Dresden 3
1Q/2008 Plant Inspection Findings

Initiating Events

Significance: G Dec 31, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to perform corrective action to mitigate excessive train unavailability for Auxiliary Electric Equipment Room Heating, Ventilation and Air Conditioning
The inspectors identified a performance deficiency involving a non-cited violation of 10 CFR 50.65, “Requirements for monitoring the effectiveness of maintenance at nuclear power plants,” Section (a) (1), for the failure to take appropriate corrective action to mitigate excessive system unavailability of the Auxiliary Electric Equipment Room (AEER) ventilation system. The AEER heating, ventilation, and air conditioning (HVAC) system is designed to ensure adequate cooling to the AEER which contains several safety related systems, such as low voltage buses that provide power to mitigating systems. The Unit 2 and Unit 3 AEER ventilation systems have been classified 10 CFR 50.65 (a)(1) since the fourth quarter of 2005 with an established performance goal of no functional failures. Corrective actions to clean the condensing units before the onset of warm weather and cottonwood fuzz season were inappropriately rescheduled to mid-July. As a result, the AEER ventilation system air conditioning compressors tripped on high pressure on July 16 and 17, 2007, due to fouling of the condensing unit fins with cottonwood tree fuzz. The licensee’s corrective actions included coding the cleaning of the condensing unit as a summer readiness activity for future work control implementation.

This finding was more than minor in accordance with IMC 0612, Appendix E, example 7.a due to the licensee’s failure to take timely and effective corrective actions when goals were not met. The finding was considered to be of very low safety significance because the poor performance of this system had not resulted in an actual loss of a safety function of safety related equipment or resulted in an initiating event. The licensee would be able to reasonably perform controlled procedure steps to shutdown the reactor if AEER room temperature exceeded 104°F. The primary cause of this finding was related to the cross-cutting issue of Human Performance, Work Control. Specifically, the licensee did not plan work activities by incorporating environmental conditions which impacted plant structures, systems, and components. (H.3.(a))
Inspection Report# : 2007005 *(pdf)*

Mitigating Systems

Significance: G Mar 28, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
Acceptance Criteria for Stem Factor in MOV Testing Did Not Account for Uncertainty
The inspectors identified an NCV of 10 CFR Part 50, Appendix B, Criterion XI, “Test Control,” for failure to include an uncertainty value, to account for test equipment accuracy and lubricant degradation, in the acceptance criteria for the stem factor in the diagnostic test of MOV 3-2301-3, “Unit 3 high pressure coolant injection (HPCI) steam admission valve.” The stem factor is used to calculate the coefficient of friction (COF) to determine the predicted stroke opening time of the MOV under design basis conditions. Corrective actions for this issue included a re-calculation of the stem factor to account for instrument accuracy and lubricant degradation and an evaluation for guidance to be added to the test procedure.
This finding was more than minor because, if the finding was left uncorrected it would become a more significant safety concern. Specifically, the acceptance criteria specified in the diagnostic test for the stem factor did not assure that MOV 3-2301-3 would meet its design stroke time value to open in less than or equal to 30 seconds. The finding was of very low safety significance based on a Phase 1 screening in accordance with IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” because the finding did not

result in an actual loss of a safety function. (Section 4OA2.a)

Inspection Report# : 2008008 *(pdf)*

Significance: G Mar 28, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Develop a Pre-Fire Plan for Fire Zone 18.6
The inspectors identified an NCV for the licensee’s failure to develop a pre-fire plan for fire zone 18.6. The finding was a violation of Dresden Nuclear Power Station Renewed Operating License. License conditions 2.E and 3.G for Unit 2 and Unit 3, respectively, of the Dresden Nuclear Power Station Renewed Facility Operating Licenses state, in part, that: “The licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report (UFSAR) or the facility....” Pre-fire plans are described in the UFSAR as “provided for all safety-related areas of the plant.” Corrective actions by the licensee included the development of a pre-fire plan for fire zone 18.6.

