October 30, 2006

Mr. Christopher M. Crane
President and Chief Nuclear Officer
Exelon Nuclear
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3
NRC INTEGRATED INSPECTION REPORT 05000237/2006010;
05000249/2006010

Dear Mr. Crane:

On September 30, 2006, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Dresden Nuclear Power Station, Units 2 and 3. The enclosed integrated inspection report documents the inspection findings, which were discussed on October 5, 2006, with Mr. D. Wozniak and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, four self-revealed findings of very low safety significance (Green) were identified. All of these issues involved violations of NRC requirements. However, because of the very low safety significance and because they were entered into your corrective action program, the NRC is treating these violations as non-cited violations (NCVs) consistent with Section VI.A.1. of the NRC Enforcement Policy.

If you contest the subject or severity of a Non-Cited Violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator, Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Dresden Nuclear Power Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Mark A. Ring, Chief
Branch 1
Division of Reactor Projects

Docket Nos. 50-237; 50-249; 72-037
License Nos. DPR-19; DPR-25

Enclosure:
Inspection Report 05000237/2006010; 05000249/2006010
w/Attachments:
1. Supplemental Information
2. Split Sample Report
3. Tritium Sample Results

cc w/encl:
Site Vice President - Dresden Nuclear Power Station
Dresden Nuclear Power Station Plant Manager
Regulatory Assurance Manager - Dresden
Chief Operating Officer
Senior Vice President - Nuclear Services
Senior Vice President - Mid-West Regional Operating Group
Vice President - Mid-West Operations Support
Vice President - Licensing and Regulatory Affairs
Director Licensing - Mid-West Regional Operating Group
Manager Licensing - Dresden and Quad Cities
Senior Counsel, Nuclear, Mid-West Regional Operating Group
Document Control Desk - Licensing
Assistant Attorney General
Illinois Emergency Management Agency
State Liaison Officer
Chairman, Illinois Commerce Commission

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-237; 50-249; 72-037
License Nos:	DPR-19; DPR-25
Report No:	05000237/2006010; 05000249/2006010
Licensee:	Exelon Generation Company
Facility:	Dresden Nuclear Power Station, Units 2 and 3
Location:	Morris, IL 60450
Dates:	July 1 through September 30, 2006
Inspectors:	C. Phillips, Senior Resident Inspector M. Sheikh, Resident Inspector A. Barker, Project Engineer L. Ramadan, Reactor Engineer D. Melendez-Colon, Reactor Engineer T. Ploski, Senior Emergency Preparedness Analyst M. Gryglak, Reactor Inspector, Decommissioning Branch W. Slawinski, Senior Radiation Specialist B. Dickson, Senior Resident Inspector, Clinton J. Jacobson, Senior Reactor Inspector A. Klett, Reactor Engineer J. McGhee, Reactor Engineer R. Schulz, Illinois Emergency Management Agency S. Orth, Health Physics Program Manager W. Snell, Senior Health Physicist E. Bonano, Health Physicist
Accompanying Personnel:	Tony Go, Radiation Specialist
Approved by:	M. Ring, Chief Branch 1 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000237/2006010; 05000249/2006010; 07/01/2006 - 09/30/2006; Exelon Generation Company, Dresden Nuclear Power Station, Units 2 and 3; surveillance, followup of events and notices of enforcement discretion, public radiation safety, and other activities.

This report covers a 3-month period of baseline resident inspection, a routine inspection by a regional inspector of Independent Spent Fuel Storage Installation activities, a routine public radiation safety inspection, and an announced baseline inspection in emergency preparedness. The inspection was conducted by Region III inspectors and the resident inspectors. Four Green findings, involving four non-cited violations, were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. <u>NRC-Identified and Self-Revealing Findings</u>

Cornerstone: Initiating Events

Green. A performance deficiency involving a non-cited violation of Technical Specification (TS) 5.4.1 was self revealed after the Unit 2 reactor scram on July 4, 2006. The licensee's root cause report determined that the cause of scram was that the Unit 2 inboard main steam isolation valve, (MSIV) 2-203-1A, drifted closed. The pilot air sensing line tubing to the 2-203-1A valve separated from the compression fitting holding it in place. The tubing slipped out of the compression fitting because the fitting was either improperly installed or the fitting may have been too big for the tubing installed.

The finding was greater than minor because it was a precursor to a significant event. The finding was of very low safety significance because all the equipment necessary to mitigate the transient worked as expected. Corrective actions included, 1) the fitting was reinstalled with the correct parts and was leak checked; 2) seven other fittings on the inboard and outboard Unit 2 MSIVs were leak checked with satisfactory results; 3) the fittings on both units will be removed and checked for proper parts during the next refueling outages; 4) MSIV model work orders will be updated to include "Tube Fitting Repair and Replacement Instruction," and include the instructions in work orders where compression fittings are identified. (Section 4OA3.6)

Cornerstone: Mitigating Systems

Green. On July 30, 2006, a performance deficiency involving a non-cited violation of TS 5.4.1 was self revealed when two nuclear station operators (NSOs) failed to exercise appropriate three-way communication and second verification, resulting in the movement of control rod C-9 to an incorrect position during the performance of Dresden Operating Surveillance (DOS) 0300-04, "Control Rod Drive Timing," Revision 39.

The finding was greater than minor because it impacted the human performance attribute of the Reactor Safety Mitigating Systems Cornerstone objective to ensure reliability of systems that respond to initiating events to prevent undesirable consequences. The finding was of very low safety significance because the mispositioned rod did not significantly increase reactivity to a point where power limits were challenged. Corrective actions for this event included: 1) all licensed operators were to take part in a dynamic learning activity in the simulator involving control rod operations and communications; 2) the shift manager was required to be in the control room during all non-emergency control rod moves; 3) the unit supervisor was required to provide direct overview in the "horseshoe" area of the control room during all non-emergency control rod movements; 4) each shift manager was required to perform a paired observation with the crew unit supervisors specifically focused on communications and verification techniques. The primary cause of this finding was related to the cross-cutting issue of human performance (work practices) because the human performance prevention techniques provided to the NSOs, such as three-way communication and a second verifier were not effective in preventing this error. (Section 1R22)

Green. A self-revealing finding involving a non-cited violation of Technical Specification 5.4.1 was identified on February 1, 2006, due to the licensee's failure to include essential information in DOP 1300-11, "Unit 2 Isolation Condenser Fill and Vent," Revision 12, regarding backfilling of the sensing lines after completion of the filling of the isolation condenser piping. This procedural deficiency resulted in the isolation of the flow paths of the isolation condenser for an extended period of time (approximately 22 hours) and online risk changed from Green to Yellow.

This finding was considered more than minor because it affected the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding was determined to be of very low safety significance because even though the flow paths of the isolation condenser were isolated and online risk changed from Green to Yellow, the flow paths could have been restored manually by operator actions. Corrective actions by the licensee included revising procedures DOP 1300-10, "Unit 3 Isolation Condenser Fill and Vent," Revision 19, and DOP 1300-11 to include DPIS 2(3)-1349A and B sensing line backfilling following system piping filling and venting. The primary cause of this finding was related to the cross-cutting issue of human performance (resources) because the licensee did not provide complete, accurate and up-to-date procedures to plant personnel. (Section 4OA3.4)

Cornerstone: Occupational Radiation Safety

Green. A self-revealed finding of very low safety significance, and an associated violation of NRC requirements were identified for the failure to satisfy Technical Specification requirements for access into a high radiation area with dose rates in accessible areas greater than 1000 mrem/hour. As a result, a worker was allowed to enter a steam sensitive area at power that was controlled as a locked high radiation area (LHRA), 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 the worker alarmed when significantly higher than expected dose rates were encountered, resulting in some unnecessary dose to that worker.

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 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 worker's response to the ED alarm. Also, the licensee's ability to assess worker dose was not compromised. A Non-Cited Violation of TS 5.7.1 was identified for the failure to comply with the requirements for access into a high radiation area with dose rates accessible to personnel greater than 1000 mrem/hour. Corrective actions taken by the licensee included modification to the survey maps for steam sensitive areas, tagging of certain LHRA keys to remind radiation protection staff to coordinate entries into these areas with operations staff, and plans to reevaluate the radiation protection department practices for entry into steam sensitive areas, and in general for entry into high radiation areas with the potential for significant dose rate gradients. (Section 2OS1.4)

B. Licensee-Identified Violations

Violations of very low safety significance, which were identified by the licensee have been reviewed by the inspectors. Corrective actions taken by the licensee have been entered into the licensee's corrective action program. These violations and the licensee's corrective action tracking numbers are listed in Section 4OA7.

REPORT DETAILS

Summary of Plant Status

Unit 2 began the inspection period at 912 MWe (95 percent thermal power and 100 percent of rated electrical capacity).

- On July 4, 2006, the unit experienced an automatic reactor scram due to closure of the 1A main steam isolation valve from a failure of a pneumatic supply line to the pilot valve for the main steam isolation valve actuator. The unit returned to full power on July 7, 2006.
- On July 8, 2006, power was reduced to 88 percent to perform control rod adjustment due to the forced outage.
- On July 30, 2006, power was reduced to 90 percent to maintain the plant within environmental limits for river discharge temperature, and returned to full power on August 28, 2006.
- On August 28, 2006 power was reduced to 96 percent to perform control rod drive maintenance, and returned to full power on the same day.
- On September 3, 2006 power was reduced to 61 percent to perform turbine valve testing, control rod drive scram testing, and control rod pattern adjustment. The unit returned to full power on September 4, 2006.

Unit 3 began the inspection period at 912 MWe (95 percent thermal power and 100 percent of rated electrical capacity).

- On July 22, 2006, and again on August 19, 2006, power was reduced to 87 percent to perform control rod pattern adjustment, and returned to full power on the same day.
- On September 10, 2006, power was reduced to 90 percent to perform turbine valve testing and a control rod pattern adjustment, and returned to full power on the same day.
- On September 30, 2006, power was reduced to perform a control rod pattern adjustment, and returned to full power on the same day.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather (71111.01)

a. Inspection Scope

On July 27, 2006, The licensee lost power to all six lift station pumps due to heavy storms in the area causing the 480 volt feed breaker to motor control center bus 051 to trip. The tripped 480 volt breaker de-energized the bus feeding the lift station bearing lube pumps which then tripped off the lift pumps. Subsequently, the licensee restored power to the lift station and all six lift pumps were re-started. During this event, the licensee had to secure the Unit 2 circulating water pump and derate Unit 2 to 890 MWe to reduce the rate of water going to the hot canal. In addition, the plant water flow was made to operate in open cycle by opening a flow regulating gate in the discharged water flow path to prevent overflowing the hot canal. The gate was closed immediately after the lift station pumps were re-started.

The inspectors reviewed the prompt investigation that was initiated for this event and Issue Report (IR) 514086, reviewed the Updated Final Safety Analysis Report, and walked down equipment and systems to verify proper alignment in accordance with the licensee procedures.

This represented one inspection sample.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04Q and S)

.1 Routine Quarterly Reviews

a. Inspection Scope

The inspectors selected a redundant or backup system to an out-of-service or degraded train to determine that the system met the design of the Updated Final Safety Analysis Report. Piping and instrumentation diagrams were used to determine correct system lineup and critical portions of the system configuration were verified. Instrumentation, valve configurations, and appropriate meter indications were also observed. The inspectors observed various support system parameters to determine the operational status of systems. Control room switch positions for the systems were observed. Other conditions, such as adequacy of housekeeping, the absence of ignition sources, and proper labeling were also evaluated.

The inspectors performed partial equipment alignment walkdowns of the:

- Unit 2 Division 1 core spray due to 2B core spray out of service for planned maintenance;
- Unit 2 low pressure coolant injection loop I containment cooling service water; and
- Unit 2 2B reactor protection system motor generator set out-of-service for planned maintenance.

This represented three inspection samples.

b. Findings

No findings of significance were identified.

