Duane Arnold
4Q/2007 Plant Inspection Findings

Initiating Events

Significance: G Sep 30, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
INADEQUATE RECIRCULATION RISER WELD EXAMINATION (NOZZLE N2F).
The inspectors identified a non-cited violation (NCV) of 10 CFR Part 50, Appendix B, Criterion V for the licensee's failure to follow an ultrasonic examination procedure used to examine recirculation riser safe-end to nozzle weld RRF-F002. Specifically, the licensee's contracted ultrasonic data analyst failed to achieve adequate search unit contact with the weld surface in accordance with the procedure, prior to analyzing and accepting this weld examination in April of 2005. Because the licensee failed to achieve adequate ultrasonic search unit contact, an undetected intergranular stress corrosion crack was returned to service for one operating cycle, which placed the reactor coolant pressure boundary at increased risk for weld failure resulting in leakage. The licensee confirmed that this examination procedure and equipment had not been used for examination of other welds at the Duane Arnold Energy Center. In March of 2007, the licensee completed a weld overlay repair on RRF-F002 to mitigate this cracked weld.

This finding was of more than minor significance because the finding could be reasonably viewed as a precursor to a significant event involving an undetected weld crack that propagates to weld failure. This increased risk of weld failure and leakage adversely affected the Initiating Events cornerstone attribute of "Equipment Performance," and affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The inspectors applied the IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for the At-Power Situation," to this finding. The inspectors answered "yes" to Question 1 "Loss of Coolant Accident (LOCA) Initiators" of the Initiating Events Cornerstone column of the Phase 1 worksheet, which asked, "Assuming worst case degradation, would the finding result in exceeding the Technical Specification (TS) limit for identified reactor coolant system leakage?" For this finding, the worst case degradation would result from propagation of this weld crack under operating pressure and residual weld stresses causing leakage or failure at the 13-inch diameter recirculation nozzle weld RRF-F002, which would exceed the TS limit of no pressure boundary leakage. The Phase 1 worksheet required a significance determination process Phase 2 analysis for this type of finding. Because the increase in initiating event likelihood for LOCAs was not known, it was conservatively increased by one order of magnitude in accordance with Step 1.2 of Attachment 2 of Appendix A of IMC 0609. The inspectors completed the Phase 2 worksheets assuming that the initiating event frequency for small, medium and large break LOCAs had increased by one order of magnitude. Based on this Phase 2 evaluation, the NRC determined that this finding was of very low safety significance.

Inspection Report# : 2007004 *(pdf)*

Significance: G Jun 30, 2007
Identified By: Self-Revealing
Item Type: FIN Finding
OPERATORS FAILED TO CONTROL A CRITICAL PARAMETER AND RECEIVED A SUBSEQUENT AUTOMATIC SCRAM SIGNAL.
A finding of very low safety significance was self-revealed regarding the failure of the control room crew to establish control of a critical parameter, Reactor Pressure Vessel (RPV) level, in a timely manner during the feedwater recovery efforts following the manual scram initiated due to the loss of the 1A2 bus and the associated loss of the 'B' Reactor Feed Pump (RFP) and 'B' Condensate Pump. This resulted in a second automatic reactor protection system (RPS) actuation on low RPV level. The inspectors determined that the failure to ensure positive control of RPV level to preclude receiving an automatic protective system actuation was a performance deficiency warranting further evaluation. The licensee subsequently restored feedwater flow, completed the reactor shutdown, and entered this issue into their corrective action program. This finding did not result in a violation of NRC requirements.