The finding was more than minor because it involved the Mitigating Systems attribute of protection against external factors (i.e. fire), where the failure to develop a pre-fire plan for fire zone 18.6 could have adversely impacted the fire brigade’s ability to fight a fire. As such, this finding impacted the Mitigating Systems objective to ensure availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. As discussed by IMC 0609, Appendix A, Attachment 0609.04, issues related to performance of the fire brigade are not included in IMC 0609, Appendix F, “Fire Protection SDP,” and require management review. Therefore, the finding was reviewed by NRC management, and was determined to be a finding of very low safety significance (Green) because no safe shutdown equipment was located in this fire zone. The inspectors determined that this issue also affected the cross-cutting area of problem identification and resolution, CAP aspect P.1(c) because the licensee failed to thoroughly evaluate a problem previously identified by NCV 05000237/2006011-01; 05000249/2006011-01, “Licensee’s failure to develop a pre-fire plan for fire zone 8.2.6.A, elevation 534’,” such that the resolution did not fully address causes and extent of condition. (Section 4OA2.a)

Inspection Report# : 2008008 *(pdf)*

Significance: G Feb 15, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Perform Periodic Trip Tests on Thermal Overload Heaters.
Green. The inspectors identified a finding of very low safety significance and an associated NCV of 10 CFR Part 50, Appendix B, Criterion XI, “Test Control.” Specifically, the licensee failed to identify and periodically perform the necessary testing on safety-related thermal overload relays/heaters (TOLs), installed in 1993, in the alternate power feed to isolation condenser reactor inlet valves 2-1301-4 (Unit 2) and 3-1301-4 (Unit 3). Periodic testing of the TOLs is required to ensure the valves can perform their Appendix “R” safe shutdown functions, when required. Upon discovery, the licensee entered the issue into its corrective action program, initiated predefine parameters (PMID) and created surveillance work orders to test the TOLs at the next opportunity. There was not a cross-cutting aspect to this violation.

This issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Disposition Screening," because the finding was associated with the “Equipment Performance” attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). The finding was of very low safety significance because the finding did not represent an actual loss of functionality of the isolation condenser system containment isolation valves. (Section1R05.7)

Inspection Report# : 2008006 *(pdf)*

Significance: SL-IV Dec 18, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Perform a 10 CFR50.59 evaluati0on for exceeding continuous raiting on the EDG.

The inspectors identified a Severity Level IV NCV associated with the failure to perform a safety evaluation in accordance with 10 CFR 50.59. Specifically, the licensee failed to perform a safety evaluation when non-conservatively changing the design basis loading for the emergency diesel generators (EDG) in the design calculation. This resulted in the expected loading during a design basis accident no longer being bounded by the EDG endurance testing requirements contained in the Technical Specifications (TS). Because the licensee did not also evaluate the effect on the existing endurance test loading requirements, the testing no longer adequately verified the capability of the EDG to power its predicted loading during a LOOP/LOCA. This adverse change increased the probability of a malfunction of equipment important to safety EDG during a LOOP/LOCA event.

Because the issue affected the NRC=s ability to perform its regulatory function, this finding was evaluated using the traditional enforcement process. The finding was determined to be more than minor because the inspectors could not reasonably determine that the change in EDG loading, which adversely affected equipment important to safety, would not have ultimately required NRC approval. The finding was determined to be of very low safety significance (Green) based on the results of the SDP Phase 1 screening worksheet. Specifically, the licensee was eventually able to demonstrate through an engineering evaluation, that the EDG loads would not exceed the bounding values contained in the endurance test criteria. The inspectors determined that there was no cross cutting aspect to this issue. (Section 1R21.b.1)