.2 Complete System Walkdown (71111.04S)

a. Inspection Scope

The inspectors performed an equipment alignment check on Unit 2 and Unit 3 instrument air systems. Both units were at full power and all compressors were aligned for service during the walkdown. Plant procedures as well as piping and instrumentation drawings were reviewed to determine the appropriate equipment alignment prior to the walkdown. Compressor, instrumentation, and valve configurations were observed. Material condition of piping, pipe supports and receivers were also observed. Work orders were reviewed to determine if there were any outstanding issues that could impact system performance. Deficiencies identified in the field were verified to have been entered in the licensee's work control process for resolution and corrective actions were being accomplished in a timely manner. The licensee's corrective action program records were reviewed for the period from July 2004 to July 2006 to verify that equipment issues were being identified at the appropriate threshold and resolution of issues was appropriate. Maintenance rule performance criteria and evaluations were also reviewed. The inspectors verified proper operation of components served by the instrument air system during the plant and control room walkdowns.

This complete system walkdown of Unit 2 and Unit 3 instrument air represented two samples.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05Q)

a. Inspection Scope

The inspectors toured plant areas important to safety to assess the material condition, operating lineup, and operational effectiveness of the fire protection system and features to ensure compliance with the station's Fire Hazard Analysis Report. The review included control of transient combustibles and ignition sources, fire suppression systems, manual fire fighting equipment and capability, passive fire protection features, including fire doors, and compensatory measures. The following areas were walked down:

- Unit 3 reactor building, isolation condenser room, elevation 589', Fire Zone 1.1.1.5.A

This represented one inspection sample.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

1. Internal Flooding

a. Inspection Scope

The inspectors performed a review of the following:

- Unit 2/3 cable tunnel

The inspection focused on reviewing IR 511369, “Water in-leakage - Unit 2/3 cable tunnel.” On June 6, 2006, the licensee identified water in the 2/3 cable tunnel. The water was determined to be approximately two inches deep. The cable tunnel contains safety related and non-safety related cabling for both units. The tunnel was subsequently cleaned and left dry. The licensee walked the tunnel down to look for leaks and determined that ground water was seeping through grooves cut in the tunnel wall.

The inspection activities included, but were not limited to, visually inspecting the cable tunnel for any noticeable damage that could be of a concern, reviewing the Updated Final Safety Analysis Report (UFSAR) and station flooding procedures, and interviewing engineering personnel. The inspectors questioned engineering staff about the impact on the instrument cables in the lowest trays should the water due to flooding rise over the support tray level and immerse the cables. The licensee provided documentation demonstrating the cables were suitable for wet locations. The licensee’s corrective actions included planning on performing repairs to the exposed concrete walls in the areas of water leakage. This would reduce the inflow of ground water in the tunnel.

This represented one inspection sample.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11Q)

a. Inspection Scope

The inspectors observed an evaluation of an operating crew on August 21, 2006. The scenario consisted of a trip of a recirculation pump, a fuel element failure, trip of a circulating water pump, an unisolable isolation condenser steam line break in the reactor building, and an anticipated transient without scram. The inspectors evaluated the licensee's performance against the requirements of 10 CFR 55.59 by verifying that the operators were able to complete the tasks in accordance with applicable plant procedures. The inspectors observed the licensee's evaluators to ensure that no inappropriate cues were provided by the evaluators while assessing the operators' performance.

This represented one inspection sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12Q)

a. Inspection Scope

The inspectors assessed the implementation of the licensee's maintenance rule program to evaluate maintenance effectiveness for the selected system in accordance with 10 CFR 50.65, Maintenance Rule. The following system was selected based on being designated as risk significant under the Maintenance Rule, being in the increased monitoring (Maintenance Rule Category a(1) group), or due to an inspector's identified issue or problem that potentially impacted system work practices, reliability, or common cause failures:

- Unit 3 reactor feedwater pumps.

The inspectors verified the licensee's categorization of specific issues, including evaluation of the performance criteria, appropriate work practices, identification of common cause errors, extent of condition, and trending of key parameters. Additionally, the inspectors reviewed the licensee's implementation of the Maintenance Rule requirements, including a review of scoping, goal-setting, performance monitoring, short-term and long-term corrective actions, functional failure determinations associated with the condition and issue reports reviewed, and current equipment performance status.

This represented one inspection sample.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13)

a. Inspection Scope

The inspectors evaluated the implementation of the licensee's maintenance risk program with respect to the effectiveness of the risk assessments performed before maintenance activities were conducted on structures, systems, and components and verified that the licensee managed the risk in accordance with 10 CFR 50.65, "Maintenance Rule." The inspectors evaluated whether the licensee had taken the necessary steps to plan and control emergent work activities. The inspectors also verified that equipment necessary to complete planned contingency actions was staged and available. The inspectors completed evaluations of maintenance activities on the:

- Unit 2 345 KV line 1220 out of service due to equipment problem;
- Unit 3 unplanned Technical Specification entries for inoperable off-site power; and
- Unit 2/3 B standby gas treatment system failure.

This represented three inspection samples.

b. Findings

No findings of significance were identified.

1R14 Operator Performance During Nonroutine Evolutions and Events (71111.14)

.1 Unit 2 Main Steam Isolation Valve Closure, Group I Isolation, and Reactor Scram

a. Inspection Scope

On July 4, 2006, the Unit 2 inboard main steam isolation valve, 2-203-1A, drifted closed. Steam flow increased in the other 3 steam lines. The increased steam line flow resulted in a Group I Primary Containment Isolation System Signal and a reactor scram. The inspectors responded to the site, verified that equipment expected to mitigate the event worked properly, interviewed members of the operations crew, and reviewed emergency and abnormal operating procedures.

b. Findings

The findings associated with this inspection are found in Section 4OA3.6 of this report.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed operability evaluations (OE) to ensure that operability was properly justified and the component or system remained available, such that any non-conforming conditions were in compliance with Generic Letter 91-18, "Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability." The review included issues involving the operability of:

- Issue Report 505614, "U2 overpower range monitor #6 inoperability light;"
- Issue Report 477486, "Potential non-conformance in shroud repair analysis" and GE-NE-0000-0053-1162-R0 "Dresden shroud repair tie rod toggle bolt reassessment (CAR-40905);"
- Issue Report 514003, "Unit 2/3 cable tunnel-record of observations by NRC;"
- Kewaunee Operability Evaluation OPR 151 - Emergency Diesel Generators 1A (134-031) and 1B (134-032) - Incorrect assumption regarding de-rating of EDGs during elevated load operation;
- Operability evaluation # 06-001, Unit 2, Unit 3, and Unit 2/3 emergency diesel generator rooms ventilation fans; and
- Issue Report 521962, "Senior NRC Resident Inspector (C. Phillips) Inquiry."

This represented six inspection samples.

b. Findings

One licensee identified finding associated with an NRC violation was identified and is described in Section 4OA7.

1R17 Permanent Plant Modifications (71111.17A)

.1 Steam Dryer Replacement

a. Inspection Scope

The inspectors reviewed a permanent plant modification associated with the Units 2 and 3 steam dryer replacement. The inspectors reviewed Engineering Change 356598, "Reactor Building Opening and Replacement Siding and Permanent Exterior Door;" Revision 0, to verify that the completed design changes were in accordance with the specified design requirements, and the licensing bases, and to confirm that the changes did not adversely affect any system's safety function. Design and post-modification testing aspects were verified to ensure the functionality of the modification, its associated system, and any support systems. The inspectors also verified that the modifications performed did not place the plant in an increased risk configuration.

The inspectors also used applicable industry standards to evaluate acceptability of the modifications. This represented one inspection sample.

b. Findings

No findings of significance were identified.

.2 Load Tap Changer

a. Inspection Scope

The annual modification review was performed for the new automatic load tap changer modification included in Engineering Change (EC) 349539: "Replacement of TR32 Transformer." All 5 revisions of the EC were reviewed to evaluate sequencing of change documentation and accuracy of information included. The inspectors reviewed the design adequacy of the modifications by verifying the following:

- energy requirements were able to be supplied by supporting systems under accident and event conditions;
- replacement component properties met functional requirements under event and accident conditions;
- sequence changes remained bounded by the accident analyses and loading on support systems was acceptable;
- structures, systems, and components response times were sufficient to serve accident and event functional requirements assumed by the design analyses;
- control signals were appropriate under accident and event conditions; and
- affected operations procedures were revised and training needs were evaluated in accordance with station administrative procedures.

The inspectors verified that the post modification testing demonstrated system operability by verifying no unintended system interactions occurred, system performance characteristics met the design basis, and post-modification testing results met all acceptance criteria. The inspectors also reviewed issue reports related to permanent plant modifications to ensure that the licensee was entering issues into its corrective action program at an appropriate threshold.

This represented one inspection sample.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed post-maintenance test results to confirm that the tests were adequate for the scope of the maintenance completed and that the test data met the acceptance criteria in Technical Specifications or other design documents. The inspectors also reviewed the tests to determine if the systems were restored to the operational readiness status consistent with the design and licensing basis documents. The inspectors reviewed post-maintenance testing activities associated with the following:

- WO 935092 “Channel A Reactor ½ Scram Received from OPRM 2 Power Loss;”
- WO 935597 “Mechanical Maintenance Replace Tube Fittings to MSIV 2-0204-1A Manifold Block July 4, 2006;”
- WO 99047366-01 “Unit 2, 2 B Core Spray System, Perform Preventive Maintenance on 480v Breaker MCC 29-1 Cubicle E2;”
- WO 871629-36 “Perform DOS 1600-32 Prior to Siding Removal;” and
- WO 00752097-01, “Replace the Motor Control Center Bucket in MCC 39-1, Cubicle E4.”

This represented five inspection samples.

b. <u>Findings</u>

No findings of significance were identified.

1R22 <u>Surveillance Testing</u> (71111.22)

.1 <u>Routine Inspections</u>

a. <u>Inspection Scope</u>

The inspectors observed surveillance testing on risk-significant equipment and reviewed test results. The inspectors assessed whether the selected plant equipment could perform its intended safety function and satisfy the requirements contained in Technical Specifications. Following the completion of each test, the inspectors determined that the test equipment was removed and the equipment returned to a condition in which it could perform its intended safety function.

The inspectors observed surveillance testing activities and/or reviewed completed packages for the tests, listed below, related to systems in the Initiating Events, Mitigating Systems, and Barrier Integrity cornerstones:

- Unit 3 DOS 2300-03, “High Pressure Coolant Injection System Operability And Quarterly IST Verification Test Surveillance,” Revision 84;
- Unit 2 DOS 300-04, “Control Rod Drive Timing,” Revision 39; and
- Unit 2 and 3 Appendix A, Revision 103, Unit NSO daily surveillance log attachment A, 8 hour shift schedule, shift 1 routine checklist for drywell air sampling.

This represented a total of three inspection samples, of which one was In-Service Testing, and two were Routine Surveillance tests.

b. <u>Findings</u>

No findings of significance were identified.

.2 Mispositioning of Control Rod During Single Notch Timing

a. Inspection Scope

The inspectors reviewed Quick Human Performance Investigation Report (IR 514789), and WO 935378, "D2 1M AD CRD Notch Timing for Selected Full Out Drives," dated July 30, 2006. Under this work order the licensee performed DOS 0300-04, "Control Rod Drive Timing," Revision 39. The inspectors interviewed a member of the crew at the time of the mispositioning event and members of the Operations Department Management Staff. The inspectors also observed portions of the training given to the members of the operations department as part of the corrective actions for this event.

b. Findings

Introduction: A Green finding involving a non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was self revealed when a control rod was mispositioned on July 30, 2006.

Description: On July 30, 2006, a Nuclear Station Operator (NSO) mispositioned control rod drive C-9. Single notch rod timing was being performed in accordance with DOS 0300-04, "Control Rod Drive Timing." The operators were attempting to move rod C-9 from position 48 to 46. The operators had attempted to move the rod several times with no success. During these multiple attempts, communication standards between operators broke down. Operations supervision failed to immediately correct the break down in communications. Control rod C-9 finally moved from position 48 to 46 but the NSO failed to notice that the rod had moved. The NSO stated that he was attempting to move the rod again. There was another NSO stationed to act as a second verifier for rod movements, however, the NSO moving the rod did not wait for the confirmation of the second verifier before attempting to move the rod. Control rod C-9 then went from position 46 to 44, which was not the correct position for the control rod.