The finding was more than minor because it adversely impacted the initiating events cornerstone attribute for human performance which limits the likelihood of events that upset plant stability and challenge critical safety functions. Although the crew's actions resulted in an automatic RPS actuation, the finding was determined to be of very low safety significance since it did not impact any mitigating systems capability. Additionally, no violations of NRC requirements occurred.
Inspection Report# : 2007003 *(pdf)*

Significance: G Mar 31, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO PERFORM QUALIFIED UT REACTOR VESSEL SHELL WELDS.
The inspectors identified a Non-Cited Violation (NCV) of 10 CFR 50.55a(g)4 for failure to complete a Code qualified volumetric examination of the reactor vessel shell welds during the previous refueling outage No.19. Specifically, the licensee used a longer cable length than that used in the ultrasonic examination procedure demonstration, which may have affected the flaw detection capability. On February 19, 2007, the licensee submitted a relief request to allow deferral of the affected reactor vessel weld examinations until the next refueling outage. The cause of this finding was related to the work practices component of the human performance cross-cutting area because the licensee failed to ensure adequate oversight of vendor activities with respect to review of the vendor's procedure for examination of reactor vessel welds. Specifically, the licensee approved procedure ISwT-PDI-AUT1, "Automated Inside Surface Ultrasonic Examination of Ferritic Vessel Wall Greater Than 4.0 Inches in Thickness," without adequately understanding and challenging the vendor's basis for changing essential procedure variables.

This finding was of more than minor significance because the finding could be reasonably viewed as a precursor to a significant event involving the ability to detect weld flaws prior to weld failure. In addition, the finding was associated with the Initiating Events cornerstone attribute of "Equipment Performance," and affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Absent NRC intervention, the licensee would have relied on an unqualified ultrasonic examination of reactor vessel shell welds for an indefinite period of service, which may have placed reactor coolant pressure boundary welds at increased risk for undetected cracking, leakage, or component failure. Based on review of industry operational experience, the inspectors did not identify any active degradation mechanisms which affect reactor vessel shell welds. Absent active degradation mechanisms, the inspectors concluded that a structurally significant flaw had not likely developed since completion of the last Code qualified vessel weld ultrasonic examination during the second Code interval (i.e., about 11 years earlier). Therefore, based upon NRC management review using qualitative measures of risk in accordance with Appendix M of Inspection Manual Chapter 609, the NRC determined that this finding was of very low safety significance.
Inspection Report# : 2007002 *(pdf)*

Significance: G Mar 31, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
UNQUALIFIED MAIN STEAM SAFETY RELIEF VALVE WELD REPAIR.
The inspectors identified a NCV of 10 CFR 50.55a(g)4 for failure to complete Code qualified weld repairs for the main steam safety relief valve PSV-4401. Specifically, the weld procedures for this repair were not qualified by performing tensile and guided bend tests intended to demonstrate that the weld procedure produced welds with satisfactory strength and ductility for the intended service. Without these tests, the inspectors were concerned that these non-Code conforming weld repairs affecting the pressure boundary (valve body) could lead to cracking and failure of PSV-4401 valve body or bellows when this valve was placed in service. The licensee determined that this issue affected the structural integrity of the safety relief valve (SRV) pilot bellows and could cause the SRVs to not operate in an overpressure condition and declared all of the relief valves inoperable and entered this issue into the corrective action program. The cause of this finding was related to the work practices component of the human performance cross-cutting area because the licensee failed to ensure adequate oversight of vendor activities with respect to review of the vendor's weld procedures for repair of reactor coolant pressure boundary retaining components (PSV-4401). Specifically, during review of vendor procedures 889C W-6d and 889C W-1, the licensee did not demonstrate adequate understanding of Code requirements and/or did not sufficiently challenge the vendor's basis for not performing weld procedure qualification tests.