Inspection Report# : 2008007 *(pdf)*

Significance: G Dec 18, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to meet the EDG Power Factor Testing Requirement
The inspectors identified an NCV for the failure to meet the requirements contained in SR 3.8.1.15. Specifically, the testing that the licensee performed to meet SR 3.8.1.15 did not test to a power factor as close to the accident load power factor as possible. These testing methods did not demonstrate the capability of the EDG to support the emergency core cooling system loading and were non-conservative. The 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 AEquipment Performance,@ and affected the cornerstone objective of ensuring the availability and reliability of the EDGs. Specifically, the licensee=s testing methods for SR 3.8.1.15 did not demonstrate the capability of the EDG to support ECCS loading and was non-conservative. This finding was of very low safety significance, because the inspectors answered ANo@ to all five questions under the Mitigating Systems Cornerstone column of the Phase 1 worksheet. Specifically, the licensee subsequently performed the required testing in SR 3.8.1.15 to the expected power factor, and the EDGs performed satisfactorily. The inspectors determined that there was no cross-cutting aspect to this issue. (Section 1R21.b.2)

Inspection Report# : 2008007 *(pdf)*

Significance: G Sep 30, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Identify and Correct Issues with the Operation and Testing of the Isolation Condenser Emergency Make-up Pump
The inspectors identified a performance deficiency involving a Non-Cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," having very low safety significance for the failure to promptly identify and adequately correct issues with the operation and testing of the isolation condenser emergency make-up pump. The licensee's corrective actions for this issue included restoring the inventory of hoses and connectors to the appropriate number, sizes, and locations.

This finding was more than minor because it involved the equipment performance and procedure quality attributes of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The issue was of very low safety significance based on the low initiating event probability and, because of the slow onset of the flooding and the reduced decay heat in the reactor core at the time recovery actions would be necessary, the licensee would be able to

reasonably perform recovery actions that would prevent core damage. The primary cause of this finding was related to the cross-cutting issue of Problem Identification and Resolution, Corrective Action Program, because the licensee failed to take appropriate corrective actions to address safety issues in a timely manner, commensurate with their safety significance and complexity (P.1.(d)).
Inspection Report# : 2007004 *(pdf)*

Significance: G Sep 30, 2007
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
High Pressure Coolant Injection System Removed from Service Due to a Steam Leak on the Inlet Drain Pot Drain Piping
A finding involving a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance was self revealed after the Unit 3 High Pressure Coolant Injection (HPCI) system was removed from service on March 2, 2007, due to a steam leak on the inlet drain pot drain piping. The licensee failed to control changes in the field to a design change on the Unit 3 HPCI system inlet drain pot drain lines with measures commensurate with those applied to the original design. This resulted in a failure of the HPCI inlet drain pot drain lines which made the Unit 3 HPCI system inoperable on March 2, 2007. The corrective actions included repair of the HPCI drain pot drain line leak with the appropriate piping. Unit 2 and 3 HPCI system carbon steel piping susceptible to flow accelerated corrosion was identified and evaluated for acceptance of the degraded condition until replacement.

The inspectors determined that this finding was more than minor because the performance deficiency impacted the Mitigating Systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events. The inspectors determined that the finding was of very low safety significance because the system was inoperable for only a short period of time. The finding was determined not to have a cross-cutting aspect because it was greater than two years old and not reflective of current performance.
Inspection Report# : 2007004 *(pdf)*

Significance: SL-IV Jun 30, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Change of Systems Credited to Mitigate a High Pressure Coolant Injection Pump Room High Energy Line Break
The inspectors identified a finding having very low safety significance and an associated Severity Level IV Non-Cited Violation (NCV) of 10 CFR 50.59 for the licensee's failure to perform an adequate safety evaluation review for changes made to the facility per safety evaluation 2005-02-001. Specifically, the licensee failed to provide an adequate basis as to why changes that credited the isolation condenser for decay heat removal in lieu of the automatic depressurization system and low pressure coolant injection (LPCI)/containment cooling service water and credited the control rod drive system for control of reactor coolant inventory in lieu of LPCI during a postulated high pressure coolant injection (HPCI) room high energy line break (HELB) did not require a license amendment. The licensee entered this issue into its corrective action program.