Although the mispositioning of the control rod was due specifically to the failure to use the human performance tools provided to the NSOs, a contributor to this problem was the degraded material condition of the control rod drive (CRD) system. The need to repeatedly attempt to move a control rod challenged the operators. Previous problems, specifically with rod C-9, were documented in IR 506730 on July 6, 2006, and an adverse trend in CRD reliability was documented in IR 513020 on July 25, 2006. The IR stated that CRD notching reliability had been poor for a number of years and was attributed to two primary causes: aged hydraulic control unit components, and crud wedging and fouling of critical sealing surfaces internal to the control rod drive mechanism. Some of the corrective actions for the degraded CRD system that licensee management was pursuing but had not committed to were: 1) a modification to pre-filter 100 percent condensate flow versus the current 30 percent; 2) replacement of the low pressure turbine casing with corrosion resistant material; 3) request additional guide tube vacuuming to help remove vessel crud; and 4) request additional control rod drive replacements during upcoming refueling outages.

Analysis: The inspectors concluded that the failure to move control rod C-9 to the correct location was a performance deficiency warranting a significance evaluation. The effect of the control rod mispositioning was an unexpected change in reactivity. Using Inspection Manual Chapter (IMC) 0612, “Power Reactor Inspection Reports,” dated September 30, 2005, Appendix B, the finding was more than minor because it impacted the human performance attribute of the Mitigating Systems Cornerstone objective to ensure reliability of systems that respond to initiating events to prevent undesirable consequences.

The inspectors completed a Phase 1 significance determination of this issue using IMC 0609, “Significance Determination Process,” Appendix A, Attachment 1, dated November 22, 2005. The inspectors concluded that the finding impacted the Mitigating Systems Cornerstone in that the human performance degraded the control of reactivity. The inspectors answered “No” to all five questions under the Mitigating Systems Cornerstone column on page A1-9. Therefore, the issue screened as having very low safety significance (Green).

The primary cause of this finding was related to the cross-cutting issue of human performance (work practices) because the human performance prevention techniques provided to the NSOs, such as three-way communication and a second verifier, were not effective in preventing this error.

Enforcement: Technical Specification 5.4.1 required, in part, that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, dated February 1978. Regulatory Guide 1.33, Revision 2, Appendix A, dated February 1978, paragraph 4.b. required operating procedures for the control rod drive system.

Surveillance procedure DOS 0300-04, “Control Rod Drive Timing,” Revision 39, Step G.4, states that all control rod movements shall be performed in accordance with the requirements of general procedure DGP 03-04.

Procedure DGP 03-04, “Control Rod Movements,” Revision 54, Steps a. through h. state, in part, that the NSO moving the control must notify the second verifier of the intended movement of the control rod and receive concurrence from the second verifier before moving the control rod.

Contrary to the above, on July 30, 2006, two NSOs were performing DOS 0300-04, and one NSO moved control rod drive C-9 in the wrong direction prior to the second NSO concurring with the direction of the movement resulting in the mispositioning of control rod drive C-9.

Corrective actions for this event included: 1) all licensed operators were to take part in a dynamic learning activity in the simulator involving control rod operations and communications; 2) the shift manager was required to be in the control room during all non-emergency control rod moves; 3) the unit supervisor was required to provide direct overview in the “horseshoe” area of the control room during all non-emergency control rod movements; and 4) each shift manager was required to perform a paired observation with the crew unit supervisors specifically focused on communications and verification techniques.

Because this violation was of very low safety significance and it was entered into the licensee's corrective action program as IR 514789, this violation is being treated as a non-cited violation, consistent with Section VI.A of the NRC Enforcement Policy. **(NCV 05000237/2006010-01)**

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors screened two active temporary modifications and assessed the effect of the temporary modifications on safety-related system functions as specified in the Updated Final Safety Analysis Report and Technical Specifications. The inspectors also determined if the installation was consistent with system design.

- Engineering Change 353670 (U3) & Engineering Change 353669 (U2) Revision 0, "Remove Neutral Overcurrent Relay for Reserve & Main Feeds of Switchgear Buses 33&34;"
- Engineering Change 356592, " Temporary Platform Outside Reactor Building for Activities Associated with Moving the Steam Dryers and Transportation Containers," Revision 0; and
- Engineering Change 356597, "Installation and Removal of Temporary Platform," Revision 0.

This represented three inspection samples.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP4 Emergency Action Level and Emergency Plan Changes (71114.04)

a. Inspection Scope

The inspectors completed a screening review of Revision 20 of the Dresden Station Annex to the Exelon Standardized Emergency Plan, to determine whether changes identified in this revision may have reduced the effectiveness of the licensee's emergency planning, and to verify that emergency action level and definitions changes associated with NRC Bulletin 2005-02 were adequately incorporated in this revision. The screening review of Revision 20 does not constitute approval of the changes and, as such, the changes are subject to future NRC inspection to ensure that the emergency plan continues to meet NRC regulations.

This represented one inspection sample.

b. Findings

No findings of significance were identified.

1EP6 Drill and Training Evaluations (71114.06)

a. Inspection Scope

The inspectors observed the emergency response activities associated with the drill conducted on August 24, 2006. Specifically, the inspectors verified that the emergency classification and simulated notifications were properly completed, and that the licensee adequately critiqued the drill. Additionally, the inspectors attended the post-drill critique. The inspectors discussed drill discrepancies with the emergency preparedness manager. The inspectors completed one inspection sample by observing the following emergency drill:

- Dresden 2006 Off-year Exercise, “Flooding in the Unit 2 Reactor Building East Corner Room and Subsequent Loss of Offsite Power and Anticipated Transient Without Scram Conditions.”

This represented one inspection sample.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

.1 Review of Licensee Performance Indicators (PI) for the Occupational Exposure Cornerstone

a. Inspection Scope

The inspectors reviewed licensee event reports, corrective action documents, electronic dosimetry transaction data for radiologically controlled area egress, and data reported on the NRC’s web site relative to the licensee’s occupational exposure control performance indicator to determine whether or not the conditions surrounding any actual or potential PI occurrences had been evaluated and that identified problems had been entered into the corrective action program for resolution. In particular, the inspectors reviewed one incident which occurred in August 2005, (described in Section 2OS1.4) that involved entry into a steam sensitive area at power without correctly determining the radiation levels in the area.

This review represented one inspection sample.

b. Findings

No findings of significance were identified.

.2 Plant Walkdowns/Boundary Verifications and Radiation Work Permit Reviews

a. Inspection Scope

The inspectors identified ongoing work performed within high and locked high radiation areas of the plant and other potentially exposure significant work activities and selectively reviewed radiation work permit (RWP) packages and radiation surveys for these areas. The inspectors evaluated the radiological controls to determine if these controls including postings and access control barriers were adequate.

The inspectors reviewed active and recently completed RWPs and work packages which governed activities in radiologically significant areas to identify the work control instructions and control barriers that had been specified. For these activities, electronic dosimeter alarm set points for both integrated dose and dose rate were evaluated for conformity with survey indications and plant procedures.

The inspectors walked down and surveyed (using an NRC survey meter) radiologically significant area boundaries and other radiological areas in the Unit 2 and 3 Reactor, Turbine, and Radwaste Buildings to determine if the prescribed radiological access controls were in place, licensee postings were complete and accurate, and physical barricades/barriers were adequate. During the walkdowns, the inspectors challenged access control boundaries to determine if high radiation area (HRA), locked high radiation area (LHRA), and very high radiation area (VHRA) access was controlled in compliance with the licensee’s procedures, Technical Specifications, and the requirements of 10 CFR 20.1601, and was consistent with Regulatory Guide 8.38, “Control of Access to High and Very High Radiation Areas in Nuclear Power Plants.”

The inspectors selectively reviewed RWP and post-job review documents for selected activities completed during approximately the 7-month period that preceded the inspection to determine if barrier integrity and engineering controls performance (e.g., filtered ventilation system operation) were adequate and to determine if there was a potential for individual worker internal exposures of greater than 50 millirem committed effective dose equivalent. The inspectors reviewed the licensee’s procedures and its methods for the assessment of internal dose, as required by 10 CFR 20.1204, to ensure methodologies were technically sound and included assessment of the impact of hard to detect radionuclides such as pure beta and alpha emitters, as applicable. The inspectors reviewed internal dose assessment results and associated calculations for selected workers that had intakes in 2005 through June 2006. No worker internal exposures greater than 50 millirem committed effective dose equivalent occurred for the period reviewed by the inspectors.

The inspectors reviewed the licensee’s physical and programmatic controls for highly activated and/or contaminated materials (non-fuel) stored within the spent fuel pools. Specifically, radiation protection (RP) procedures were reviewed; RP staff were interviewed; and a walkdown of the refuel floor was conducted. In particular, the radiological control for non-fuel materials stored in the spent fuel pools was evaluated to ensure adequate barriers were in-place to reduce the potential for the inadvertent movement of these materials and to determine compliance with the licensee’s procedure and for consistency with NRC regulatory guidance.

These reviews represented six inspection samples.

b. Findings

No findings of significance were identified.

.3 Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed the results of RP department self-assessments related to the radiological access control program, nuclear oversight department field observations of various radiological activities, and the issue report (IR) database along with individual IRs related to the radiological access and exposure control programs to determine if identified problems were entered into the corrective action program for resolution. In particular, the inspectors reviewed radiological issues which occurred over the 12-month period that preceded the inspection including the review of any high radiation area (HRA) radiological incidents (non-PI occurrences identified by the licensee in high and locked high radiation areas) to determine if follow-up activities were conducted in an effective and timely manner commensurate with their importance to safety and risk based on the following:

- Initial problem identification, characterization, and tracking;
- Disposition of operability/reportability issues;
- Evaluation of safety significance/risk and priority for resolution;
- Identification of repetitive problems;
- Identification of contributing causes; and
- Identification and implementation of corrective actions.

The inspectors evaluated the licensee's process for problem identification, characterization, and prioritization and determined if problems were entered into the corrective action program and were being resolved in a timely manner. For potential repetitive deficiencies or possible trends, the inspectors determined if the licensee's self-assessment activities were capable of identifying and addressing these deficiencies, if applicable.

The inspectors reviewed the licensee's documentation for all potential PI events occurring since the last radiological access control inspection performed in July 2005 to determine if any of these events involved dose rates greater than 25 Rem/hour at 30 centimeters or greater than 500 Rem/hour at 1 meter or involved unintended exposures greater than 100 millirem total effective dose equivalent (or greater than 5 Rem shallow dose equivalent or greater than 1.5 Rem lens dose equivalent). None were identified.

Additionally, the inspectors reviewed the circumstances surrounding an August 2005 LHRA access control issue identified by the NRC while reviewing IRs. This issue is described in Section 2OS1.4.

These reviews represented four inspection samples. Specifically, the samples pertained to the licensee's self-assessment capabilities, its problem identification and resolution program for radiological incidents, a review of the licensee's ability to identify and address repetitive deficiencies, and a review of those radiological incidents and potential PI occurrences of greatest radiological risk.

b. Findings

No findings of significance were identified.

.4 Job-In-Progress Reviews and Review of Work Practices in Radiologically Significant Areas

a. Inspection Scope

The inspectors accompanied radiation protection and maintenance staffs into the penthouse areas of the Radwaste Building (a posted LHRA) and evaluated the radiological controls, job coverage and radiation worker practices during maintenance on a radwaste system pump and waste concentrator. Radiation survey information to support these work activities was reviewed by inspectors, and the radiological job requirements and the access control provisions for these areas was assessed for conformity with Technical Specifications and with the licensee's procedures. The inspectors also attended the pre-job briefings for these activities to assess the adequacy of the information exchanged.