This finding was of more than minor significance because the finding could be reasonably viewed as a precursor to a significant event involving the failure of repair welds from weld flaws introduced by use of an unqualified welding process. In addition, the finding was associated with the Initiating Events cornerstone attribute of "Equipment Performance," and affected the cornerstone objective to limit the likelihood of those events that upset plant stability, and challenge critical safety functions during shutdown as well as power operations. Absent NRC intervention, the licensee would have relied on unqualified weld repairs on PSV-4401 for an indefinite period of service, which may have placed the reactor coolant pressure boundary at increased risk for weld failure resulting in leakage, or an inoperable relief valve. The NRC evaluated this finding in accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for the At-Power Situation," and because this issue was identified prior to repressurizing the plant, determined that this finding was of very low safety significance.
Inspection Report# : 2007002 *(pdf)*

Significance: G Mar 31, 2007
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
UNPLANNED REACTOR PROTECTION SYSTEM AUTOMATIC SCRAM DUE TO INADEQUATE PROCEDURE.
A finding of very low safety significance and an associated NCV of 10 CFR 50, Appendix B, Criterion 5, was self-revealed when an unplanned RPS reactor scram occurred during surveillance testing due to a scram discharge volume (SDV) high level. On March 2, 2007, with the reactor shutdown for a planned refuel outage, operators were performing surveillance testing to verify the backup scram valves port air when a scram occurs. After inserting a manual scram and verifying that the backup scram valves ported air, the operators reset the scram. A short time later an unanticipated automatic scram was inserted due to a SDV high level. The operators bypassed the SDV high level scram and reset the scram. Corrective Action Process document (CAP) 048038 was entered into the licensees corrective action program to document the automatic scram.

This issue was more than minor because it directly affects the Initiating Events Cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown operations. Specifically, the Human Performance attribute as well as the configuration control attribute for controlling the shutdown equipment lineup. The NRC evaluated this finding in accordance with IMC 0609, Appendix G, “Shutdown Operation Significance Determination Process,” and the finding was determined to be of very low safety significance because it did not impact any of the 5 shutdown safety functions identified. The inspectors also determined that the cause of this finding was related to the work practices component of the human performance cross-cutting area because operations personnel failed to communicate human error prevention techniques, such as holding pre-job briefings, self and peer checking, and proper documentation of activities during performance of the surveillance testing.
Inspection Report# : 2007002 *(pdf)*

Mitigating Systems

Significance: G Oct 19, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
DROP LOAD EVALUATION FOR STUD TENSIONER.
The inspectors identified an NCV of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” that was of very low safety significance for the failure to translate the design bases into procedures and instructions. Specifically, the lift height limit assumed in the drop load analysis for transporting reactor vessel head stud tensioners over the refueling floor was not translated into the lift procedure allowing the licensee to potentially exceed the lift height established in the design basis calculation. This issue was entered into the licensee’s corrective action program.

The issue was more than minor, because the failure to provide procedural controls for lifting of the reactor head tensioner could become a more significant safety concern. Specifically, a load drop from a higher elevation could

have led to slab failure and potential damage to safe shutdown and safety related equipment on the floors below. This finding was of very low safety significance, because the inspectors answered “no” to all five questions under the Mitigating Systems Cornerstone column of the Phase 1 worksheet. Specifically, even though procedural controls were not in place to ensure that the reactor head tensioner would not be lifted above 6 feet, it could not be determined whether the head had actually ever been lifted above that threshold.
Inspection Report# : 2007007 *(pdf)*

Significance: G Oct 19, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO ACCOUNT FOR DELAYS IN ECCS MOVS DUE TO VOLTAGE DIPS DURING LOAD SEQUENCING.
The inspectors identified an NCV of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” that was of very low safety significance. Specifically, MOV stroke time delays which result from Emergency Diesel Generator (EDG) voltage drops during load sequencing were not accounted for in assumed Emergency Core Cooling System (ECCS) required Motor Operated Valve (MOV) stroke times. This issue was entered into the licensee’s corrective action program.

The issue was more than minor because it was associated with the Mitigating System Cornerstone attribute of “Design Control,” and affected the cornerstone objective of ensuring the capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the MOV delays caused by voltage dips during ECCS load sequencing were not accounted for in the licensee’s design basis and resulted in a substantive margin reduction (up to 5.3 seconds) in the ECCS injection response time. This finding was of very low safety significance, because the inspectors answered “no” to all five questions under the Mitigating Systems Cornerstone column of the Phase 1 worksheet. Specifically, even though the MOV delays were substantial and resulted in a large margin reduction, a comparison of current In-Service Testing (IST) times verses design basis maximum stroke times revealed that adequate margin still existed to meet the required ECCS response times.