Because the issue potentially impacted the NRC's ability to perform its regulatory function, this finding was evaluated using the traditional enforcement process. The finding was determined to be more than minor because the inspectors could not reasonably determine that the activity implemented per 10 CFR 50.59 safety evaluation 2005-02-001, which adversely affected systems important to safety, would not have ultimately required NRC approval. The inspectors completed a significance determination of the underlying technical issue using NRC's Inspection Manual Chapter (IMC) 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," and answered "no" to all of the questions under the Mitigating Systems cornerstone. Based upon this Phase 1 screening, specifically this issue did not represent a loss of function, did not result in exceeding a Technical Specification (TS) allowed outage time, and did not affect external event mitigation, the inspectors concluded that the issue was of very low safety significance (Green). The inspectors determined there was not a cross-cutting aspect to this finding. In accordance with the Enforcement Policy, the violation was therefore classified as a Severity Level IV violation.
Inspection Report# : 2007003 *(pdf)*

Significance: G Jun 12, 2007
Identified By: NRC

Item Type: NCV NonCited Violation

Failure to Restore Fire Brigade Equipment to a Ready Status in a Timely Manner after the Performance of a Fire Drill

The inspectors identified a Non-Cited Violation (NCV) of the Dresden Nuclear Power Station Renewed Facility Operating License having very low safety significance (Green) for the licensee’s failure to restore fire brigade equipment to a ready status in a timely manner after the performance of a fire drill on June 12, 2007. Corrective actions by the licensee included the restoration of the fire brigade equipment to a ready status at 8:22 p.m. on June 13, 2007.

The inspectors determined that this finding was more than minor because the failure to restore the fire brigade equipment to a ready status, if left uncorrected, would become a more significant safety concern. The inspectors completed a Phase 1 significance determination of this issue using IMC 0609, “Significance Determination Process,” Appendix A, Attachment 1, dated March 23, 2007. The inspectors determined that the finding affected the Mitigating Systems cornerstone and the fire protection defense-in-depth strategies. However, as discussed by IMC 0609, Appendix A, Attachment 1, issues related to performance of the fire brigade are not included in IMC 0609, Appendix F, “Fire Protection Significance Determination Process,” and require management review. Therefore, the finding was reviewed by NRC management, and was determined to be a finding of very low safety significance (Green) because the condition existed for slightly more than 24 hours, the delay in getting additional self-contained breathing apparatus would be a maximum of about 10 minutes, and the majority of the safety significant equipment in the turbine building is protected by an automatic fire suppression system. The primary cause of this finding was related to the cross-cutting issue of human performance (work practices) because the licensee failed to ensure adequate supervisory and management oversight of work activities (restoration of the fire brigade equipment), such that nuclear safety was supported (H.4(c)).

Inspection Report# : 2007003 *(pdf)*

Significance: G May 18, 2007

Identified By: NRC

Item Type: NCV NonCited Violation

Inadequate Acceptance Criteria in 125 VDC Station Battery Service Test Procedures

The inspectors identified a finding having very low safety significance and an associated NCV of 10 CFR Part 50, Appendix B, Criterion XI, “Test Control.” Specifically, the licensee failed to incorporate the 125 VDC system minimum required voltage value as the acceptance criteria for the minimum battery terminal voltage inservice test procedure DES-8300-28 “Unit 2 - 125 Volt Main Station Battery Service Test.” Following discovery, the licensee entered the issue into its corrective action program to revise the station batteries test procedures to include the minimum required voltage values.

This finding was more than minor in accordance with IMC 0612, Appendix B, “Issue Screening,” because, if the finding was left uncorrected it would become a more significant safety concern. Specifically, the failure to ensure that the battery terminal voltage during the battery discharge per the service test did not drop below the 125 system design input value could have affected the operability of safety-related equipment in the event of a design basis accident condition. The issue was of very low safety significance based on a Phase 1 screening in accordance with IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations.”

