condition report also identified that procedure guidance did not exist for operators to utilize an existing flush valve on the strainer if the strainer became clogged during use. The inspectors identified that in August 2004, the condition report was closed with no actions taken to address this condition adverse to quality. At the end of the inspection, the licensee took corrective actions to ensure that as a minimum, the appropriate procedural guidance existed if the strainer became clogged during use.

The inspectors also concluded the primary cause of this finding was related to the cross-cutting area of problem identification and resolution, because the licensee failed to take any corrective actions to correct this condition adverse to quality.

This finding was more than minor because if left uncorrected the finding could become a more significant safety concern. In addition, the finding affected the mitigating systems cornerstone objective to ensure the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. In accordance with the Significance Determination Process, this finding was determined to be a Non-Cited Violation of very low safety significance because it was not a design or qualification deficiency that was confirmed to result in a loss of function per Generic Letter 91-18.

Inspection Report# : 2004012*(pdf)*

Significance: SL-IV Dec 31, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Perform a Safety Evaluation as Required by 10 CFR 50.59, "Changes, Tests and Experiments"
The inspectors identified a Severity Level IV Non-Cited Violation of 10 CFR 50.59(d)(1) for the licensee's failure to perform a safety evaluation for changes made to the Final Safety Analysis Report. Specifically, the licensee ‘screened out' a change to the Final Safety Analysis Report which modified operator response times for the Steam Generator Tube Rupture Chapter 14 Accident Analysis contained in the Final Safety Analysis Report. Specifically, a time requirement for equalizing primary and secondary pressure was removed from the Final Safety Analysis Report. In addition, the licensee changed the time in which isolation of the affected Steam Generator could be achieved from 10 minutes to 30 minutes. At the end of the inspection period the licensee initiated a corrective action to perform a safety evaluation in accordance with 10 CFR 50.59 for this Final Safety Analysis Report change.

Because the Significance Determination Process is not designed to assess the significance of violations that potentially impact or impede the regulatory process, this issue was dispositioned using the traditional enforcement process in accordance with Section IV of the NRC Enforcement Policy. However, the results of the violation were assessed using the Significance Determination Process.

This finding was determined to be more than minor because the inspectors could not reasonably determine that the change would not ultimately require NRC approval. The inspectors determined that even though the change was not adequately evaluated in accordance with 10 CFR 50.59, this violation was of very low safety significance because the design basis safety-related functions of mitigating systems to respond to this initiating event scenario were not adversely affected. The inspectors evaluated the results of the finding using the Significance Determination Process for the mitigating systems cornerstone. The inspectors determined that the results of the finding were of very low safety significance because the finding was not a design or qualification deficiency that was confirmed to result in a loss of function per Generic Letter 91-18. Therefore, the results of the violation were determined to be of very low safety significance and the violation was classified as a Severity Level IV Non-Cited Violation.
Inspection Report# : 2004012*(pdf)*

Significance: G Dec 31, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
10 CFR 50, Appendix B, Criterion XI, "Test Control." Failure to Have Adequate Test Procedures for the Testing of Safety-Related Switches
A Green finding associated with a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," was identified by the inspectors for the failure to establish and perform testing required to demonstrate that components will perform satisfactorily in service with written test procedures which incorporate applicable requirements and acceptance limits. The licensee performed post-maintenance testing of a component cooling water pump control switch, a safety-related component, without the use of a written test procedure which incorporated the applicable requirements and acceptance limits for testing to demonstrate the component would perform satisfactorily in service. The licensee's extent of condition identified the potential for at least 11 additional activities for which safety-related components did not have the appropriate test procedures established. At the end of the inspection period, the licensee developed actions to correct the identified deficiencies and to ensure licensee personnel were aware of the requirements to use procedures for the testing of safety-related components.