Inspection Report# : 2007007 *(pdf)*

Significance: G Oct 19, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO INITIATE A CORRECTIVE ACTION DOCUMENT FOR DEGRADING CABLING.
The inspections identified an NCV of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” that was of very low safety significance. Specifically, the licensee found safety related cable 1S0104-E to be severely degraded due to heat related aging and failed to initiate a corrective action document to evaluate the condition and perform an extent of condition in accordance with plant procedures. This issue was entered into the licensee’s corrective action program.

The issue was more than minor because the failure to identify safety related cable failures and perform a proper extent of condition could lead to more significant safety conditions. Specifically, cables failures are adverse conditions that are primarily caused by heat induced aging. If a heat source exists, it is highly probable that other cables are adversely affected. By not writing a corrective action document and performing an extent of condition to replace damaged cables, those cables would instead fail potentially causing plant transients or a loss of mitigating equipment. This finding was of very low safety significance, because the inspectors answered “no” to all five questions under the Mitigating Systems Cornerstone column of the Phase 1 worksheet. Specifically, the cable that was degraded was replaced during the last outage and no additional cables have yet failed in the proximity of the original failed cable. The primary cause of this finding was related to the cross-cutting area of problem identification and resolution because the licensee did not properly identify the cracked and brittle cabling through their corrective action program. (P.1.a.)

Inspection Report# : 2007007 *(pdf)*

Significance: G Sep 30, 2007

Identified By: Self-Revealing
Item Type: NCV NonCited Violation
FAILURE TO PROPERLY USE ERROR PREVENTION TOOLS RESULTS IN A LOSS OF SAFETY FUNCTION BY INSTALLING A RELAY JUMPER ON THE INCORRECT RELAY.
A finding of very low safety significance and an associated NCV of TS 5.4.1a, associated with Regulatory Guide 1.33, Revision 2, Appendix A, Section 8 were identified through a self-revealing event when the licensee failed to properly implement procedures for configuration control during planned surveillance activities which resulted in the loss of a required safety feature. Specifically, during performance of STP 3.3.6.1-02, "Main Steam Line Low Pressure Instrument Calibration," maintenance personnel incorrectly installed a relay jumper on the relay for PS1014, Primary Containment Isolation System (PCIS) channel A1 instrument, and subsequently isolated PS1016, PCIS channel A2 instrument, for the calibration check. During the period of time that the jumper was installed on the channel A1 instrument and the channel A2 instrument was isolated and pressurized, a Group 1 isolation would not have occurred if an actual main steam line low pressure condition had occurred. The primary cause of this violation was related to the cross-cutting area of human performance. Specifically, personnel work practices failed to use human performance prevention tools, commensurate with the risk of the task being performed, to ensure work activities are performed safely. The failure to use proper concurrent verification and place keeping techniques resulted in the test jumper being installed on the relay for a previously tested channel instead of the relay for the pressure instrument which was being tested. (H.4.a)

This finding was more than minor because it is associated with the Mitigating Systems cornerstone attribute of equipment performance, and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. (IMC 0612 Appendix B, "Issue Screening.") The inspectors performed a Phase 1 analysis of this finding in accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." Since this issue was not a design or qualification deficiency, involved the loss of a safety feature and did not result in a loss of safety function, and was not considered potentially risk significant to a seismic, flooding, or severe weather initiating event, the issue was of very low safety significance.
Inspection Report# : 2007004 *(pdf)*