Inspection Report# : 2007006 *(pdf)*

Significance: G May 18, 2007

Identified By: NRC

Item Type: NCV NonCited Violation

Adequate Control Voltage for 4160 Breaker’s Closing Coil was not Assured

The inspectors identified a finding having very low safety significance and an associated NCV of 10 CFR Part 50, Appendix B, Criterion III, “Design Control.” Specifically, the licensee failed to assure and verify that the minimum available control voltage at the 4160 circuit breakers was adequate for the closing coils to close the breakers, following a design basis accident and loss of offsite power condition. Following identification of this issue, the licensee obtained a letter from the vendor (General Electric Nuclear Energy) suggesting that it was reasonable to conclude that the closing coils will operate at Dresden’s minimum available voltage (58 volt) level based on ageing testing conducted in 1999 and 2007 testing of one of Dresden’s breakers.

This finding was more than minor in accordance with IMC 0612, Appendix B, “Issue Disposition Screening,” because

the finding was associated with the Mitigated Systems cornerstone attribute of design control and affected the cornerstone objective of ensuring capability and reliability of systems that respond to initiating events. Specifically, the failure to assure adequate control voltage was available to close the 4160 breakers would 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 based on a Phase 1 screening in accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations
Inspection Report# : 2007006 *(pdf)*

Significance: May 18, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Procedurally Control Regulatory Guide 1.97 Control Board Labeling
The inspectors identified a performance deficiency involving a Non-Cited Violation of Technical Specification (TS) 5.4.1 for the licensee's failure to provide procedural controls for the unique identification of Regulatory Guide (RG) 1.97 post-accident instrumentation to aid the control room operator. Specifically, the licensee failed to adequately control the labeling on both unit's control panels and the simulator, resulting in several improperly marked post-accident indicators.

The finding was greater than minor because, if left uncorrected, it could become a more significant safety concern. Inaccurately labeled control room indicators of RG 1.97 post-accident instrumentation could lead to confusion and hamper the response of operators if conflicting indications resulted due to accident conditions. The issue was of very low safety significance based on a Phase 1 screening in accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations."
Inspection Report# : 2007006 *(pdf)*

Barrier Integrity

Significance: Dec 31, 2007
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Failure to Take Corrective Actions to Repair Unit 3B Refuel Floor Fuel Pool Area Radiation Monitor in a Timely Manner
On May 9, 2007, a performance deficiency involving a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," for the failure to promptly identify and adequately correct deficiencies with the Unit 3B Refuel Floor Fuel Pool Area Radiation Monitor (ARM) was self-revealed. The licensee's corrective actions for this issue included replacing associated degraded cables and the radiation monitor's detector on May 10, 2007, and discussing with the ARM system manager the importance of applying adequate technical riqor.

The inspectors concluded that the finding was greater than minor in accordance with Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," issued on September 20, 2007, because the failure to take corrective actions evaluated and described in a root cause or common cause assessment to prevent an unnecessary challenge to a safety system could result in a more safety significant issue. This deficiency unnecessarily challenged a safety system and could have affected the availability and capability of components and systems that respond to initiating events. The issue was of very low safety significance because there was no actual event in progress.

Inspection Report# : 2007005 *(pdf)*

Significance: Dec 31, 2007
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Standby Gas Treatment 'A' Train Exhibiting Flow Oscillations
A performance deficiency involving a non-cited violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test

Control," was self-revealed following 2/3A standby gas treatment (SBGT) train flow controller settings and post maintenance testing which did not ensure that the system would perform satisfactorily in service. The maintenance and testing on the 2/3A SBGT system from September 10 through 13, 2007, did not challenge controller operation because the reactor building ventilation system was also operating during these activities. As a result, SBGT system oscillations were identified a few days later on September 17, 2007, during system operation. Corrective actions by the licensee include revising test procedure DOS 7500-02 to include required test conditions to test the standby gas treatment system in the expected post accident configuration without reactor building ventilation operating.