This issue was more than minor because if left uncorrected the finding could become a more significant safety concern. In addition, the finding affected the mitigating systems cornerstone attribute of procedure quality, specifically maintenance and testing (pre-event) procedures, and the cornerstone objective to ensure the reliability of systems that respond to initiating events to prevent undesirable consequences. In accordance with the Significance Determination Process, this finding was determined to be a Non-Cited Violation of very low safety significance because the finding was not a design or qualification deficiency that was confirmed to result in a loss of function per Generic Letter 91-18.
Inspection Report# : 2004012*(pdf)*

Significance: G Nov 19, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Ensure That a Safe Shutdown Procedure Directed Alignment of Instrumentation to a Direct Current Bus with a Battery Charger

A finding of very low safety significance was identified by the inspectors for failure to align safe shutdown instrumentation to an electrical bus with a battery charger in procedure AOP-10A, "Safe Shutdown - Local Control." Specifically, the procedure aligned Units 1 and 2 safe shutdown instrumentation to a 125Vdc bus that did not have a battery charger available to support the selected instrumentation.

This issue was more than minor because it affected the procedure quality attribute of the Reactor Safety Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events. Specifically, the safe shutdown instrumentation associated with this bus, without a battery charger, could potentially become inoperable as the voltage of the battery supplying the bus decreased. Operators could select another bus with a safe shutdown inverter, however, the procedure did not direct this action. To correct this procedural error, the licensee issued Temporary Change Notice 2004-0762. This issue was entered into the licensee's corrective action program as CAP059262 and CE014635. The issue was of very low safety significance because it did not represent an actual loss of a safety function. The issue was a Non-Cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criterion V, "Instruction, Procedures, and Drawings," for failure to provide a procedure of a type appropriate to the circumstances.
Inspection Report# : 2004010*(pdf)*

Significance: G Nov 03, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
Vendor Breaker Testing Requirements Not Incorporated in Procedure
The inspectors identified a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," because the licensee did not evaluate a Technical Bulletin issued by Westinghouse in March 2004 regarding safety-related breakers and incorporate the testing instructions specified in the Bulletin into the applicable station procedures.

The finding was greater than minor because it was associated with the procedure quality attribute of the Reactor Safety Mitigating Systems cornerstone and affected the associated cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). The finding is of very low significance as it did not involve a design or qualification deficiency, did not represent a loss of safety function, and did not involve an external initiating event. The licensee entered the issue into its corrective action program. As part of corrective actions, the licensee evaluated the Technical Bulletin and incorporated the testing instructions into applicable station procedures.
Inspection Report# : 2004008*(pdf)*

Significance: G Nov 03, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
Corrective Actions for a Part 21 Notification on Diesel Governors Were Not Timely
The inspectors identified a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," because the licensee failed to promptly evaluate and resolve a 10 CFR Part 21 issue from 2001 involving the governors on all four emergency diesel generators (EDGs). The Part 21 issue pertained to the service life of electrolytic capacitors in the governor control system of all four safety-related EDGs. The capacitors in the four EDGs were beyond the service life specified by the vendor in the Part 21 and, in three of four EDGs, the capacitors were beyond the industry's slightly longer replacement interval.

The finding is greater than minor because it was associated with the equipment performance attribute of the Reactor Safety Mitigating Systems cornerstone and affected the associated cornerstone objective of ensuring the availability, reliability, and capability of systems (the EDGs) that respond to initiating events to prevent undesirable consequences (i.e., core damage). The finding is of very low safety significance because it did not involve a design or qualification deficiency, did not represent a loss of safety function, and did not involve an external initiating event. The licensee entered the issue into its corrective action program and evaluated a recent industry study that indicated a slightly greater service life of the capacitors. In addition, the licensee has made plans to replace the capacitors on an accelerated schedule.
Inspection Report# : 2004008*(pdf)*

Significance: G Nov 03, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Implement a Molded-Case Circuit Breaker Test Program
The inspectors identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," because the licensee failed to implement a program to assure that the installed molded-case circuit breakers (MCCBs) will perform satisfactorily in service.

The finding was greater than minor because it was associated with the Reactor Safety Mitigating Systems cornerstone attribute of equipment performance and affected the cornerstone objective of ensuring the availability, reliability, capability of systems that responds to initiating events to prevent undesirable consequences (i.e., core damage). Molded-case circuit breakers provide for breaker coordination, over-current protection, fire prevention, and multiple other safety-related functions. The finding is of very low safety significance because it did not involve a design or qualification deficiency, did not represent a loss of safety function, and did not involve an external initiating event. The licensee entered the issue into its corrective action program. As part of its corrective actions, the licensee planned to institute an exercising and testing program for safety-related MCCBs.
Inspection Report# : 2004008*(pdf)*

Significance: G Nov 03, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
Vendor Torque Values Not Listed in Procedure
The inspectors identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," having very low safety significance. Specifically, the licensee failed to incorporate the vendor's torque requirements for breaker arc chute fasteners into station procedures.