Significance: G Sep 30, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Inadequate Design Calculations
Green. The inspectors identified a finding having very low safety significance and an associated non cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." Specifically, the licensee failed to verify the adequacy of design calculations performed to verify the acceptability of a steam void in the High Pressure Coolant Injection (HPCI) pump discharge piping. Following discovery, the licensee performed informal analyses to show that the HPCI system remained operable. The primary cause of this violation was related to the cross-cutting area of human performance. Specifically, the licensee failed to use conservative assumptions in decision making and appeared to adopt a requirement to demonstrate that continued presence of a steam void was acceptable rather than to analyze the effects of a steam void of the size and under the conditions which the licensee originally determined existed. (H.1.b)

This issue was more than minor because it fit the more than minor example from Appendix E, "Examples of Minor Issues," example 3j, in that the licensee had to perform additional informal analyses to demonstrate the acceptability of the formal calculations and to show that the HPCI system remained operable. This performance deficiency impacted the Mitigating Systems Cornerstone objective of ensuring the operability and reliability of the HPCI system because it affected the design control attribute of structural integrity.

The issue was of very low safety significance based on a Phase I analysis performed in accordance with IMC 0609, "Significance Determination of Reactor Inspection Findings for At-Power Situations," Appendix A.
Inspection Report# : 2007004 *(pdf)*

Significance: G Jun 30, 2007
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
LIFTING OF HPCI DISCHARGE RELIEF VALVE DURING PLANNED SURVEILLANCE TESTING.

A finding of very low safety significance and an associated Non-Cited Violation (NCV) of 10 CFR 50 Appendix B, Criterion 3, was self-revealed when PSV2302, the HPCI discharge pressure relief valve, stuck open during planned testing of the High Pressure Coolant Injection (HPCI) System. The inspectors determined that the failure to provide sufficient margin between the HPCI discharge relief valve setpoint and the peak discharge pressure of the HPCI system upon startup was a performance deficiency warranting further evaluation. The licensee completed a temporary modification to remove the HPCI keep-fill modification and the HPCI system was returned to operable status.

The finding was determined to be more than minor because the engineering calculation error resulted in a condition where there was a reasonable doubt on the operability of the HPCI system. This issue screened as having a very low safety significance since the finding is a design deficiency confirmed not to result in a loss of operability per the part 9900 technical guidance for operability determination process for operability and functional assessment. This issue was also related to the decision making component of the human performance cross-cutting area, because engineering personnel failed to conduct an effective review of the safety-significant HPCI keep-fill modification and identify that the relief valve setpoint did not provide sufficient margin to prevent an unintended consequence. Specifically, the lifting of the relief valve due to the peak HPCI system discharge pressure seen during system startup.
Inspection Report# : 2007003 *(pdf)*

Significance: G Jun 30, 2007
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
TS ALLOWED OUTAGE TIME EXCEEDED FOR INOPERABLE EDGs.
A finding of very low safety significance and an associated NCV of Technical Specification (TS) 3.8.1.b, Electrical Power Systems, AC Sources-Operating, was self-revealed when a leak was discovered coming from the lube oil filter (LOF) cover on the ‘B’ Emergency Diesel Generator (EDG) during surveillance testing. The leak rate was approximately 0.21 gallons per minute, and the licensee determined that the ‘B’ EDG would not have been capable of performing its 7-day unassisted operation design requirement. The licensee declared the ‘B’ EDG inoperable, entered the issue into their corrective action program, and initiated a work order to repair the oil leak. During the licensee’s investigation, the apparent cause of the LOF leak was the installation of the wrong oil filter cover o-ring while performing the liner replacement maintenance during the recent refueling outage conducted by the vendor six weeks prior.

