Duane Arnold
1Q/2007 Plant Inspection Findings

Initiating Events

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 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: G 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 Dec 31, 2006
Identified By: NRC
Item Type: FIN Finding
CREW PERFORMANCE ON THE DYNAMIC SCENARIO PORTION OF THE 2006 FACILITY-ADMINISTERED ANNUAL REQUALIFICATION EXAMINATION OPERATING TEST.
A finding of very low safety significance was identified. The finding was associated with unsatisfactory operating crew

performance on the simulator during facility-administered licensed annual operator requalification examinations. Of the ten crews evaluated, two did not pass their annual operating tests. The finding is of very low safety significance because the failures occurred during testing of the operators on the simulator, because there were no actual consequences to the failures, and because the crews were removed from watch-standing duties, retrained, and re-evaluated before they were authorized to return to control room watches. This issue was documented in the licensee's corrective action process (CAP) as CAP 044379.

Inspection Report# : 2006005 *(pdf)*

Significance: G Jun 30, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO PERFORM AN ADEQUATE RISK ASSESSMENT.
A finding of very low safety significance was identified by the inspectors for the licensee's failure to conduct an adequate risk assessment of the Standby Liquid Control System (SLCS) which was removed from service for scheduled surveillances December 1, 2005, and March 1, 2006. This resulted in an unrecognized increase in the level of risk as determined by the licensee's Probabilistic Risk Assessment (PRA) model. This issue was documented in the licensee's corrective action program (CAP) as CAP 042499. The corrective actions taken included revising the procedure to insert detailed restoration steps, communications and dedicated operator requirements, as well as requirements for declaring the system inoperable and unavailable during performance of the surveillance test. An NCV of 10 CFR 50.65(a)(4) was identified for the failure to conduct an adequate risk assessment prior to conducting online maintenance involving the SLCS.

This finding is more than minor because it is associated with the Mitigating Systems Cornerstone attribute of equipment performance and adversely affected the cornerstone objective in that the licensee failed to perform an adequate risk assessment prior to conducting online maintenance. The licensee's risk assessment did not consider the risk-significant SLCS system that was out of service which, when properly evaluated, did result in an increased level of risk from a PRA perspective and would have put the licensee in a higher risk category. However, the finding was of very low safety significance because the risk deficit for Incremental Core Damage Probability was less than 1E-6 and for Incremental Large Early Release Probability was less than 1E-7.
Inspection Report# : 2006003 *(pdf)*

Significance: G Apr 21, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
Calculation Deficiency for Potential Vortexing in CST

The team identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance involving the calculation of low level setpoints for the CST. Specifically, the licensee did not include a quantitative analysis of the transfer time in the calculation and subsequently, did not fully address the potential for air entrainment in the high pressure injection pump due to vortexing. The licensee determined the high pressure injection system was operable based on available margin in the calculation. The licensee entered the finding into their corrective action program as CAP 040973.

The finding was more than minor because the failure to account for this transfer time reduced the margin available to prevent air entrainment into the high pressure coolant injection (HPCI) system and affected the Mitigating Systems cornerstone attribute of design control. The finding was of very low safety significance based on the results of the licensee's analysis and screened as Green using the SDP Phase 1 screening worksheet. The cause of the finding was related to the cross-cutting element of problem identification and resolution.

Inspection Report# : 2006007 *(pdf)*

Significance: G Apr 21, 2006
Identified By: NRC
Item Type: NCV NonCited Violation

Inadequate Torquing of 250Vdc, 125Vdc and 48Vdc Batteries Electrical Connections

The team identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” having very low safety significance involving licensee’s failure to ensure that the torque values specified in the maintenance procedure for safety related and important to safety 250Vdc, 125Vdc, and 48Vdc batteries, were correctly incorporated from vendor specified design data and from the licensee’s design standard into the procedure. Consequently, all 250Vdc, 125Vdc, and 48Vdc battery electrical terminal connections were under-torqued during battery replacement activities, in 2003. The licensee’s corrective action included performing a condition evaluation to determine status of the batteries, and entering this performance deficiency into their corrective action program for resolution as CAP041156, CAP041422, and CAP 041734.

This finding was more than minor because the batteries procedure deficiency affected plant equipment and was associated with the attribute of design control and equipment performance of the Mitigating Systems cornerstone. Specifically, improper torquing could result in unacceptable battery terminal connection resistance and decreased battery capacity, rendering the dc system incapable of performing its required safety function. The finding was of very low safety significance based on the results of the licensee’s analysis and screened as Green using the SDP Phase 1 screening worksheet.