The finding is greater than minor because it was associated with the procedure quality attribute of the Reactor Safety Mitigating System cornerstone and affected the associated cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). The finding is of very low safety significance because it did not involve a design or qualification deficiency, did not represent a loss of safety function, and did not involve an external initiating event. The licensee entered the issue into its corrective action program and revised the procedure to include the vendor's torque requirements.
Inspection Report# : 2004008*(pdf)*

Significance: G Sep 30, 2004
Identified By: NRC
Item Type: FIN Finding
Unit 1 Residual Heat Removal Heat Exchanger Bypass Valve Drifts Open While in Automatic
The inspectors identified a workaround regarding the operation of the Unit 1 residual heat removal system heat exchanger bypass flow control valve in automatic mode during a shutdown loss-of-coolant-accident. The primary cause of this finding was related to the cross-cutting area of problem identification and resolution in two respects. First, the initial extent-of-condition review did not consider the impact of the issue on shutdown plant operations. Second, following initial instrumentation and control troubleshooting efforts, a corrective action item was not assigned to operations personnel to evaluate the issue as a potential operator workaround. This contributed to a 3-month delay in completing the evaluation.

The finding is greater than minor because it affected the equipment performance attribute of the Reactor Safety Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events. The finding was considered to be of very low safety significance (Green) because it did not degrade short term (safety injection) decay heat removal capability or reactivity control; result in a design or qualification deficiency or an actual loss of safety function; or involve internal or external initiating events. The finding did not involve a violation of regulatory requirements. The licensee has entered this finding into its corrective action program. In addition, the finding was reviewed by the licensee's Operator Workaround Committee and the Committee classified the problem as an operator challenge in accordance with site procedures.
Inspection Report# : 2004006*(pdf)*

Significance: G Jul 16, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Test Service Water Headers
The inspectors identified a Non-Cited Violation of 10 CFR 50.55a(g)(4) and 10 CFR 50.55a(g)(5)(iv) associated with failure to perform testing of the buried service water header piping in accordance with the American Society of Mechanical Engineers Code Section XI requirements. The licensee's corrective actions included verifying that quarterly system flow tests provided basis for service water header operability.

This finding was more than minor because it affected the Mitigating Systems Cornerstone objective of equipment reliability and if left uncorrected, could have allowed undetected through-wall flaws to develop in the header piping. These flaws could then continue to grow in size until leakage from the buried headers degraded system operation or if sufficient general corrosion occurs, a gross rupture or collapse of the piping sections could occur. The finding is of very low safety significance and screened as Green using the Significance Determination Process Phase 1 screening worksheet.
Inspection Report# : 2004004*(pdf)*

Significance: G Jul 16, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
Non-Code Repair to Service Water (SW) Valve SW 0322
The inspectors identified a Non-Cited Violation of 10 CFR 50.55a(g)(4) associated with failure to conduct non-destructive examinations and repair of valve SW 0322 in accordance with American Society of Mechanical Engineers Code Section XI requirements. The licensee's corrective actions included replacement of the valve during the next opportunity.

This finding was more than minor because it affected the Mitigating Systems Cornerstone objective of equipment reliability and if left uncorrected, could have allowed unacceptable base metal flaws to remain in service. Additionally, the failure to heat treat the weld repairs could have resulted in high welding residual stresses and untempered martensite formation. Untempered martensite is a hard brittle phase of steel (e.g., not flaw tolerant) and can serve to allow rapid crack propagation that could jeopardize the pressure retaining function of the valve

body. The finding is of very low safety significance and screened as Green using the Significance Determination Process Phase 1 screening worksheet.
Inspection Report# : 2004004*(pdf)*

Significance: G Jul 16, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
Non-Code Repair to Service Water (SW) Valves SW 32C and SW 32F
The inspectors identified a Non-Cited Violation of 10 CFR 50.55a(g)(4) associated with failure to implement the American Society of Mechanical Engineers Code Section XI examinations and repair requirements for service water pump discharge check valves SW 32C and SW 32F. The licensee's corrective actions included verifying that quarterly surveillance tests verified check valve operability.