The finding was determined to be more than minor because the ‘B’ EDG was returned to service with the incorrect o-ring installed and the leak that developed resulted in subsequent equipment inoperability. Additionally, based upon the licensee’s past operability evaluation, the TS limiting condition for operation (LCO) allowed outage time for one EDG inoperable with the plant at power was exceeded. Since this issue was not a design or qualification deficiency, did not result in a loss of safety function, and was not considered potentially risk significant to a seismic, flooding, or severe weather initiating event, the issue screened as having a very low safety significance. This issue was also related to the work practices component of the human performance cross-cutting area, because maintenance personnel failed to ensure supervisory and management oversight of work activities, including contractors, supported nuclear safety. Specifically, the personnel performing maintenance activities for reassembly of the LOF were not supervised, an incorrect LOF cover 0-ring was installed, and the equipment was subsequently returned to service.
Inspection Report# : 2007003 *(pdf)*

Significance: SL-IV Jun 30, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO PROVIDE COMPLETE AND ACCURATE INFORMATION TO THE NRC ON NRC FORM 396.
The inspectors identified a Level IV NCV of 10 CFR 50.9, “Completeness and Accuracy of Information.” The inspectors identified that the facility licensee, on March 30, 2007, submitted to the NRC, an NRC Form 396, “Certification of Medical Examination By Facility Licensee,” for a licensed operator applying for renewal of his reactor operator license, that was not complete and accurate in all material respects. Specifically, the NRC Form 396 certified that the licensed operator was not required to have a “corrective lens” restriction on his license. When the NRC questioned the licensee on the accuracy of the most recent biennial medical examination on the submitted NRC Form 396, the licensee submitted a revised NRC Form 396 on April 19, 2007. The revised NRC Form 396 included a new date for the most recent biennial medical examination, but also showed that the licensed individual was required to have a “corrective lens” restriction added to his license. This information was material to the NRC because the

NRC relies on this certification to determine whether an applicant meets the requirements to operate the controls of a nuclear power plant pursuant to 10 CFR Part 55.

The finding was determined to be more than minor because the information associated with the license renewal of the individual was provided to the NRC under a signed statement by the Site Vice President and could have impacted an NRC licensing decision. The licensed operator could have been, without NRC intervention, issued a license without a "corrective lens" restriction added to his license. The finding was determined to be of low safety significance because the license renewal application for the reactor operator was not renewed until complete and accurate information was received on a revised NRC Form 396 that showed a "corrective lens" restriction for the licensed individual.
Inspection Report# : 2007003 *(pdf)*

Significance: Mar 31, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO TAKE PROMPT CORRECTIVE ACTION TO CORRECT AN OPERABLE BUT NONCONFORMING CONDITION ON THE EMERGENCY DIESEL GENERATORS.
A finding of very low safety significance and an associated NCV of 10 CFR 50, Appendix B, Criterion 16, was identified by the inspectors for failure to take prompt corrective action to repair an operable but nonconforming condition on the 'A' and 'B' Emergency Diesel Generators (EDGs). On December 7, 2005, engineering personnel identified that during testing to simulate a Loss of Offsite Power concurrent with a Loss of Coolant Accident (LOOP/LOCA), the output voltage of the EDGs momentarily dropped below 75 percent of nominal voltage during the loading sequence of the EDG. The Updated Final Safety Analysis Report (UFSAR) states that the output voltage of the EDG shall not drop below 75 percent of nominal with the exception of the initial loading. The licensee failed to correct the nonconforming condition on the EDGs during the first available opportunity, which was the refueling outage that occurred in the first quarter of 2007. The failure to correct the nonconforming condition was entered into the licensee's corrective action program as CAP 047955.

This issue is more than minor because it directly impacts the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The NRC evaluated this finding using IMC 0609 Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations." The finding was determined to be of very low safety significance since the finding is a design deficiency confirmed not to result in a loss of operability per part 9900 technical guidance for the operability determination process for operability and functional assessment.
Inspection Report# : 2007002 *(pdf)*

Significance: Feb 11, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Recorded Pressure above Design Limits Not Entered into Corrective Action System
The inspectors identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action" having very low safety significance (Green). Specifically, the licensee failed to identify and correct a condition adverse to quality regarding a pressure significantly over the design value recorded on a high pressure coolant injection system vent line during a surveillance test on February 11, 2006 until prompted by the NRC. As corrective actions, the licensee performed calculations to assess the issue. The primary cause of this violation was related to the cross-cutting area of human performance because the licensee failed to use a systematic process when faced with an unexpected plant condition during a special test.