Job performance was observed to determine if radiological conditions in the work areas were adequately communicated to workers through the pre-job briefings and area postings. The inspectors also evaluated the adequacy of the oversight provided by the radiation protection staff including the performance of radiological surveys and air sampling, the work oversight provided by the radiation protection technicians (RPTs), and the administrative and physical controls used over ingress/egress into these areas.

The inspectors also reviewed the licensee's procedures and discussed with RP staff its practices for access into locked high and very high radiation areas and for areas with the potential for changing radiological conditions such as the drywell and steam sensitive areas at power. This included the review of the circumstances and consequences associated with a steam sensitive area entry incident that occurred in August 2005. These reviews were conducted to determine the adequacy of the radiological controls and the radiological hazards assessment associated with such entries. Work instructions provided in RWPs and in pre-entry briefing documents were discussed with RP staff to determine their adequacy relative to industry practices and NRC Information Notices.

The inspectors also reviewed the licensee's procedure and generic practices associated with dosimetry placement and the use of multiple whole body dosimetry for work in high radiation areas having significant dose gradients for compliance with the requirements of 10 CFR 20.1201(c) and applicable industry guidelines. Additionally, previously completed work in areas where dose rate gradients were subject to significant variation such as work under-vessel were reviewed to evaluate the licensee's practices for placement.

These reviews represented three inspection samples.

b. Findings

Introduction: A self-revealed finding of very low safety significance and an associated violation of NRC requirements were identified for the failure to satisfy Technical Specification requirements for worker access into a high radiation area (HRA) with dose rates greater than 1000 mrem/hour. As a result, a worker was allowed to enter a LHRA without adequate recognition of the radiological conditions and without adequate radiological job coverage. The licensee was alerted to the problem when the electronic dosimetry (ED) worn by the worker alarmed.

Description: On August 24, 2005, a contractor conducting helium leak testing of the off-gas system in the Unit-3 off-gas condenser room, a room controlled as a locked high radiation area at power, received an ED dose rate alarm moments after entering the room. The ED dose rate alarm setpoint was established at 1500 mrem/hour for the work activity as specified in RWP No. 10004875, "Unit-3 Steam Sensitive Area Activities at Power." The worker exited the area upon hearing the ED alarm and reported to the radiation protection technician (RPT) providing job coverage for the work. The maximum dose rate recorded by the worker's ED was 2.267 rem/hour. The accumulated whole body dose to the worker for the entry into the off-gas condenser room was less than 40 mrem.

For several days leading up to the August 24th incident, a contract crew was performing leak testing and trouble-shooting activities related to Unit 3 condenser in-leakage. Prior to the initiation of the work project, the RP staff provided the crew a written list of the specific areas/components that were likely to be examined for the project which included expected dose rate information based on historical survey data for the various steam sensitive areas the crew was to enter at power. The work crew also participated in daily pre-job ALARA briefings with the RP staff, during which time the expected radiological conditions for the specific areas to be entered that shift were discussed. The work crew was informed during the daily briefings that dose rates in steam sensitive areas, in general, could be as high as 1000 mrem/hour, while the specific area/component listing indicated that the Unit-3 off-gas condenser room dose rates were expected to be 500 mrem/hour at a specified power level. Based on those expected conditions, worker EDs were set to alarm at 1500 mrem/hour as specified in the RWP.

Shortly before entry into the Unit-3 off-gas condenser room on August 24, 2005, a member of the RP staff briefed members of the crew that dose rates were anticipated to be only about 10 mrem/hour based on historical survey data. However, the historical survey used for that briefing was performed with the off-gas condenser train out-of-service, a condition which was inconsistent with the actual operational status of the equipment. With the off-gas train in-service and the reactor at slightly higher power levels compared to the historical data used for that brief, the radiological conditions in the room were significantly greater. The individual that provided the briefing just prior to the entry failed to recognize the status of the off-gas system train in the room and consequently its impact on the area radiological conditions. Following that briefing, one of the contractor crew members was allowed entry into the room while the RPT assigned job coverage was positioned by the door to prevent unauthorized entry into the room. Although the RPT was equipped with a radiation survey instrument, the area dose rates were not verified prior to the entry because the historical data used to define the area radiological conditions indicated that dose rates were low. The ED alarm occurred when the worker approached the off-gas condenser, which was subsequently determined to have dose rates in accessible areas up to about 2.2 rem/hour at a distance of one foot.

Neither the listing of area/component dose rates provided to the contractor crew days earlier at the outset of the project nor the daily ALARA pre-job briefings informed the workers that area dose rates could exceed 1 rem/hour. Moreover, shortly before the entry into the off-gas condenser room, the worker was informed that dose rates were only about 10 mrem/hour. While historical survey data indicating significantly elevated dose rates in the off-gas condenser room with the off-gas train in-service was available, the correct data was not used to brief the worker before entry into the room.

Technical Specification 5.7.1 governing high radiation area entry requires, in part, that the licensee: (1) establish the dose rates levels in the area and make personnel aware of those levels prior to area entry; and/or (2) accompany those personnel requiring entry into the area with a individual qualified in radiation protection procedures, (i.e., a RPT), equipped with a radiation survey instrument who provides positive control over the activities in the area and performs periodic radiation surveillance. Prior to the entry into the Unit 3 off-gas condenser room on August 24, 2005, the licensee failed to accurately establish (and bound) the area radiological conditions so the worker was not made aware of the significantly elevated dose rates that existed. Also, the licensee failed to provide positive radiological control over the activities in the area and perform adequate radiation surveillance since a worker was allowed to enter a dose rate field in excess of 2 rem/hour, unbeknownst to the RP staff until the workers ED alarmed.

<u>Analysis</u>: The failure to recognize all reasonably anticipated operational conditions that could significantly affect area dose rates and correctly define the radiological conditions prior to entry, or alternatively verify the radiological conditions through a radiation survey, resulted in the failure to meet Technical Specification requirements for high radiation area entry. This failure represents a performance deficiency as defined in NRC Inspection Manual Chapter (IMC) 0612, “Power Reactor Inspection Reports,” Appendix B, “Issue Screening.” The inspectors determined that the issue 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. Therefore, the issue was more than minor and represented a finding which was evaluated using the Significance Determination Process (SDP).

Since the finding involved a locked high radiation area radiological control issue, the inspectors utilized IMC 0609, Appendix C, “Occupational Radiation Safety SDP,” to assess its significance. The inspectors determined that the finding did not involve ALARA planning or work controls. Also, given the radiological conditions in the area coupled with the workers response to the ED alarm, no overexposure occurred, nor did a substantial potential for an overexposure exist. The licensee's ability to assess dose was also not compromised for this incident. Consequently, the inspectors concluded that the SDP assessment for this finding was of very low safety significance (Green).

The licensee evaluated the event at the time it occurred, and correctly determined those factors that contributed to it. The corrective actions taken by the licensee were adequate; however, following the inspection, the licensee planned to re-evaluate its practices for steam sensitive area entry. No cross-cutting aspects associated with the finding were identified by the inspectors.

Enforcement: Technical Specification 5.7.1, High Radiation Area Entry, requires that radiological conditions of high and locked high radiation areas be established and personnel made aware of these conditions prior to entry and/or an RP qualified individual equipped with a radiation survey instrument provide positive control over the activities in the area and perform periodic radiation surveillance. Contrary to these requirements, on August 24, 2005, as described the foregoing paragraphs, a worker was allowed entry into an area with dose rates in accessible areas greater than 1000 mrem/hour and none of the entry options required by the Technical Specification were met.

Corrective actions taken by the licensee included modification to the survey maps for steam sensitive areas to alert staff about train/system status, tagging of certain LHRA keys to remind staff to coordinate entries into these areas with the operations department, and plans to reevaluate the radiation protection department practices for entry into steam sensitive areas and in general for entry into areas with the potential for significant dose rate gradients. Since the licensee documented this issue in its corrective action program (IRs 366249 and 514894) and because the violation is of very low safety significance, it is being treated as a Non-Cited Violation.
(NCV 50-237/2006010-02; 50-249/2006010-02)

.5 High Risk Significant, LHRA and VHRA Access Controls

a. Inspection Scope

The inspectors reviewed the licensee's procedures and RP job standards and evaluated RP practices for the control of access to radiologically significant areas (high, locked high, and very high radiation areas). The inspectors assessed compliance with the licensee's Technical Specifications, procedures and the requirements of 10 CFR Part 20, and for consistency with the guidance contained in Regulatory Guide 8.38. In particular, the inspectors evaluated the RP staff's control of keys to LHRAs and VHRAs, the use of access control guards during work in these areas, and methods and practices for independently verifying proper closure and locking of access doors upon area egress. The inspectors selectively reviewed key issuance/return and door lock verification records and key accountability logs for selected periods in 2006 to determine the adequacy of accountability practices and documentation. The inspectors also reviewed selected records and evaluated the RP staff's practices for radiation protection manager and station management approval for access into Level 2 LHRAs and VHRAs, and for the use of flashing lights in lieu of locking areas to verify compliance with procedure requirements and those of 10 CFR 20.1602.

The inspectors discussed with RP staff the controls that were in place for areas that had the potential to become high radiation areas during certain plant operations to determine if these plant operations required communication before hand with the RP group, so as to allow corresponding timely actions to properly post and control the radiation hazards.

The inspectors conducted plant walkdowns to verify the posting and locking of entrances to numerous LHRAs in the Unit 2 and 3 Reactor and Turbine Buildings and the common Radwaste Building, and for VHRAs (TIP rooms and Drywell airlocks).

These reviews represented three inspection samples.

b. Findings

No findings of significance were identified.

.6 Radiation Worker Performance

a. Inspection Scope

During work in the penthouse and mezzanine areas of the Radwaste Building, the inspectors evaluated radiation worker performance for conformity with radiation protection work requirements, and to determine whether workers were aware of the radiological conditions, the RWP controls and limits in place, and their performance had accounted for the level of radiological hazards present.

The inspectors also reviewed radiological problem reports, which found the cause of the event was due to radiation worker errors, to determine if there was an observable pattern traceable to a similar cause, and to determine if this matched the corrective action approach taken by the licensee to resolve the identified problems.

These reviews represented two inspection samples.

b. Findings

No findings of significance were identified.

.7 Radiation Protection Technician Proficiency

a. Inspection Scope

During job observations and general plant walkdowns, the inspectors evaluated RPT performance with respect to radiation protection work requirements, conformance with procedures and those requirements specified in the RWP, and assessed overall proficiency with respect to radiation protection requirements, station procedures and health physics practices.

The inspectors reviewed selected radiological problem reports generated between July 2005 and July 2006 to determine the extent of any specific problems or trends which may have been caused by deficiencies with RPT work control and to determine if the corrective action approach taken by the licensee to resolve the reported problems, if applicable, was adequate.

These reviews represented two inspection samples.

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation and Protective Equipment (71121.03)

.1 Self-Contained Breathing Apparatus (SCBA) Maintenance/Inspection and Staff Qualifications

a. Inspection Scope

The inspectors reviewed aspects of the licensee's respiratory protection program for compliance with the requirements of Subpart H of 10 CFR Part 20 and to determine if SCBA were properly maintained and ready for emergency use. The inspectors reviewed records of inspection and functional tests for all SCBAs staged in the plant that were required by the licensee's emergency plan. The inspectors evaluated the licensee's capabilities for refilling and transporting SCBA air bottles to and from the control room during emergency conditions. The inspectors determined if all control room staff designated for the active on-shift duty roster, including those individuals on the station's fire brigade, were trained, respirator fit tested, and medically certified to use SCBAs. Additionally, the inspectors reviewed SCBA qualification records for the licensee's radiological emergency teams including the radiation protection, chemistry, and maintenance staffs to determine if a sufficient number of staff were qualified to fulfill emergency response positions consistent with the licensee's emergency plan and the requirements of 10 CFR 50.47. The inspectors also reviewed the respiratory protection training lesson plan to assess its overall adequacy relative to Subpart H of 10 CFR Part 20 and to determine if personal SCBA air bottle change-out was adequately covered as part of the lesson plan.