This finding was more than minor because it affected the Mitigating Systems Cornerstone objective of equipment reliability and if left uncorrected, the failure to perform the required examinations could have allowed unacceptable base metal flaws to remain in-service. Additionally, the failure to select and follow a repair Code or standard may have resulted in inadequate post weld heat treatments for the weld repairs that could result in high welding residual stresses and untempered martensite formation. Untempered martensite is a hard brittle phase of steel (e.g., not flaw tolerant) and can serve to allow rapid crack propagation which could jeopardize the pressure retaining function of these valve disks. The finding is of very low safety significance and screened as Green using the Significance Determination Process Phase 1 screening worksheet.
Inspection Report# : 2004004*(pdf)*

Significance: G Jul 16, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Correctly Translate Condensate Storage Tank Temperature Limits into Procedures and Instructions
The inspectors identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," in that the design bases for the maximum Condensate Storage Tank (CST) temperature was not correctly translated into procedures and instructions. Specifically, the Main Steam Line Break (MSLB) Containment Integrity Analysis assumed a maximum value of 100 degrees Fahrenheit for the temperature of the water in the CST, while operations procedures allowed a maximum of 120 degrees Fahrenheit for the CST temperature. This finding applies to both units. The licensee's corrective actions included procedural changes to reflect the correct temperature limit.

This finding was more than minor because an evaluation was required to ensure that accident analysis requirements were met, since the CST was heated up to greater than the maximum analysis value of 100 degrees Fahrenheit during unit startup/shutdown operations with the CST aligned to the operating unit. The finding is of very low safety significance and screened as Green using the Significance Determination Process Phase 1 screening worksheet.
Inspection Report# : 2004004*(pdf)*

Significance: G Jul 16, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Periodically Verify Position of Valves in the Service Water (SW) System
The inspectors identified a Non-Cited Violation of Technical Specification Surveillance Requirements SR 3.7.8.1 and SR 3.6.3.2 associated with the periodic verification of the position of valves and flanges in the SW system flow paths servicing safety related equipment and in lines associated with containment isolation. Specifically, the licensee did not verify that approximately 100 valves in the SW system flow path servicing safety related equipment that were not locked, sealed, or otherwise secured in position, were in the correct position every 31 days while the Units were in Mode 1, 2, 3, or 4. In addition, the licensee did not verify that 12 containment isolation manual valves were closed and two pipe fittings associated with containment isolation were in place every 31 days while the Units were in Mode 1, 2, 3, or 4. This finding applies to both units. The licensee's corrective actions included locking the appropriate valves and procedural changes.

This finding was more than minor because it was, for the most part, associated with the Mitigating Systems attribute of Configuration Control, which affected the Mitigating Systems Cornerstone objective of ensuring the availability and reliability of the SW system to respond to initiating events to prevent undesirable consequences. The finding is of very low safety significance and screened as Green using the Significance Determination Process Phase 1 screening worksheet.
Inspection Report# : 2004004*(pdf)*

Significance: G Jul 16, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Translate Original Design Requirements for the 480-Volt Alternating Current (Vac) System
The inspectors identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to adequately translate original design requirements for the 480 Vac system into specifications during procurement of new and replacement

equipment. The original specifications for equipment such as motors and cables identified the intended service as suitable for a 480 Vac ungrounded system. Specifications for replacement motors did not specify the intended service as an ungrounded system. The licensee's corrective actions included a verification that the identified equipment that did not specify use in a 480 Vac ungrounded system could withstand the overvoltage conditions that can occur on ungrounded systems.

This finding was more than minor because it involved the design control attribute of the Mitigating Systems cornerstone and affected the objective of ensuring the capability of the safety related 480 Vac system in response to initiating events to prevent undesirable consequences. Specifically, the failure to specify the correct service conditions may have resulted in motors being supplied without the enhanced insulation systems required to withstand the overvoltage conditions that can occur on ungrounded systems when a single line to ground occurs. The finding is of very low safety significance and screened as Green using the Significance Determination Process Phase 1 screening worksheet.
Inspection Report# : 2004004*(pdf)*

Significance: G Jun 30, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
Substitution of Weld Surface Examinations for Volumetric Examinations
The inspectors identified a Non-Cited Violation of 10 CFR 50.55a(a)(3)(i) for the licensee's incorrect substitution of weld surface examinations into the risk-based portion of the Inservice Inspection Program, which required volumetric weld examinations.