This issue was more than minor because the finding was associated with the Mitigating Systems cornerstone attribute of equipment performance and affected the cornerstone objective of ensuring the reliability of systems that respond to initiating events. Specifically, the pressure pulse exceeded the design pressure rating of the piping. Without evaluation, the licensee could not ensure the availability and reliability of the over-stressed vent piping to withstand normal operation. The issue was of very low safety significance based on a Phase 1 screening in accordance with IMC 0609, Appendix A.
Inspection Report# : 2006008 *(pdf)*

Significance: G Feb 11, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Non-conservative Analysis Methodology
The inspectors identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance (Green). Specifically, the licensee's calculation to show that the existing feedwater piping system configuration met the acceptance criteria of ASME Boiler and Pressure Vessel Code, Section III, Appendix F used a method of analysis that did not evaluate the dynamic effect of impact forces as specified by the design basis piping code, ANSI B31.1, "Power Piping." As corrective actions, the licensee performed calculations to assess the issue. The primary cause of this violation was related to the cross-cutting area of human performance because the licensee did not have adequate guidance on how to evaluate the dynamic effect of impact for variable spring hanger determined to exceed their available seismic travel.

The inspectors concluded that the finding was greater than minor because it affected the Mitigating Systems Cornerstone attribute of design control, and if left uncorrected, the finding could become a more significant safety concern. Specifically, the failure to evaluate the dynamic effect of impact as required by the ANSI B31.1 design basis code in similar operability calculations could result in exceeding the ASME Section III, Appendix F acceptance limits used to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The issue was of very low safety significance based on a Phase 1 screening in accordance with IMC 0609, Appendix A.
Inspection Report# : 2006008 *(pdf)*

Significance: G Jun 07, 2005
Identified By: NRC
Item Type: VIO Violation
Failure to Demonstrate Adequacy of Design Assumption for Torus Attached Piping
A violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control" having very low safety significance was identified by the inspector. Specifically, the licensee failed to demonstrate that a 1996 high pressure coolant injection (HPCI) modification was subjected to design control measures commensurate with those applied to the original design. The licensee also failed to apply design control measures to verify the adequacy of the design in order to assure that the design basis for torus attached piping was correctly translated into the modification's specifications, drawings, procedures and instructions.

The finding was more than minor because the finding was associated with the cornerstone attribute of design control in the mitigating system cornerstone and the finding was determined to affect the associated cornerstone objective of ensuring the availability of the HPCI system when called upon. Under the worst case scenario, movement of the torus with the additional valve weight on the HPCI turbine exhaust line would result in crimping of the line. Crimping of the line would create additional backpressure in the HPCI turbine and would result in a decrease in the amount of water being injected into the reactor vessel. The finding was determined to be of very low safety significance based upon a Phase 2 analysis of those transients which would involve movement of the torus.

The finding was cited since the licensee did not enter the issue into its corrective action program and did not take actions to correct the noncompliance.
Inspection Report# : 2005010 *(pdf)*

Barrier Integrity

Significance: G Jun 30, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO IMPLEMENT THE APPROPRIATE PROCEDURAL CONTROLS PRIOR TO USING NYLON ROPE TO SECURE UNDERWATER LIGHTS IN THE SPENT FUEL POOL.

The inspectors identified a finding of very low safety significance and an associated NCV of 10 CFR 50 Appendix B, Criterion 5, when licensee staff failed to implement the appropriate controls to properly store underwater lights in the spent fuel pool. The licensee entered the issue into the corrective action program for resolution. This issue was also related to the work practices component of the human performance cross-cutting area. Specifically, the aspect related to procedural compliance, as the station procedure that described the appropriate controls for storing items in the pool, was not followed.