The inspectors walked down SCBA equipment maintained in the control room, spare SCBA air bottle stations, and other SCBA equipment staged for emergency use in various other areas of the plant. During the walkdowns, the inspectors examined several SCBA units to assess their material condition to determine if air bottle hydrostatic tests were current and if bottles were pressurized to meet procedural requirements. The inspectors reviewed records of SCBA equipment inspection and testing, including regulator flow tests, and observed a member of the licensee's staff demonstrate the methods used to conduct the inspections and functional tests to determine if these activities were performed consistent with procedure and the equipment manufacturer's recommendations. The inspectors also determined through record reviews if the required air cylinder hydrostatic testing was documented and current, if the Department of Transportation required retest air cylinder markings were in place for numerous randomly selected SCBA units and spare air bottles, and if air quality for the compressor used to fill SCBA air bottles was routinely tested to verify Grade-D quality. Additionally, the inspectors reviewed Mine Safety Appliance (MSA) issued training certificates for those contractor staff that performed repairs of SCBA pressure regulators in 2005 and 2006 to determine if those employees that performed maintenance on components vital to equipment function were qualified.

These reviews represented two inspection samples.

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety

2PS3 Radiological Environmental Monitoring and Radioactive Material Control Programs (71122.03)

.1 Reviews of Radiological Environmental Monitoring Reports, Data and Quality Control

a. Inspection Scope

The inspectors reviewed the 2004 and 2005 Annual Radiological Environmental Operating Reports (AREOR), the results of monthly radiological environmental monitoring analyses for 2006 through July 2006, and the most recent licensee assessment results to determine if the Radiological Environmental Monitoring Program (REMP) was implemented as required by the Radiological Effluent Technical Specifications (RETS) and the Offsite Dose Calculation Manual (ODCM). The inspectors reviewed the radiological environmental reports for changes to the ODCM with respect to environmental monitoring, commitments in terms of sampling locations, monitoring and measurement frequencies, land use census, the sample analysis vendor's inter-laboratory comparison program, and analysis of radiological environmental sample data. The inspectors reviewed the ODCM to identify the environmental monitoring stations and evaluated the locations of these stations and the types of samples collected from each to determine if they were consistent with the ODCM and NRC guidance in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes, and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light Water Cooled Nuclear Power Plants," and in Regulatory Guide 4.8, "Environmental Technical Specifications for Nuclear Power Plants" and an associated NRC Branch Technical Position. The inspectors reviewed the Updated Final Safety Analysis Report (UFSAR) for information regarding the monitoring program and the Emergency Response Plan for information regarding meteorological monitoring instrumentation to determine whether the environmental monitoring program was developed consistent with its design basis. The inspectors reviewed the scope of the licensee's audit program to verify that it met the requirements of 10 CFR 20.1101(c).

The inspectors reviewed each event documented in the AREORs which involved a missed sample, inoperable sampler, lost thermoluminescent dosimeter (TLD), or anomalous measurement for the cause and corrective actions and reviewed the licensee's assessment of any positive sample results (i.e., licensed radioactive material detected above the lower limits of detection (LLD)).

The inspectors reviewed sampler station modifications since the last inspection and/or significant changes made by the licensee to the ODCM, as dictated by the 2004 or 2005 land use census. The inspectors reviewed technical justifications for changed sampling locations. The inspectors verified that the licensee performed the reviews required to ensure that the changes did not affect its ability to monitor the impacts of radioactive effluent releases on the environment.

These reviews represented three inspection samples.

b. Findings

No findings of significance were identified.

.2 Examination of the Radiological Environmental Monitoring Stations and Meteorological Towers

a. Inspection Scope

The inspectors walked-down seven of nine “indicator” environmental air sampling stations, all three “special” air sampling stations, and approximately 25 percent of the TLD monitoring stations. The walkdowns were performed to determine whether these environmental stations were located as described in the ODCM. Each station walked-down was examined to assess equipment material condition and operability and to verify that monitoring station orientation relative to plant effluent release points, equipment configuration, and vegetation growth control allowed for the collection of representative samples. In addition, the inspectors evaluated the surface water and ground/well water sampling locations (indicator and control sites) to evaluate the suitability of each in complying with the RETS requirements of the REMP. The inspectors also examined equipment located at the meteorological tower to verify that the tower was sited adequately and that instrumentation was installed consistent with applicable industry guidance. The inspectors determined through review of contractor monthly reports that the meteorological instruments were operable, calibrated, and maintained in accordance with the guidance provided in NRC Regulatory Guide 1.23 (Safety Guide 23, “Onsite Meteorological Programs”) and were consistent with the requirements of the licensee’s Emergency Response Plan. In addition, data recording capabilities were discussed with the licensee’s chemistry staff to verify that meteorological data was sampled and compiled consistent with the aforementioned guidance.

These reviews represented three inspection samples.

b. Findings

No findings of significance were identified. However, the inspectors questioned the basis for the licensee’s waterborne ground/well offsite sampling locations and corresponding compliance with the RETS surveillance requirement specified in Chapter 12.5 of the ODCM. Specifically, Table 12.5-1 of the RETS, “Radiological Environmental Monitoring Program,” requires that quarterly ground/well waterborne samples be collected and analyzed “from three sources only if likely to be affected.”

Waterborne sources likely to be affected are defined in Table 12.5-1 as those that are "tapped for drinking or irrigation purposes in areas where the hydraulic gradient or recharge properties are suitable for contamination."

The licensee has historically sampled from two offsite wells located to the south and west of the Dresden site, both south of the Illinois River. Liquid radwaste effluents are discharged to the Illinois River which flows in a westerly direction. As determined during the licensee's most recent land use census, several residential wells of varying depths located on the northern banks of the Illinois River downstream of the licensee's liquid effluent (radwaste) discharge point are used as potable sources and/or for irrigation purposes and potentially may be affected sources if the hydraulic gradient or recharge properties are suitable. However, the technical basis for limiting the well water sampling program to the two wells historically sampled versus other offsite wells including residential wells downstream of the station's liquid effluent discharge into the Illinois River on the northern banks of the river could not be provided by the licensee. Consequently, compliance with Table 12.5-1 of the RETS could not be determined.

As documented in issue report (IR) 532766, the licensee is contemplating plans to further evaluate Dresden site hydrologic data, including several existing hydrogeology studies, to validate its historical well sampling activities and to assess compliance with Table 12.5-1 of the RETS. Pending the outcome of the licensee's evaluation and the NRCs review of that information, this issue is categorized as an Unresolved Item **(URI 050-237/2006010-03; 050-249/2006010-03)**.

.3 Reviews of Radiological Environmental Monitoring Equipment Maintenance and Testing

a. Inspection Scope

The inspectors reviewed calibration and maintenance records for 2005 through July 2006 for all indicator, control and special environmental air sampling pumps. Additionally, records of 2005 and 2006 quarterly flow rate checks for each of the two "field" use rotameters and records of annual calibrations of the "master" rotameter (all used to measure or validate air sample pump flow rates) were reviewed to determine if the testing and maintenance program for this equipment was implemented consistent with procedural requirements and industry standards, including traceability to the National Institute of Standards and Technology. The inspectors discussed air sample pump maintenance practices with the licensee's chemistry staff and reviewed the actions taken to address the minor equipment failures which were experienced.

These reviews represented one inspection sample.

b. Findings

No findings of significance were identified**.**

.4 Reviews of REMP Sample Collection and Laboratory Analyses

a. Inspection Scope

The inspectors accompanied a REMP technician and observed sample collection and handling associated with the changing-out of air particulate filters and charcoal cartridges and also observed surface water sampling practices at the licensee's cooling water canal. The inspectors verified that the samples were collected in accordance with the applicable sampling procedure and determined whether appropriate practices were used to ensure sample integrity and chain-of-custody. The inspectors also observed the REMP technician perform pump sampling train leak checks to verify that they were accomplished consistent with the procedure and were adequate to ensure no in-leakage paths existed which could impact sample representativeness.

The inspectors reviewed the results of the vendor's inter-laboratory comparison and internal cross-check programs for 2004 and 2005, including radio analytical cross-checks of various environmental media associated with REMP sampling program. The inspectors also reviewed LLD values achieved by the vendor for various sample media and the LLDs achieved by the licensee for its gamma spectroscopy equipment that was used for environmental sample analyses. The inspectors also reviewed the most recent calibration records and selected quality control charts for these gamma spectroscopy detectors. These reviews were performed to assess the analytical detection capabilities for radio-analyses of environmental samples and to determine whether the vendor and the licensee had demonstrated capability to perform precise and accurate radiological measurements with the necessary sensitivity.

These reviews represented one inspection sample.

b. Findings

No findings of significance were identified

.5 Unrestricted Release of Material From Radiologically Controlled Areas (RCAs)

a. Inspection Scope

The inspectors observed locations where the licensee typically monitors potentially contaminated material and individuals leaving the RCA and evaluated the procedures and practices used for control, survey, and release of materials and workers from these areas. The inspectors questioned several radiation protection (RP) staff responsible for the performance of personnel surveying and for releasing material for unrestricted use to assess their knowledge of procedures and protocols and to determine if release surveys were performed appropriately. Through interviews, the inspectors verified that the RP staff had a clear understanding of the radioactive material control program requirements and understood the proper radiation survey equipment to use for various unconditional release applications.

The inspectors assessed the radiation monitoring instrumentation used for both the unrestricted release of workers and for material/equipment from the RCA to determine if it was appropriate for the radiation types present and was calibrated with radiation sources consistent with the plant's nuclide mix. The inspectors reviewed the licensee's criteria for the survey and release of potentially contaminated material and workers to verify that there was guidance on how to respond to an alarm, which indicates the potential presence of licensed radioactive material. The inspectors reviewed the licensee's radiation survey equipment to ensure the radiation detection sensitivities were consistent with the NRC guidance for surface contamination contained in Circular 81-07, "Control of Radioactively Contaminated Material," and Information Notice 85-92, "Survey of Wastes Before Disposal from Nuclear Reactor Facilities," and with Health Physics Positions (position-221) in NUREG/CR-5569 for volumetrically contaminated material. The inspectors reviewed the licensee's program to determine if it adequately identified and evaluated the impact of difficult-to-detect radionuclides (i.e., those that decay via electron capture) and accounted for those nuclides during routine unrestricted release surveys. The inspectors reviewed the licensee's procedures and records to verify that the radiation detection instrumentation was used at its typical sensitivity level based on appropriate counting parameters (i.e., counting times and background radiation levels). The inspectors verified that the licensee had not established a "release limit" by altering the instrument's typical sensitivity through such methods as raising the energy discriminator level or locating the instrument in a high radiation background area.

These reviews represented two inspection samples.

b. Findings

No findings of significance were identified.

.6 Identification and Resolution of Problems for the Radiological Environmental Monitoring and Radioactive Material Control Programs

a. Inspection Scope

The inspectors reviewed licensee corrective action documents generated between January 2005 and July 2006 that related to the REMP or to radioactive material control issues. The reports of two Nuclear Oversight Department audits and a self-assessment of the REMP completed in the same time frame were also reviewed. These reviews were conducted to determine if the licensee adequately assessed the effectiveness of its programs and whether the licensee, through its corrective action program, identified individual problems and trends, evaluated contributing causes and extent of condition, and developed corrective actions to achieve lasting results. These reviews were also performed to determine whether the licensee met its ODCM requirements. The inspectors determined if the licensee's self-assessment and/or audit program was capable of identifying repetitive deficiencies or significant individual deficiencies in problem identification and resolution.

The inspectors reviewed several IRs related to the REMP and the radioactive material control program since the previous inspection, interviewed staff, and reviewed documents to determine if the following activities were being conducted in an effective and timely manner, commensurate with their importance to safety and risk:

- Initial problem identification, characterization, and tracking;
- Disposition of operability/reportability issues;
- Evaluation of safety significance/risk and priority for resolution;
- Identification of repetitive problems;
- Identification of contributing causes;
- Identification and implementation of effective corrective actions; and
- Implementation/consideration of risk significant operational experience feedback.

These reviews represented one inspection sample.

b. Findings

No findings of significance were identified.