This finding is greater than minor because it affected the Mitigating Systems Cornerstone objective of equipment reliability and, if left uncorrected, could allow unacceptable piping system weld flaws to remain in-service and render safety-related systems inoperable. The finding is of very low safety significance because the licensee had sufficient time left in the Code interval to perform the required number of volumetric examinations of piping welds in the affected risk-based category during future Unit 1 outages. The licensee has entered this finding into its corrective action program

Inspection Report# : 2004003*(pdf)*

Significance: G Jun 30, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Control Unit 1 Emergency Operating Procedure Sub-Steps Committed to as Compensatory Measures in Accordance with NRC Bulletin 2003-01 Option 2
The inspectors identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion VI, "Document Control," having very low safety significance associated with Unit 1 emergency operating procedures when a software error deleted reference to two of five indications intended to monitor primary containment sump performance during the recirculation phase of a design basis accident. Specifically, the RHR Pump Operation - NORMAL and SI Pump Operation - NORMAL substeps of Unit 1 emergency operating procedure EOP-1, "Loss of Reactor or Secondary Coolant," Step 29c, Revision 35, were deleted by the software program and not detected by operations personnel for a period of approximately 9 months. The primary cause of this finding was related to the cross-cutting area of human performance in that despite previous knowledge of the software problem and operations department management expectations to perform line-by-line reviews prior to distribution, 16 errors occurred in safety-related emergency operating, emergency contingency action, critical safety, and shutdown emergency procedures for Units 1 and 2.

The inspectors determined that the finding is more than minor because it affected the procedure quality attribute of the Mitigating Systems Cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events. The finding was considered to be of very low safety significance because it did not result in a design or qualification deficiency, an actual loss of safety function, or involve internal or external initiating events. The licensee has entered this finding into its corrective action program.
Inspection Report# : 2004003*(pdf)*

Significance: G Mar 31, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
Sprinkler Head Locations Not in Accordance with Fire Code
The inspectors identified a Non-Cited Violation of the license for the failure of the licensee to install sprinkler heads in accordance with the applicable fire code in the component cooling water pump area. Specifically, the sprinkler heads were located a greater distance below the ceiling than permitted by code.

This finding was more than minor because it was associated with the protection against external factors (i.e., fire) attribute of the mitigating systems reactor safety cornerstone and affected the cornerstone objective in that a fire protection feature (i.e., an automatic suppression system) was adversely affected. The finding was of very low safety significance because manual fire fighting and auxiliary feedwater could be credited. This issue is a violation of a license condition and the applicable fire code which requires that sprinkler heads be located near the ceiling.
Inspection Report# : 2004002*(pdf)*

Significance: N/A Mar 24, 2003

Identified By: NRC
Item Type: VIO Violation
The failure to identify the root cause and implement corrective actions for the AFW/IA issue, a significant condition adverse to quality, so as to prevent recurrence.
A violation was identified for the licensee's failure to implement adequate corrective actions to effectively address a previous Red finding and preclude recurrence (Inspection Report 50-266/01-17; 50-301/01-17). Specifically, the licensee failed to identify potential common mode failures that existed involving power supplies to the recirculation line air-operated valve and other system components. In addition, the licensee's corrective actions for the potential common mode failure associated with a loss of instrument air did not preclude repetition. Specifically, the licensee's corrective actions, to upgrade the safety function of the air-operated recirculation valve, failed to ensure that successful operation of the recirculation line air-operated valve was dependent only on safety-related support systems. Following the corrective actions, successful operation of the valve was still dependent upon nonsafety-related power to an interposing relay. Additionally, the corrective actions failed to discover a single failure mechanism involving a system orifice modification.