The finding was determined to be more than minor because the finding could be reasonably viewed as a precursor to a more significant event. Specifically, the failure to follow the approved process for controlling the use of nylon ropes in the spent fuel pool could result in the ropes being in place for an extended period of time. This increased the potential for unplanned radiation exposure either due to wicking or from damage to the underlying fuel assemblies, if the ropes degraded causing the lights to fall. The finding was considered to be of very low safety significance since it was determined to affect only the fuel cladding function of the Barrier Cornerstone.
Inspection Report# : 2007003 *(pdf)*

Significance: G Mar 31, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO INCLUDE ACCEPTANCE CRITERIA IN TROUBLESHOOTING INSTRUCTION FORM FOR CONTROL BUILDING STANDBY FILTER UNIT TESTING.
A finding of very low safety significance and an associated NCV of 10 CFR 50, Appendix B, Criterion 5, was identified by the inspectors when engineering and operations personnel failed to include acceptance criteria in a Troubleshooting Instruction Form (TIF). On February 12, 2007, engineering and operations personnel completed a TIF to determine the effects upon the control building envelope of open penetrations between the cable spreading room and the turbine building. The TIF failed to include acceptance criteria to identify whether the Standby Filter Units (SFUs) were being left in an operable condition at the completion of the troubleshooting activity. At the completion of the TIF, operations personnel failed to immediately identify that the as-left control building static pressure was less that the Technical Specification (TS) required limit of > 0.1 inches water gauge relative to the outside atmosphere. When the Shift Manager later identified that the TS requirement was not met, core alterations and fuel moves were secured and the issue was entered into the licensee's corrective action program as CAP 047315.

This issue was more than minor because it directly impacts the barrier integrity cornerstone objective to provide reasonable assurance that physical barriers (containment) protect the public from radio nuclide release caused by accident and events. The NRC evaluated this finding in accordance with IMC 0609, Appendix G, "Shutdown Operation Significance Determination Process," and the finding was determined to be of very low safety significance because it did not require a phase 2 quantitative assessment. The inspectors also determined that the cause of this finding was related to the work control component of the human performance cross-cutting area because engineering and operations personnel failed to appropriately coordinate work activities by communicating, coordinating, and cooperating with each other during activities in which interdepartmental coordination is necessary to assure plant and human performance.
Inspection Report# : 2007002 *(pdf)*

Emergency Preparedness

Occupational Radiation Safety

Significance: G Mar 31, 2007
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
WORKERS INAPPROPRIATELY ENTERED A HIGH RADIATION AREA.
A self-revealed finding of very low safety significance and an associated NCV of TS 5.7.1 were identified for the

failure to satisfy TS requirements for worker access into a high radiation area with dose rates in accessible areas between 100 and 1000 mrem/hour at 30 centimeters. Workers entered the reactor building 716' Northwest Corner Room (NWCR) which was posted as a high radiation area (HRA), without adequate recognition of the area radiological conditions and without positive radiological control over the activities within the area. The electronic dosimetry (ED) worn by one of the workers alarmed when significantly higher than expected dose rates were encountered.

The issue was more than minor because it was associated with the Program/Process attribute of the Occupational Radiation Safety Cornerstone and affected the cornerstone objective to ensure adequate protection of worker health and safety from exposure to radiation. The issue represents a finding of very low safety significance because it did not involve As-Low-As-Is-Reasonably-Achievable (ALARA) planning or work controls, there was no overexposure, nor did a substantial potential for an overexposure exist given the radiological conditions in the area and the workers response to the ED alarm. Also, the licensee's ability to assess worker dose was not compromised. Corrective actions taken by the licensee included reminding radiation protection staff to better coordinate entries into these areas with operations staff, and plans to reevaluate the radiation protection department practices for entry into high radiation areas, and in general for entry into high radiation areas with the potential for significant dose rate gradients. A cross-cutting aspect in the area of human performance was associated with this finding in the work practices component.
Inspection Report# : 2007002 *(pdf)*

Public Radiation Safety

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