Inspection Report# : 2006007 *(pdf)*

Significance: G Apr 21, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
Electrical Components Downgraded from SR to NSR Without Appropriate Isolation Devices

The team identified a Non-Cited Violation of 10 CFR 50 Appendix B, Criterion III, “Design Control,” having very low safety significance for failure to ensure that proper design control was maintained. Specifically, the licensee failed to perform a comprehensive design review of a 1992 modification that had incorrectly downgraded the quality classification of two level indicating switches. As a result of this team’s inquiries, four additional examples of mis-classified equipment were identified. The licensee entered the finding into their corrective action program as CAP041107 and CAP041731.

The finding was more than minor because, without proper electrical isolation devices, failure of QL4 (non-safety) classified devices could cause a loss of QL1(safety related) classified equipment. This finding was of very low safety significance based on the results of the licensee’s analysis and screened as Green using the SDP Phase 1 screening worksheet. The cause of this finding was related to the cross-cutting aspect of problem identification and resolution, in that, the licensee did not fully evaluate the condition adverse to quality in 2004.
Inspection Report# : 2006007 *(pdf)*

Significance: G Apr 21, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
UFSAR Table 8.2-1 Had No Documented Basis

The team identified a Non-Cited Violation of 10 CFR 50, Appendix B,
Criterion III, Design Control, having a very low safety significance pertaining to lack of design basis for the values listed in Updated Final Safety Analysis Report Table 8.1-2. The licensee could not identify an active calculation that supported the values listed in the table. In response to this deficiency, the licensee initiated CAP 041395 to develop the basis for the values indicated in the UFSAR table.

The finding was more than minor because control relay settings and design voltage values could be incorrectly set based on these unsupported values. The finding was of very low safety significance based on the results of the licensee’s analysis and screened as Green using the SDP Phase 1 screening worksheet.
Inspection Report# : 2006007 *(pdf)*

Significance: G Apr 21, 2006

Identified By: NRC
Item Type: NCV NonCited Violation
Non-Safety Related Charger Used to Charge a Cell of a 125Vdc SR Battery Without Electrical Isolation

The team identified a Non-Cited Violation of TS 5.4.1a, “Procedures,” having a very low safety significance pertaining to licensee’s failure to establish and use an appropriate procedure for charging a single cell of a safety related battery. A portable non-safety related charger was used to charge a single cell of a safety related battery without maintaining the required electrical isolation between the safety related battery and the non-safety related charger. The licensee initiated CAP 041099 to modify existing maintenance procedures.

This finding was more than minor because failure to maintain electrical isolation could render the safety related battery inoperable. The finding was of very low safety significance based on the results of the licensee’s analysis and screened as Green using the SDP Phase 1 screening worksheet.
Inspection Report# : 2006007 *(pdf)*

Significance: G Apr 21, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Establish a Testing Program for Molded Case Circuit Breakers (MCCBs)

The team identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XI, “Test Control,” having very low safety significance for failure to implement a testing program to ensure that the installed safety related and important-to-safety molded case circuit breakers (MCCBs) will perform satisfactorily in service. This issue was entered into the licensee’s corrective action program as CAP041363. The licensee was planning to purchase new test equipment and commence testing a statistical sample of the installed MCCBs to corroborate MCCB operability.

The finding was more than minor because the installed MCCBs were not adequately exercised or tested and were beyond the manufacturer’s design life. This condition could effect breaker coordination, over-current protection, fire prevention, and multiple other safety related and important to safety functions. The finding was of very low safety significance because licensee determined the issue was a qualification deficiency confirmed not to result in loss of operability per “Part 9900, Technical Guidance, Operability Determination Process for Operability and Functional Assessment.” The cause of the finding was related to the cross-cutting aspect of problem identification and resolution.
Inspection Report# : 2006007 *(pdf)*

Significance: G Apr 21, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
Simulation of Operator response to an SBO event

The team identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” having very low safety significance for failing to maintain adequate procedures to establish alternate ventilation within a minimum time after the onset of a station blackout event. The licensee entered the finding into their corrective action program as CAP 041379 and commenced an extensive root cause investigation.

The finding was more than minor because failure to establish alternate ventilation within the analyzed time limit could result in excessive temperatures in the rooms and impact the performance of equipment. Although the use of an inadequate procedure increased the likelihood of undesirable consequences from an SBO event, the finding was 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 cause of the finding is related to the cross-cutting element of problem identification and resolution.
Inspection Report# : 2006007 *(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 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)*

Significance: SL-III May 01, 2006
Identified By: NRC
Item Type: VIO Violation
Failure to complete the “Pre Fuel Move Checklist” before moving irradiated fuel bundles in the DAEC spent fuel pool
Duane Arnold Energy Center (DAEC) Refueling Procedure No. 403, “Performance of Fuel Handling Activities,” Revision

No. 16, was issued on June 16, 2004, and required that the designated fuel handling supervisor complete applicable sections of the “Pre Fuel Move Checklist” before starting fuel handling activities. On November 9, 2004, a refueling floor supervisor, who was primarily responsible for preparing Revision No. 16 to the procedure, was the designated fuel handling supervisor and he failed to complete the “Pre Fuel Move Checklist” before moving three irradiated fuel bundles in the DAEC spent fuel pool, as required by the refueling procedure.