The issue was more than minor because the failure to implement appropriate corrective actions resulted in the auxiliary feedwater system continuing to rely on nonsafety-related support systems and to be susceptible to a single event causing a total system failure. The failure of nonsafety-related support systems and single event failures are an expected condition during several design basis accidents and should not cause a safety system to fail. The failure of the licensee to implement adequate corrective actions is a violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action."

This violation is associated with a previously identified RED finding (IR 50-266;50-30/01-17).
Inspection Report# : 2002015*(pdf)*

Significance: Y Mar 24, 2003
Identified By: NRC
Item Type: VIO Violation
Apparent violation of 10 CFR Part 50, Appendix B, Criterion III for the failure to establish appropriate design control measures for the installation of orifices to the AFW recirculation lines
An apparent violation was identified, in part, through a self-revealing event when decreased auxiliary feedwater pump recirculation flow was noted during post-maintenance testing. Subsequent licensee and NRC review of the event determined that the licensee had installed incorrectly designed orifices in each of the pump recirculation lines. The orifices, due to small clearances, were susceptible to plugging. The primary causes of this finding were inadequacies in the licensee's design process and the licensee's implementation of the process, including the identification of system design requirements and the development of supporting safety evaluations.

The issue has been preliminarily determined to have high safety significance (Red). Following installation of the inadequately designed orifices, the entire auxiliary feedwater system was susceptible to a common mode failure during operations using service water. Failure of auxiliary feedwater during several initiating events could lead to core damage. The installation of the incorrectly designed orifices in the recirculation lines is an apparent violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control."

On December 11, 2003, the final significance determination letter was issued for this finding. It was determined that this is a RED finding for Unit 2 and a YELLOW finding for Unit 1. For tracking purposes, identical findings were opened for Unit 1 (designated as YELLOW) and Unit 2 (designated as RED).
Inspection Report# : 2002015*(pdf)*

Significance: R Feb 28, 2002
Identified By: Licensee
Item Type: VIO Violation
POTENTIAL COMMON MODE FAILURE OF AUXILIARY FEEDWATER PUMPS DUE TO INADEQUATE PROCEDURAL GUIDANCE
Units 1 and 2. The licensee identified a potential common mode failure of the auxiliary feedwater pumps due to operator actions specified in plant procedures. The team identified that procedural guidance provided to operators was inadequate to prevent such a common mode failure. In addition, the team identified that the licensee had seven opportunities, from 1981 through 1997, to identifiy the problem and take appropriate corrective actions. After considering the information developed during the inspection and the information the licensee provided at the April 29, 2002, regulatory conference, the NRC concluded that a violation of 10 CFR Part 50, Appendix B, Criterion XVI, was appropriate for two of the originally proposed seven examples. The failures to provide adequate procedural guidance and to take appropriate corrective actions are both a violation of 10 CFR Part 50, Appendix B, Criteria V and XVI. This issue has been determined to have high safety significance (Red). A common mode failure of the auxiliary feedwater pumps would result in substantially reduced mitigation capability for safely shutting down the plant in response to certain transients. The significance was determined to be high largely due to the relatively high initiating event frequencies associated with the involved transients and the high likelihood of improper operator actions due to the procedural inadequacies. The final significance determination for the Red finding and Notice of Violation were issued to the licensee in a letter dated July 12, 2002.

Inspection Report 50-266/02-15; 50-301/02-15, issued April 2, 2003, documented the NRC decision that this finding is not an Old Design Issue.
Inspection Report# : 2001017*(pdf)*
Inspection Report# : 2003003*(pdf)*

Barrier Integrity

Significance: G Mar 31, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
Containment Upper Hatch Interlock
The inspectors identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance (Green) for failing to properly document a modification of the containment hatch interlock. The licensee failed to perform an engineering design change analysis for the Unit 1 personal containment hatch upper interlock cable when it was identified that original design specifications were not met. Specifically, the cable was replaced with a smaller cable prior to 2000 and again in 2000. When the cable broke in 2004, engineers replaced the cable with one that met the original design specifications, correcting the violation.