.7 Reviews of Radiological Environmental Monitoring Reports, Data and Quality Control

a. Inspection Scope

The NRC performed a number of confirmatory measurements of water samples to evaluate the licensee's proficiency in collecting and in analyzing water samples for tritium and other radioactive isotopes. The samples were collected independently by the inspectors and by licensee personnel and sent to the NRC's contract laboratory for the analysis of tritium. The NRC and licensee obtained these samples from surface water and groundwater sampling points identified in the licensee's Radiological Environmental Monitoring Program and from onsite and offsite groundwater monitoring wells. In particular, samples were obtained as part of the licensee's environmental study of tritium and potential groundwater contamination (ADAMS ML062760005) and as part of an evaluation of leakage from piping beneath the condensate storage tank that is documented in NRC Inspection Report 05000237/2006003; 05000249/2006003 (ADAMS ML061290091). While tritium was the primary radionuclide of concern, selected samples were also analyzed for gamma emitting radionuclides and for strontium. The inspectors performed these reviews to assess the licensee's analytical detection capabilities for radio-analysis of environmental samples and its ability to accurately quantify radionuclides to an acceptable level of sensitivity. The criteria used to compare the sample results is provided in Attachment 2, and the results of the comparisons between the NRC and licensee results is provided in Attachment 3.

The inspectors considered the following activities in evaluating the cause of any comparisons that did not result in an agreement:

a. re-analysis by licensee or NRC's contract laboratory;

b. review of licensee's interlaboratory cross check program results; and

c. review of data for any apparent statistical biases.

Although the comparisons did not result in any significant technical implications for the underlying analyses, the inspectors also discussed with the licensee's chemistry staff ongoing actions to address non-agreements with the NRC's tritium results and to improve the licensee's analytical capabilities for tritium, including the evaluation of analysis times, sample and calibration geometries, and liquid scintillation cocktail composition.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

Cornerstone: Public Radiation Safety

.1 Radiation Safety Strategic Area

a. Inspection Scope

The inspectors sampled licensee submittals for the performance indicator (PI) listed below for the period indicated. To determine the accuracy of the PI data reported during that period, PI definitions and guidance contained in Revision 3 of Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," were used. The following PI was reviewed:

- Radiological Effluent Technical Specification/Offsite Dose Calculation Manual Radiological Effluent Occurrence

The inspectors reviewed the licensee's assignment report database and selected IRs generated since this indicator was last reviewed in May 2005 to identify any potential occurrences such as unmonitored or uncontrolled effluent releases that may have significantly impacted offsite dose to the public. The inspectors reviewed gaseous and liquid effluent summary data, and the results of associated offsite dose calculations for selected periods between May 2005 and August 2006 to determine if indicator results were accurately reported. Additionally, the inspectors discussed methods for quantifying effluents and determining effluent dose with the chemistry staff.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Routine Quarterly Review

On June 5, 2004, the inspectors identified that the licensee's abnormal operating procedure instructions for response to external flooding, and surveillance test procedure for the diesel driven pump necessary to provide make-up to the isolation condenser for response to external flooding, were not adequate for the circumstances. As a corrective action the licensee planned to change the surveillance test procedure and perform a full flow test of the pump. This was documented in issue report IR 246038. The inspectors performed a detailed review of IR 246038 to determine if problem characterization was accurate and to verify corrective action reviews were adequately completed.

a. Effectiveness of Problem Identification

(1) Inspection Scope

The inspectors reviewed the information provided in IR 246038 and the associated action tracking items (ATIs) to verify that the licensee's identification of the problems was complete, accurate, and timely, and that the consideration of extent of condition review, generic implications, and common cause was adequate.

(2) Issues

There were no issues in the area of Effectiveness of Problem Identification.

b. Prioritization and Evaluation of Issues

(1) Inspection Scope

The inspectors reviewed IR 246038 and the associated ATIs. The inspectors considered the licensee's evaluation and disposition of performance issues, and application of risk insights for prioritization of issues.

(2) Issues

There were no issues in the area of Prioritization and Evaluation of Issues.

c. Effectiveness of Corrective Actions

(1) Inspection Scope

The inspectors reviewed the corrective actions which resulted from the ATIs associated with IR 246038 to determine if the IR addressed generic implications and that corrective actions were appropriate.

(2) Issues

In June 2004, the licensee performed a full flow test of the diesel driven emergency flood pump, however, the test was unsuccessful due to instrumentation problems. Subsequently, the licensee created an assignment to test this pump using the pump vendor to perform this testing. The pump vendor could not perform the testing. Additional time was needed to design a new test procedure and revise testing instructions. The isolation condenser make up pump was finally tested on November 21, 2005.

The inspectors reviewed the details of the pump test and the licensee's conclusion as documented. The inspectors noted that the pump capacity at the maximum pump speed was measured at 298 gpm at a discharge head of 114 psig. This was only 80.5 percent of the expected capacity of 370 gpm based on the manufacturers' pump curve. The inspectors were concerned that the lower than expected flowrate may be an indication of pump degradation. In addition, the licensee tested the pump at only one point and assumed that the pump curve would follow the same pump curve established by the manufacturer. The inspectors questioned the method of testing and the method used to extrapolate the flowrate to generate a pump curve.

Also during this inspection, the inspectors questioned licensee personnel to determine if any actions had been taken to address the cause of the 19.5 percent degraded pump test results, and whether the licensee initiated any actions to ensure the pump would not degrade further over time. The inspectors learned that no actions had been taken. Licensee personnel stated that the test result did not necessarily indicate that the pump was degraded. The inspectors concluded that the lack of additional points tested on the pump curve did not ensure the pump would provide adequate flow at design conditions. The inspectors concluded that the licensee's corrective actions of IR 246038 were not fully effective. Specifically, the lack of a robust testing methodology to ensure performance of the emergency flood pump to the manufacturer's pump curve resulted in the licensee's planning to send the pump to an offsite facility for adequate testing. This is an unresolved item pending NRC review of the licensee's planned corrective action to perform full flow testing of the diesel driven pump at design conditions. (**URI 05000237/2006010-04; 05000249/2006010-04**)

This represented one inspection sample.

.2 Cumulative Review of Operator Workaround Program

a. Inspection Scope

In accordance with Inspection Procedure 71152, the inspectors reviewed the cumulative effects of deficiencies that constituted operator workarounds and challenges to determine whether or not they could affect the reliability, availability, and potential for misoperation of a mitigating system; affect multiple mitigating systems; or affect the ability of operators to respond in a correct and timely manner to plant transients and accidents by inspecting the items on the current operator workaround/challenge list. The inspectors also assessed whether operator workarounds and challenges were being identified and entered into the licensee's corrective action program at an appropriate threshold.

This represented one inspection sample.

b. Findings

Introduction: The inspectors identified an unresolved item regarding the performance of DOA 1900-01, "Loss of Fuel Pool Cooling," Revision 14. DOA 1900-01, step D.1.c. can not be performed under a loss of AC power coincident with a loss of coolant accident (LOCA) conditions.

Description: On January 18, 2006, during testing of the 2A fuel pool cooling pump, per DOA 1900-01, heat exchanger tube side relief valves 2-1999-279 (A relief valve) and 2-1999-280 (B relief valve) lifted. On January 20, 2006, during testing of the 2B fuel pool cooling pump, per DOA 1900-01, both A and B heat exchanger tube side relief valves (2-1999-279 and 2-1999-280) lifted. The 2A fuel pool cooling pump was tested again on January 20, and both A and B relief valves lifted. Following each of the incidents, DOP 1900-01, "Fuel Pool Cooling and Cleanup System Startup," was utilized to reseat the relief valves and return the system to a stable condition. The licensee concluded that after a fuel pool cooling pump trip, the pump can not be re-started without operator manual actions in the reactor building.

On January 20, 2006, the licensee determined that DOA 1900-01, step D.1.c. can not be performed under a loss of AC power coincident with loss of coolant accident (LOCA) conditions. Step D.1.c. provides guidance on how to start a fuel pool cooling pump in case access to the reactor building is not possible. This condition affects Unit 2 and likely affects Unit 3. These events were documented in IR 444332.

The inspectors challenged the licensee as to whether the condition of Unit 2 (and potentially Unit 3) fuel pool cooling system should be an operator workaround or challenge. The licensee initiated IR 528541 to address the inspectors' concern. Also, the inspectors inquired as to whether any compensatory actions were in place and if there was an alternate success path to accomplish the re-start of the fuel pool cooling pumps under a loss of AC power coincident with loss of coolant accident (LOCA) conditions. The compensatory action in place directed operations personnel to take actions to ensure DOA 1900-01, step D.1.c. is not used on either unit until a solution to the problem is implemented.

At the end of the inspection period, the licensee was still evaluating if there is an alternate success path to accomplish the re-start of the fuel pool cooling pumps. The inspectors considered this issue to be an unresolved item pending evaluation efforts. **(URI 05000237/2006010-05; 05000249/2006010-05)**

4OA3 Other Activities (71153)

.1 Preoperational Testing of Independent Spent Fuel Storage Installation (ISFSI) at Operating Plants (60854.1)

a. Inspection Scope

The inspector reviewed the licensee's heavy load program to verify licensee's compliance with commitments contained in the Final Safety Analysis Report and applicable industry standards. The inspector reviewed procedures associated with control, rigging and lifting of heavy loads, normal operation and operation in restricted mode of the Unit 2/3 overhead crane, periodic inspections and preventive maintenance on the crane, calibration, and testing of crane components. The inspector reviewed inspection records for the crane annual electrical and mechanical inspections performed recently, the load cell calibration as well as the wire rope and crane hook inspection. The inspector assessed crane modifications performed since 2003 and evaluated their effect on the crane operation. The inspector also evaluated the resolution of a number of condition reports associated with the Unit 2/3 overhead crane.

The inspector also reviewed annual inspection records associated with the special lifting devices and other equipment used for dry fuel storage activities. The inspector reviewed records of Non-Destructive Examination and visual inspection and testing performed on the Multi-Purpose Canister (MPC) lift cleats, the transfer cask trunnions, the transfer cask neutron shield relief valves, and the 125 ton lift yoke. The inspector reviewed certification documentation related to the MPC slings to verify they were adequately tested and designed for the load rating and temperature.

The inspector verified that select licensee personnel were qualified to operate the overhead crane. The inspector confirmed the crane operators received the appropriate technical training and the medical examinations.

b. Findings

No findings of significance were identified.

.2 (Closed) Unresolved Item (URI) 07200037/2004-001-01: "Multi-Purpose Canister Lid Design Change"

a. Inspection Scope

The inspector reviewed documents associated with a change to the MPC lid design and evaluated its impact on the short term and long term capability of the canister to store and transport spent fuel.

b. Findings

No findings of significance were identified.

Introduction: During a 2004 inspection of dry fuel storage activities, NRC inspectors identified that the MPC lid configuration and dimensions had changed (URI 07200037/2004-001-01). The original MPC lid was constructed from a single-piece of metal with a thickness of 10 inches. The revised design was a two-piece construction with an overall thickness of 9.5 inches. The inspectors questioned the adequacy of the new MPC design and its long term storage and transportation capability.

Description: During the inspection, the inspector reviewed in detail the 10 CFR 72.48 completed evaluation for the change in the MPC lid design which was performed by the cask vendor. The vendor addressed the lid design change and evaluated its impact on the structural, thermal and shielding capability of the MPC to store fuel under normal and accident conditions. The vendor concluded that the lid thickness reduction would result in minimal adverse consequences to the thermal and shielding effectiveness of the MPC with no impact on the structural capability. Additionally, none of the limits imposed in the license and the Technical Specifications as well as the requirements in 10 CFR Part 72 would be violated as a result of this change. Therefore, a license amendment request was not required.

The inspector reviewed the vendor's 10 CFR 72.48 evaluation and concluded that it was acceptable. The inspector confirmed that an MPC with a 9.5 inch lid will function as designed and that the onsite and offsite radiation dose limits specified in the license and 10 CFR Part 72 will not be exceeded during operation of an ISFSI.