The NRC Office of Investigations (OI) conducted an investigation of the event involving a former supervisor’s apparent willful violation of a DAEC refueling procedure on November 9, 2004. The OI investigation was completed on October 26, 2005. (OI Case 3-2004-033).

Based on the information developed during the OI investigation and the information provided by the licensee in an April 7, 2006 letter, the NRC has determined that a SL III violation of NRC requirements occurred. The NRC has determined that this was a willful violation, demonstrating at least careless disregard of a procedure required by DAEC Technical Specification 5.4.1.
Inspection Report# : 2006016 *(pdf)*

Significance: G Apr 21, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
RCIC Pump Suction Valves Automatic Control Logic

The team identified a Non-Cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” having very low safety significance involving the control logic of reactor core isolation cooling (RCIC) pump suction valves MO-2516 and MO-2517. Design Change Request 1040 modified the control logic and did not retain the remote-manual closure capability of these containment isolation valves. This remote-manual closure capability was specifically addressed in NRC correspondence. As an interim measure, the licensee revised an operating procedure to allow the operators to manually block specific relay contacts in the control room, allowing these valves to be closed if required. The licensee entered the finding into their corrective action program as CAP 041114.

The finding was more than minor because failure to retain the remote-manual closure capability of these valves was associated with the attribute of design control, which affected the barrier integrity cornerstone objective of ensuring the functionality of the primary containment isolation valves. The finding was of very low safety significance based on the results of the licensee’s analysis and screened as Green using the SDP Phase 1 screening worksheet.

Inspection Report# : 2006007 *(pdf)*

Emergency Preparedness

Significance: W Dec 05, 2006
Identified By: NRC
Item Type: VIO Violation
Failure of the licensee’s exercise critique process
The inspectors identified a finding when the licensee’s critique presented to the NRC on October 19, 2006, failed to properly identify a weakness that was associated with a risk significant planning standard.

The finding was more than minor because it was associated with the Emergency Preparedness Cornerstone and affected the cornerstone objective of ensuring that the licensee was capable of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency. Specifically, the licensee’s critique did not identify a performance weakness associated with a failure to recognize an emergency action level entry condition. Additionally, the critique failed to adequately identify a weakness that demonstrated a level of performance that could preclude effective implementation of the Emergency Plan in an actual emergency. This finding potentially has low to moderate safety significance (White) because the licensee’s exercise critique failed to identify a weakness that is associated with a RSPS.

After considering the information developed during the inspection, the information presented at the Regulatory Conference on March 1, 2007, and the additional information you provided in your apparent cause evaluation dated December 21, 2006, the NRC has concluded that the inspection finding was appropriately characterized as White, an issue with low to moderate increased importance to safety, which may require additional NRC inspections.
Inspection Report# : 2006009 *(pdf)*
Inspection Report# : 2007501 *(pdf)*

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)*

Significance: SL-III May 01, 2006
Identified By: NRC
Item Type: VIO Violation
Failure to notify health physics or ensure that health physics personnel were present prior to relocating irradiated items in the cask pool
Duane Arnold Energy Center (DAEC) Control Procedure ACP1407-2, "Material Control in the Spent Fuel Pool and Cask Pool," Revision 10, dated November 4, 2002, a procedure that implemented Technical Specification 5.4.1 and Regulatory Guide 1.33, provided, in part, that health physics shall be notified and present prior to relocating or removing any item stored in the spent fuel pool and cask pool. On July 23, 2003, a Refueling Floor Supervisor directed an operator to relocate irradiated items in the cask pool without notifying health physics or ensuring that health physics personnel were present prior to relocating the irradiated items.

The NRC Office of Investigations (OI) conducted an investigation of the event involving a former supervisor's apparent deliberate violation of a radiation protection procedure on July 23, 2003. The OI investigation was completed on February 6, 2004 (OI Report No. 3-2003-021).

Based on information developed during the OI Investigation, information provided during the June 1, 2004, PEC, and all other pertinent information, the NRC determined that a SL III violation of NRC requirements occurred at DAEC on July 23, 2003. The NRC has determined that this was a deliberate violation of NRC requirements.

Inspection Report# : 2006017 *(pdf)*

Public Radiation Safety

Physical Protection

Physical Protection information not publicly available.

Miscellaneous

Last modified : June 01, 2007