The inspectors concluded that the finding was greater than minor in accordance with Inspection Manual Chapter (IMC) 0612, Appendix B, "Issue Screening," issued on September 20, 2007, because it impacted the Barrier Integrity Cornerstone (containment) objective. The failure to perform adequate post maintenance testing on systems, structures or components (SSC) can result in SSC not performing satisfactorily in service. The issue was of very low safety significance because the 2/3B SBGT train remained operable and available. This finding has a cross-cutting aspect in the area of human performance (resources) because the licensee did not provide accurate procedures to plant personnel. (H.2(c))

Inspection Report# : 2007005 *(pdf)*

Significance: G Dec 31, 2007
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Unit 2/3 Standby Gas Treatment Auto-started and a Unit 2 and 3 Reactor Building Ventilation Isolation Occurred
On November 9, 2007, a performance deficiency involving a non-cited violation of TS 5.4.1 was self-revealed when a nuclear station operator (NSO) was performing Dresden Operating Procedure (DOP) 0500-03, "Reactor Protection System Power Supply Operation," Revision 36. The NSO did not verify that the area radiation monitor's (ARM) power supply voltage was normal and did not reset all trips on the ARM modules prior to removing an installed jumper which bypassed the trips. As a result, the reactor building ventilation system for both units tripped when the NSO removed the jumper. This required entry into TS 3.6.4.1 Limiting Condition of Operation, Action A for reactor building low differential pressure. The operator had been provided with a marked up copy of the procedure, assigned a concurrent verifier, and briefed on jumper placement and removal and on the use of concurrent verification prior to the event. As an immediate corrective action, the individual was temporarily removed from licensed shift duties. The operations department also modified the pre-job brief for this evolution to include the lessons learned and revised procedure DOP 0500-03.

The finding was greater than minor because it impacted the SSCs attribute of the Barrier Integrity cornerstone objective. The finding was of very low safety significance because it impacted the reactor building differential pressure for a time period of less than one hour. This finding affected the cross-cutting area of Human Performance, "Work Practices," because the NSO failed to utilize human performance error prevention techniques required to safely implement the station procedure. Specifically, the NSO did not practice self-checking and procedure adherence, and failed to use peer checking. (H.4(a))

Inspection Report# : 2007005 *(pdf)*

Emergency Preparedness

Occupational Radiation Safety

Public Radiation Safety

Significance: G Dec 31, 2007
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Failure to Continuously Sample the Unit 2 and 3 Chimney for Particulate and Iodine Effluents
A self-revealed finding of very low safety significance and an associated violation of NRC requirements were identified for the failure to continuously sample the Unit 2 and 3 chimney effluent for particulate and iodine radioactivity. Specifically, for an approximate 4-hour period on July 21, 2007, the primary particulate and iodine effluent sampling system for the chimney was inadvertently rendered inoperable and the licensee failed to establish continuous sampling with auxiliary sampling equipment. Corrective actions taken by the licensee included tailgate training, procedure revisions and the installation of hardware on the effluent monitor panel to reduce the potential for a configuration control problem.

The issue was more than minor because it was associated with the Facilities/Equipment and Program/Process attributes of the Public Radiation Safety Cornerstone, and affected the cornerstone objective to ensure adequate protection of public health and safety from exposure to radioactivity released into the public domain. The inspectors determined that the issue resulted in a failure to satisfy offsite dose calculation manual (ODCM) sampling requirements for an approximate four hour period and represented a finding of very low safety significance because the licensee was able to estimate the effluent release during the period when sampling was interrupted and to determine that the effluent dose during that short period was within regulatory limits and met the effluent dose design objectives of 10 CFR Part 50, Appendix I. A non-cited violation of Technical Specification 5.5.1 and 5.5.4 was identified for failure to satisfy ODCM effluent sampling requirements. The inspectors also determined that the finding had a cross-cutting aspect in the human performance area for inadequate work controls to ensure job site conditions, including environmental conditions that may impact human performance, plant components and the human-system interface did not adversely impact plant operations.

Inspection Report# : 2007005 *(pdf)*

Physical Protection

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

Miscellaneous

Last modified : June 05, 2008