The inspector verified that the vendor applied for and was granted an NRC license, Revision 4, to its 10 CFR Part 71 license, Certificate of Compliance 9261, to utilize an MPC design with the 9.5 inch lid configuration for transportation purposes.

Unresolved item 07200037/2004-001-01, "Multi-Purpose Canister Lid Design Change", is closed.

.3 (Closed) Unresolved Item (URI) 05000237, 249/2006003-03, "Adequacy of Emergency Diesel Generator Ventilation"

a. Inspection Scope

The inspectors reviewed this unresolved item (URI) to ensure that the issues documented in the report were adequately addressed in the licensee's corrective action program. The inspectors interviewed plant personnel and reviewed test data, calculation results, and other documents.

b. Findings

One licensee identified finding associated with an NRC violation was identified and is described in Section 4OA7.

.4 (Closed) LER 237/2006-001-00, "Unit 2 Isolation Condenser Declared Inoperable Due to Inadequate Backfilling of Instrument Sensing Lines"

a. Inspection Scope

The inspectors reviewed this licensee event report (LER) to ensure that the issues documented in the report were adequately addressed in the licensee's corrective action program. The inspectors interviewed plant personnel and reviewed operating and maintenance procedures to ensure that generic issues were captured appropriately. The inspectors reviewed operator logs, the Updated Final Safety Analysis Report, and other documents to verify the statements contained in the LER.

b. Findings

Introduction: A Green finding involving a non-cited violation of Technical Specification 5.4.1 was self revealed when instrument maintenance technicians observed unexpected indications during the return to service of the isolation condenser condensate line high flow differential pressure switch DPIS 2-1349B. This switch monitors flow in the isolation condenser condensate return line. Procedure DOP 1300-11, "Unit 2 Isolation Condenser Fill and Vent," Revision 12 failed to identify that the sensing lines were required to be backfilled after filling of the isolation condenser piping. This procedural deficiency resulted in the isolation of the flow paths of the isolation condenser and online risk changed from Green to Yellow. The finding was determined to be of very low safety significance because the isolation condenser could have been restored manually by operator actions.

Description: On February 1, 2006, during quarterly performance of DIS 1300-02, “Unit 2 Isolation Condenser Steam/Condensate Line High Flow Calibration,” Revision 26, instrument maintenance technicians observed unexpected indications during the return to service of DPIS 2-1349B that monitors flow in the isolation condenser condensate return line. Switch DPIS 2-1349B responded upscale greater than expected with a vibration and hissing sound heard as the instrument isolation valves were opened. All remaining testing was stopped until troubleshooting and evaluation was completed. Switch DPIS 2-1349B was left isolated with the isolation condenser isolation trip function disabled. Since the procedure was stopped at the point where the isolation logic was defeated, the isolation condenser flow paths were isolated to comply with Technical Specification requirements.

With the isolation condenser isolated, online risk changed from Green to Yellow. Per Technical Specification Bases, Section 3.5.3, “Applicable Safety Analysis,” credit is taken for the isolation condenser in the loss of feedwater transient analysis, therefore this condition could have prevented fulfillment of a safety function because the isolation condenser is required to mitigate the consequences of an accident.

Subsequent investigations determined that the observed unexpected response was the result of voids in the instrument sensing lines of the isolation condenser steam/condensate line high flow differential pressure switch.

The apparent cause of the voids in the instrument sensing lines of the isolation condenser steam/condensate line high flow differential pressure switch was a procedure sequence deficiency that allowed air to remain entrapped in the sensing lines associated with DPIS 2-1349A and DPIS 2-1349B. A review of the activities performed on the Unit 2 isolation condenser condensate return line identified an activity performed in the fall of 2005 that could have caused the voids in the sensing lines. The activity involved the refilling of the isolation condenser piping in accordance with DOP 1300-11, “Unit 2 Isolation Condenser Fill and Vent,” after the lines had been drained for testing. DOP 1300-11 required backfilling of the sensing lines associated with the isolation condenser steam/condensate line high flow differential pressure switch prior to completing the refilling of the isolation condenser condensate return line. To prevent air voids in the sensing lines the correct method was to backfill the sensing lines after completion of the filling of the isolation condenser piping. Procedure DOP 1300-11 did not require the backfilling of the sensing lines after the volume was refilled. Due to sensing line slopes, water drains out from the sensing line high points. Additional sensing line evaporation or leakage could also occur. Air was then entrapped in the sensing lines after the volume was filled leading to the identified unexpected instrument indications. The inspectors determined that the licensee’s failure to include pertinent information in DOP 1300-11, regarding backfilling of the sensing lines after completion of the filling of the isolation condenser piping, was a performance deficiency.

Analysis: The inspectors determined that the licensee’s failure to include essential information in DOP 1300-11 regarding backfilling of the sensing lines after completion of the filling of the isolation condenser piping was a performance deficiency warranting a significant evaluation. Using Inspection Manual Chapter (IMC) 0612, Appendix B, "Issue Screening," issued on September 30, 2005, the inspectors determined that this finding was more than minor because it impacted the Mitigating Systems objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The failure to maintain adequate procedures for working on equipment can result in degrading equipment or rendering equipment inoperable. This condition caused the unavailability of the isolation condenser for an extended period of time (approximately 22 hours). Although the flow paths of the isolation condenser were isolated and online risk changed from Green to Yellow, flow paths could have been restored manually by operator actions. This finding had a cross-cutting aspect in the area of human performance (resources) because the licensee did not provide complete, accurate and up-to-date procedures to plant personnel.

The inspectors completed a Phase 1 significance determination of this issue using IMC 0609, "Significance Determination Process," Appendix A, Attachment 1, dated November 22, 2005. The inspectors determined that the finding impacted the Mitigating Systems Cornerstone. The inspectors answered "No" to all five questions under the Mitigation Systems Cornerstone column because it did not result in a loss of operability, did not represent an actual loss of safety function, and was not potentially risk-significant due to possible external events. Therefore, the issue screened as having very low safety significance (Green).

Enforcement: Dresden Technical Specification Section 5.4.1, "Procedures," states, in part, that written procedures shall be established, implemented, and maintained covering applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, issued February 1978. Procedures addressing filling and venting of safety related system are recommended in paragraph 4 of this regulatory guide. Contrary to the above, on February 1, 2006, the licensee failed to include pertinent guidance regarding backfilling of the sensing lines after completion of the filling of the isolation condenser piping. This failure resulted in the unavailability of the isolation condenser for an extended period of time. This event was entered into the licensee's corrective action program as IR 448800. Corrective actions by the licensee included revising procedures DOP 1300-10 and DOP 1300-11 to include DPIS 2(3)-1349A and B sensing line backfilling following system piping filling and venting. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program, this violation is being treated as an Non-Cited Violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. **(NCV 05000237/2006010-06)**

.5 (Closed) LER 237/2006-003-00, "Unit 2 Reactor Steam Dome Pressure-Low Permissive Switch Determined to Have Been Historically Inoperable"

On May 31, 2006, Engineering and Operations personnel reviewed the equipment history of the Unit 2 Reactor Steam Dome Pressure-Low Permissive Switch and concluded that previous failures of the switch to pass the Technical Specification Allowable Value in 2004, 2005, and 2006 might have incorrectly assumed that the failures occurred at the time of discovery. A further evaluation was conducted which provided firm evidence that the historical failures should have been classified as a failure to meet the Technical Specifications Allowable Value for a period that exceeded Allowed

Outage Times. These events were documented in LER 237/2006-003-00 and IR 495327 and dispositioned as a NRC identified non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, “Corrective Actions,” (NCV 05000237/2006007-02) because the licensee failed to determine the cause of the repeated out-of-tolerance surveillance test results of the 2-0263-52B switch in 2005 and 2006, until prompted by the inspectors. As immediate corrective action, the licensee reduced the surveillance frequency to adequately monitor the switch’s performance. The licensee also required all system managers and first line supervisors to review the station procedure for the instrument performance trending program, and implemented a manufacturer’s recommendation to use smaller step changes in applied pressure to improve set point accuracy. Corrective actions in IR 495327 were reviewed by the inspectors and no findings of significance were identified.

This LER is closed.

.6 Licensee Event Report (LER) 237/2006-004-00, “Unit 2 Reactor Scram due to Main Steam Isolation Valve Closure”

a. Inspection Scope

The inspectors responded to the site after being notified of the above event. The inspectors interviewed operators and licensee management after the event. The inspectors reviewed the licensee’s root cause report of the event associated with Issue Report 506230.

b. Findings

Introduction: A Green finding involving a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” was self revealed after the Unit 2 reactor scram on July 4, 2006. The licensee’s root cause report determined that the cause of scram was that the Unit 2 inboard main steam isolation valve (MSIV), 2-203-1A, drifted closed. The pilot air sensing line tubing to the 2-203-1A separated from the compression fitting holding it in place. The tubing slipped out of the compression fitting because the fitting was either improperly installed or the fitting may have been too big for the tubing installed.

Description: At 2:59 a.m. on July 4, 2006, the Unit 2 inboard MSIV, 2-203-1A, drifted closed. Steam flow increased in the other 3 steam lines. The increased steam line flow resulted in a Group I primary containment isolation system (PCIS) signal and a reactor scram. All control rods inserted and all automatic actions took place as expected. Reactor pressure was controlled with the isolation condenser and no relief valves lifted.

The licensee’s investigation identified that the pilot air sensing line tubing to the 2-203-1A valve had separated from the compression fitting holding it in place. With no air to keep the pilot valve in place, the air was allowed to port off the bottom of the MSIV air cylinder and the valve went closed. The tubing did not break, it slipped out of the compression fitting holding it in place. The licensee’s root cause report said that the root cause was insufficient gripping force by the fitting. This could have been caused by one of two reasons. First, the fitting was installed incorrectly. Second, the tubing may be smaller than what the fitting was sized for.

The licensee identified that the fitting that held the tubing in place did not contain the correct parts. The fitting was a 1/4 inch type 316 stainless steel compression fitting manufactured by Crawford - Swagelock. This fitting was required to have a two piece ferrule inside that gripped the tubing. The licensee found that the fitting that came loose had a single piece ferrule manufactured by Parker Hannifin. Previous licensee guidance issued prior to 2003 was that intermixing components made by different manufacturers between fittings was not allowed.

In addition, the ferrule removed from the fitting was found to have a diameter of .212 inches. The root cause report stated that when the fitting was found the mechanical maintenance first line supervisor put the ferrule onto the tubing and saw that it slid up and down the tubing several inches. This indicated that the installed tubing was not 1/4 inch. The licensee stated that the first line supervisor tried to retighten the existing fitting configuration but it again slipped out of the fitting. Licensee management stated that the ferrule size may have been made less than .250 inches during the attempt to reconnect the fitting. The end of the tubing that slipped out of the fitting was cut off for analysis but was lost by the workers doing the repairs. The actual tubing size was not measured. The licensee performed testing on the potential configuration of a 1/4 inch Swagelock fitting with a 1/4 inch Parker ferrule onto a 3/16 inch tubing, after the repairs had been performed and the tubing size came into question, and found that this configuration could be tightened sufficiently to hold pressure. The licensee performed testing on a 1/4 inch Swagelock fitting with a 1/4 two piece Swagelock ferrule on a 3/16 inch piece of tubing and could not get that configuration to hold pressure. Based on this testing the licensee could not rule out that the tube size was smaller than expected.

The last time the valve was worked was in refueling outage D2R18 in October 2003. The work performed at that time would have required the disassembly and reassembly of the compression fitting but the licensee could not say for sure that it was at this point that the fitting pieces were intermixed since the work package details were not that specific. Dresden Station mechanical maintenance work planners currently include a standard instruction titled, “Tube Fitting Repair And Replacement Instructions,” in the MSIV work packages. This guidance specifically prohibited the intermixing of Parker and Swagelock fitting parts. This guidance was available and in place in 2003, however, it was not in the work packages used on MSIV 2-203-1A during D2R18.