The inspectors determined that the finding was more than minor because it affected the barrier integrity reactor safety cornerstone objective attribute of maintaining functionality of containment design control. The finding was considered to be of very low safety significance because it did not result in an actual open pathway in the physical integrity of the reactor containment or actual reduction of the atmospheric pressure control function of the reactor containment.
Inspection Report# : 2004002*(pdf)*

Emergency Preparedness

Significance: G Mar 31, 2004
Identified By: NRC
Item Type: FIN Finding
Steam Generator Narrow Range Level Setpoints Revised in Safety-Related Procedures but Not in Emergency Plan General Emergency EAL 3.1.1.4
The inspectors identified a finding of very low safety significance concerning an inadequate extent-of-condition review during safety-related procedure revisions associated with steam generator narrow range level setpoints, and the failure to recognize the impact of the setpoint changes on the Point Beach Emergency Plan. The primary cause of this finding was related to the cross-cutting area of human performance in four respects. First, at least four personnel, including a Shift Manager and two senior reactor operators, reviewed the procedure changes but failed to recognize the potential impact of the procedure changes on the emergency plan. Second, personnel associated with the corrective action process for the initial steam generator narrow range level density compensation issue failed to recognize the potential emergency plan impact and raise the issue to the attention of emergency preparedness personnel. Third, despite the emergency preparedness reviews completed prior to and during the 95003 supplemental inspection process, the licensee had not identified and evaluated the potential impacts of the discrepancy between the procedure setpoints and Emergency Action Level 3.1.1.4. Fourth, until identified by the inspectors, personnel involved with efforts to achieve regulatory compliance with eight emergency action levels during January 2004, had not recognized or evaluated the potential impact of the discrepancy.

This finding was considered more than minor because it: (1) involved the procedure quality attribute of the emergency preparedness reactor safety cornerstone; and (2) if left uncorrected, it could become a more significant safety concern if the discrepancy in steam generator narrow range level setpoints prevented, or caused a delay in, declaring a general emergency during a loss of electrical power event. The finding was not considered a violation of regulatory requirements.
Inspection Report# : 2004002*(pdf)*

Significance: SL-III Dec 16, 2003
Identified By: NRC
Item Type: VIO Violation
10 CFR 50.54, 10 CFR 50.47 apparent violation for failure to maintain a standard scheme of emergency action levels
The inspectors identified an apparent violation of 10 CFR 50.54(q), associated with emergency planning standard 10 CFR 50.47(b)(4), which will be subject to the NRC traditional enforcement process not the revised Reactor Oversight Process. Specifically, the licensee failed to maintain a standard scheme of emergency action levels (EALs). Eight EALs were changed in 1998 and 1999. The changes decreased the effectiveness of the Emergency Plan in that emergency conditions that would have resulted in classifications at the General Emergency (GE), Alert, and Notification of Unusual Event (NOUE) levels would result in a lesser classification under the current EAL scheme. Approval of the NRC was not obtained prior to the changes being made. Since the identification of the issue by the inspectors, the licensee has revised the eight EALs to be equivalent with those approved by the NRC in 1984.

In a letter dated March 17, 2004, a Notice of Violation and Proposed Imposition of Civil Penalty - $60,000, was issued.
Inspection Report# : 2003007*(pdf)*

Occupational Radiation Safety

Significance: G Jun 30, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Follow Procedures in the Issuance and Use of Bubble Hood-type Respiratory Protective Devices
A finding of very low safety significance and an associated Non-Cited Violation were identified through an NRC-identified event, when on April 9, 2004, while installing steam generator nozzle dams, licensee staff increased supplied breathing air pressure in excess of procedural requirements while attempting to mitigate lost or diminished air flow to contract workers who were utilizing continuous flow, supplied-air respirator "bubble hoods." The inspectors determined that the licensee failed to meet the requirements of 10 CFR 20.1703, when the licensee increased the air line pressure in excess of the procedural guidance, which resulted in the licensee utilizing a respiratory protection device contrary to its National Institute for Occupational Safety and Health certification.

The inspectors determined that the finding is more than minor because use of a respiratory protection device outside its specifications could impact internal dose, and if left uncorrected, could become a more significant safety concern. The finding was considered to be of very low safety significance because no internal exposure to radioactive material resulted from the use of the bubble hoods with higher air line pressure than allowed. The licensee has entered this finding into its corrective action program.
Inspection Report# : 2004003*(pdf)*

Public Radiation Safety

Physical Protection

Physical Protection information not publicly available.

Miscellaneous

Last modified : March 09, 2005