Duane Arnold
1Q/2008 Plant Inspection Findings

Initiating Events

Significance: G Mar 31, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO PROPERLY ADMINISTER AND DOCUMENT OVERTIME LIMITS, REQUIREMENTS, AND DEVIATIONS.
A finding of very low safety significance, and an associated NCV of Technical Specification (TS) 5.2.2.e, Administrative Controls, Organization, Unit Staff, was identified by the inspectors for the licensee's failure to follow the Administrative Control Procedure (ACP) 101.4, "Overtime Limits and Requirements," Section 3.1 (3) requirement for assuring that personnel do not exceed the overtime requirements without prior authorization. The licensee entered this issue into its corrective action program to evaluate the adequacy and effectiveness of the ACP, and to identify required procedure revisions to prevent recurrence.

Using the minor questions in Appendix B of IMC 0609, the inspectors determined that the finding was more than minor because the issue was associated with the Initiating Events Cornerstone attribute of Human Performance and affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the unrecognized periods of excessive work hours would increase the likelihood of human errors during refueling outage activities and response to plant events, i.e. fire watches. Since these periods were unrecognized and not authorized in advance by management, the excessive work hours could result in more significant safety concerns. Because this issue occurred during the last refueling outage, the finding was evaluated in accordance with IMC 0609, Appendix G, "Shutdown Operations SDP." Using Checklist 7, "Boiling Water Reactor (BWR) Refueling Operation with RCS Level > 23'", contained in Attachment 1, the inspectors determined that since the plant had appropriately met the safety function guidelines for core heat removal, inventory control, power availability, containment integrity, and reactivity control, and since a phase 2 or phase 3 analysis was not required, the finding screened as Green using Figure 1.

This finding has a cross-cutting aspect in the area of Human Performance for the Resources safety culture component because the licensee did not ensure that sufficient trained personnel and procedures were available and adequate to assure nuclear safety by maintaining work hours within working hour limits.

Inspection Report# : 2008002 *(pdf)*

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)*

Mitigating Systems

Significance: G Mar 14, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
Inadequate Calculations/Analyses for Essential 4160 VAC Circuit Breaker Close/Open Coils (1R21.3.b.(1))
A finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” was identified by the inspectors for the failure to assure and verify that adequate control voltage was available for the close and open coils of the 4160VAC [volts alternating current] safety-related breakers. The licensee entered this performance deficiency into its corrective action program, performed a simplified evaluation to determine the worst case available close coil voltage at the worst case breaker fed from the 4160VAC essential switchgear, and conducted testing to demonstrate a reasonable assurance of operability. The finding was determined to be more than minor because the failure to assure and verify that adequate control voltage was available to close and open the 4160VAC breakers could have affected the capability of emergency diesel generators and other safety-related equipment to respond to initiating events. The issue was of very low safety significance because the inspectors determined it was a design deficiency that did not result in actual loss of safety function. The inspectors determined

there was no cross-cutting aspect associated with this finding. (Section 1R21.3.b.(1))

Inspection Report# : 2008006 *(pdf)*

Significance: G Mar 14, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
Inadequate Calculations/Analyses and Testing for TOLs on Safety-Related MOVs (1R21.3.b.(2))
A finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” was identified by the inspectors for the failure to assure that thermal overload relays (TOLs) on safety-related motor-operated valve (MOV) circuits were sized properly and periodically tested. The licensee entered this issue into its corrective action program and was able to demonstrate operability in that the TOLs would not prevent any MOVs from performing their safety function. The finding was determined to be more than minor because the failure to assure that TOLs were properly sized and periodically tested could have affected the ability for MOVs to respond to initiating events. The issue was of very low safety significance because the inspectors determined it was a design deficiency that did not result in actual loss of safety function. The inspectors determined there was no cross-cutting aspect associated with this finding. (Section 1R21.3.b.(2))
Inspection Report# : 2008006 *(pdf)*

Significance: G Mar 14, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Periodically Test Reactor Protection System Key Lock Bypass Switches (1R21.6.b.(1))
A finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50, Appendix B, Criterion XI, “Test Control,” was identified by the inspectors for the failure to test reactor protection system (RPS) key locked bypass switches. The licensee entered this issue into its corrective action program and initiated procedural changes to require periodic testing of the RPS bypass switches. This finding was more than minor because the licensee did not ensure the operability and functional performance of the key lock switches used bypass automatic protection circuits in the RPS. The issue was of very low safety significance because the inspectors determined that it did not result in actual loss of safety function. The inspectors determined there was no cross-cutting aspect associated with this finding. (Section 1R21.6.b.(1))

Inspection Report# : 2008006 *(pdf)*

Significance: SL-IV Dec 31, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO MAKE AN 8 HOUR NOTIFICATION TO THE NRC FOR LOSS OF BOTH EDGS.
A finding of very low safety significance and an associated Severity Level IV NCV of 10 CFR 50.72(b)(3)(v) were identified by the inspectors for the failure of the licensee to make an eight-hour notification to the NRC for the loss of both emergency diesel generators (EDGs). The licensee entered this into their corrective action program (CAP) as CAP 053463 and updated the event notification (EN 43692) to include the loss of safety function resulting from both EDGs being inoperable from 0408 to 0715 on October 5, 2007.

The inspectors determined that the failure to report the loss of safety function of the onsite emergency AC power system in accordance with 10 CFR 50.72(b)(3)(v) was a performance deficiency. The NRC considers the safety implications of non-compliances that may impact the ability to carry out its statutory mission. Non-compliances may be significant because they may challenge the regulatory envelope upon which certain activities were licensed. This issue is greater than minor, because the failure to report the loss of the EDGs affected the NRC's ability to perform a regulatory function. Because violations of 10 CFR 50.72 are considered to be violations that potentially impede or impact the regulatory process, they are dispositioned using the traditional enforcement process instead of the SDP. However, if possible, the underlying technical issue is evaluated under the SDP to determine the severity of the violation. Using IMC 0609, "Significance Determination Process," the inspectors screened this issue as having very low safety significance using the phase 1 screening questions under the Mitigation System Cornerstone.

Inspection Report# : 2007005 *(pdf)*

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: 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: 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)*

Emergency Preparedness

Occupational Radiation Safety

Significance: G Dec 31, 2007
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
FAILURE TO ADEQUATELY SURVEY RESULTING IN UNPOSTED, UNCONTROLLED HIGH RADIATION AREA.
A self-revealed finding of very low safety significance and an associated NonCited Violation (NCV) of Title 10 CFR 20.1501 were identified for failure to adequately survey and evaluate the magnitude and extent of radiation levels to ensure that high radiation areas are adequately posted and controlled. On February 7, 2007, a worker entered into an inadequately posted and controlled area in the Reactor Building 734' West Torus Room, which had radiation levels warranting posting and controls for a high radiation area. Corrective actions taken by the licensee included a change in the procedure to survey the Torus Room in the area of shut-down cooling, specifically after low pressure core injection (LPCI) full flow testing that could result in unexpected high radiation areas. A cross-cutting aspect in human performance was associated with this finding in the area of decision-making. (H.1.a)

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 electronic dosimeter alarm. Also, the licensee=s ability to assess worker dose was not compromised.

Inspection Report# : 2007005 *(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 : June 05, 2008