Analysis: The inspectors concluded that the failure to properly secure the fitting on the 2-203-1A MSIV was a performance deficiency warranting a significance evaluation. The affect of the identified deficiency was that the 2-203-1A MSIV went closed during Unit 2 reactor operation and caused a reactor scram. Using Inspection Manual Chapter (IMC) 0612, “Power Reactor Inspection Reports,” dated September 30, 2005, Appendix B, the finding was greater than minor because it was a precursor to a significant event.

The inspectors completed a Phase 1 significance determination of this issue using IMC 0609, “Significance Determination Process,” Appendix A, Attachment 1, dated

November 22, 2005. The inspectors concluded that the finding impacted the Initiating Events cornerstone. The inspectors answered “No” to the question on transient initiators under the Initiating Events cornerstone column on page A1-9, in that the finding did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available. Therefore, the issue screened as having very low safety significance (Green). The last time maintenance was performed on MSIV 2-203-1A was in October 2003 and therefore this issue was not considered an example of current licensee performance.

Enforcement: Technical Specification 5.4.1 required, in part, that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, dated February 1978. Regulatory Guide 1.33, paragraph 9.a. required that maintenance that can affect the performance of safety-related equipment should be properly pre-planned and performed in accordance with written procedures, documented instructions, or drawings appropriate to the circumstances.

Contrary to the above, the work packages used on MSIV 2-203-1A during refueling outage D2R18 in October 2003, for disassembly and reassembly of the compression fitting that subsequently failed, were inappropriate to the circumstances, in that, Dresden Station mechanical maintenance standard instruction entitled, “Tube Fitting Repair And Replacement Instructions,” which provided guidance specifically prohibiting the intermixing of Parker and Swagelock fitting parts, was not in the work packages. Corrective actions included, 1) the fitting was reinstalled with the correct parts and was leak checked; 2) seven other fittings on the inboard and outboard Unit 2 MSIV’s were leak checked with satisfactory results; 3) the fittings on both units will be removed and checked for proper parts during the next refueling outages; and 4) MSIV model work orders will be updated to include “Tube Fitting Repair and Replacement Instruction,” and include the instructions in work orders where compression fittings are identified. Licensee Event Report 237/2006-004-00, “Unit 2 Reactor Scram Due To Main Steam Isolation Valve Closure” is closed.

Because this violation was of very low safety significance and it was entered into the licensee’s corrective action program as IR 506230, this violation is being treated as a non-cited violation (NCV), consistent with Section VI.A of the NRC Enforcement Policy. **(NCV 05000237/2006010-07)**

4OA6 Meetings

.1 Exit Meeting

The inspectors presented the inspection results to the Plant Manager, Mr. D. Wozniak, and other members of licensee management on October 5, 2006. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was discussed.

.2 Interim Exit Meetings

An interim exit was conducted for:

- c. Occupational radiation safety radiological access control inspection and protective equipment with Mr. D. Bost and other licensee staff on July 28, 2006;
- Emergency Preparedness inspection with Mr. R. Ford on August 1, 2006;
- Independent Spent Fuel Storage Installation inspection with Mr. D. Wozniak, and other members of licensee management on August 4, 2006;
- Steam Dryer Replacement inspection was discussed with Mr. J. Griffin on August 16, 2006; and
- Public radiation safety radiological environmental monitoring and radioactive material control programs with Mr. D. Wozniak and other licensee staff on August 18, 2006. Following the onsite inspection, a telephone conversation was held with Messrs. J. Griffin and R. Kalb on September 22, 2006, to discuss the Unresolved Item documented in Section 2PS3.2.
- Public Radiation Safety with Mr. J. Strmec on October 16, 2006

4OA7 Licensee-Identified Violation

The following violations of very low significance were identified by the licensee and were violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement manual, NUREG-1600, for being dispositioned as NCVs.

Cornerstone: Mitigating Systems

Title 10 of the Code of Federal Regulations (CFR) Part 50, Appendix B, Criteria XI, “Test Control,” states, in part, that “A test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures, which incorporate the requirements and acceptance limits contained in applicable design documents.” As described in IRs 347338, 449319, 462705, 466360, and 466834, the licensee failed to establish a test procedure with appropriate acceptance criteria to verify the proper operation of all three emergency diesel generator ventilation systems and no documentation was found that established it had ever been verified since installation. The finding was more than minor because it mirrored the example given in IMC 0612, Appendix E, 3i, on page E-9, dated June 22, 2006. The licensee had to perform extensive testing of the ventilation systems and re-perform environmental qualification calculations to demonstrate that the design basis requirement could be met. The test results caused the licensee to change the design basis requirement to ensure that the ventilation systems could perform their accident required function.

Cornerstone: Barrier Integrity

Title 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” requires, in part, that activities affecting quality shall be prescribed by documents, instructions, procedures, or drawings and shall be accomplished in accordance with these instructions, procedures, or drawings. After the Unit 2 reactor startup on

July 5, 2006, the licensee identified that the procedure for calculating estimated critical prediction (ECP), ECP NF-AB-715, "Critical Predictions With PowerPlex III," Revision 3, Step 4.2.4 was inadequate in that the procedure called for entering the time from shutdown to startup twice. This altered the calculation for the amount of xenon present and altered the ECP. The finding was more than minor because the failure to correctly calculate the ECP could become a more significant safety concern if left uncorrected. The finding was of very low safety significance because when the ECP was recalculated correctly it was determined the reactor core went critical at the expected time. As a corrective action the licensee planned to change NF-AB-715 to correctly account for xenon during a hot plant startup. The inspectors also concluded that this finding affected the cross-cutting issue of human performance (resources) because the licensee did not provide complete and accurate procedure documentation to plant personnel.

Attachments:
1. Supplemental Information
2. Split Sample Report
3. Tritium Sample Results

KEY POINTS OF CONTACT

Licensee personnel

D. Bost, Site Vice President
D. Wozniak, Plant Manager
C. Barajas, Senior Operations Supervisor
H. Bush, Radiation Protection Manager
J. Ellis, Regulatory Assurance Manager
R. Ford, Emergency Preparedness Manager
R. Gadbois, Operations Director
D. Galanis, Design Engineering Manager
V. Gengler, Dresden Site Security Director
G. Graff, Operations Training Manager
J. Griffin, Regulatory Assurance - NRC Coordinator
T. Hanley, Engineering Director
R. Kalb, Environmental Chemist
D. Leggett, Nuclear Oversight Manager
M. Mikota, Dry Cask Project Manager
P. O'Connor, Lead License Operator Requalification Training
M. Otten, Corporate Training
M. Overstreet, Lead RP Supervisor
E. Rowley, Chemistry
B. Rybak, Acting Regulatory Assurance Manager
C. Symonds, Training Director
K. Zerwas, Project Manager

NRC personnel

M. Ring, Chief, Division of Reactor Projects, Branch 1

IEMA personnel

R. Schulz, Illinois Emergency Management Agency
R. Zuffa, Resident Inspector Section Head, Illinois Emergency Management Agency

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000237/2006010-01	NCV	Mispositioning of Control Rod During Single Notch Timing (Section 1R22)
05000237/2006010-02 05000249/2006010-02	NCV	Failure to Satisfy Technical Specification LHRA Access Requirements During Entry Into a Steam Sensitive Area at Power (Section 2OS1)
05000237/2006010-03 05000249/2006010-03	URI	Adequacy of Ground/Well Waterborne Monitoring to Satisfy Radiological Effluent Technical Specification Surveillance Requirements (Section 2PS3.2)
05000237/2006010-04 05000249/2006010-04	URI	Full Flow Testing of the Diesel Driven Flood Pump at Design Conditions (Section 4OA2)
05000237/2006010-05 05000249/2006010-05	URI	DOA 1900-01, step D.1.c. Can Not Be Performed Under a Loss of AC Power Coincident with Loss of Coolant Accident (LOCA) Conditions (Section 4OA2)
05000237/2006010-06	NCV	Unit 2 Isolation Condenser Declared Inoperable Due to Inadequate Backfilling of Instrument Sensing Lines (Section 4OA3)
05000237/2006010-07	NCV	Failure to Include Adequate Instructions for Fitting Reassembly in Main Steam Isolation Valve Work Package (Section 4OA3)

Closed

05000237/2006010-01	NCV	Mispositioning of Control Rod During Single Notch Timing
05000237/2006010-02 05000249/2006010-02	NCV	Failure to Satisfy Technical Specification LHRA Access Requirements During Entry Into a Steam Sensitive Area at Power (Section 2OS1)
05000237/2006010-06	NCV	Unit 2 Isolation Condenser Declared Inoperable Due to Inadequate Backfilling of Instrument Sensing Lines
05000237/2006010-07	NCV	Failure to Include Adequate Instructions for Fitting Reassembly in Main Steam Isolation Valve Work Package (Section 4OA3)
07200037/2004-001-01	URI	Multi-Purpose Canister Lid Design Change
05000237/2006003-03 05000249/2006003-03	URI	Adequacy of Emergency Diesel Generator Ventilation Systems

237/2006-001-00	LER	Unit 2 Isolation Condenser Declared Inoperable Due to Inadequate Backfilling of Instrument Sensing Lines
237/2006-003-00	LER	Unit 2 Reactor Steam Dome Pressure-Low Permissive Switch Determined to Have Been Historically Inoperable
237/2006-004-00	LER	Unit 2 Reactor Scram due to Main Steam Isolation Valve Closure

Discussed

05000237/2006007-02	NCV	Unit 2 350 psig Reactor Low Pressure Emergency Core Cooling System Permissive Switch Out-of-tolerance During Surveillance Testing

LIST OF DOCUMENTS REVIEWED

The following is a list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety but rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

1R04 Equipment Alignment
-OE22932, (Cooper Nuclear Station) Manual Scram Due to Loss of Service Air System Pressure
-DOP 4700-01, Revision 039; Instrument Air System Startup
-DOP 4700-03, Revision 012; Unit 2/3 Instrument Air Cross-Connect Operation
-DOP 4700-08, Revision 022; 3C Instrument Air System Startup
-DOP 4700-09, Revision 003; 2B and 3B Instrument Air Dryer Pre-filter Replacement
-DOP 4700-10, Revision 000; 2A and 3A Instrument Air Dryer Pre-filter Replacement
-DOP 4700-M1 U3, Revision 006; 3A Instrument Air Compressor (3-4706)
-DOP 4700-M1 Unit 2, Revision 008; 2A instrument Air Compressor (2-4706A)
-DOP 4700-M2 U3, Revision 003; 3B Instrument Air Compressor (3-4715B)
-DOP 4700-M2 Unit 2, Revision 013; 2B instrument Air Compressor (2-4715)
-DOP 4700-M4 U3, Revision 004; Unit 3 Instrument Air Compressor 3C Valve Checklist
-M37, Sheet 2, Revision RU; Diagram of Instrument Air Piping (Critical Control Room Drawing)
-M37, Sheet 3, Revision AJ; Diagram of Instrument Air Piping
-M37, Sheet 5, Revision AK; Diagram of Instrument Air Piping
-M37, Sheet 7, Revision H; Diagram of 2B Instrument Air Piping
-M37, Sheet 8, Revision P; Diagram of Instrument Air Piping
-M37, Sheet 9, Revision N; Diagram of 2A Instrument Air Piping
-M367, Sheet 1, Revision 0; Diagram of Instrument Air Piping for 3A and 3B Compressors
-M367, Sheet 4, Revision L; Diagram of 3C Instrument Air Piping
-IR 00236440; 3C IAC ‘A’ Dryer Tower Blowdown Incorrectly; dated July 15, 2004
-IR 00243856; Unit 3C Instrument Air Compressor Atlas Copco; dated August 12, 2004
-IR 00245437; 3C Unloaded Intercooler Pressure Degrading; dated August 18, 2004
-IR 00259300; 3C IAC Fails To Load; dated October 2, 2004
-IR 00353892; 2A IAC Failed to Load; dated July 17, 2005
-IR 00368370; CDE and EPIX Form Completion for MR Functional Failure; dated August 30, 2005
-IR 00385164; Required Shutdown of Risk Equipment (3B IAC); dated October 12, 2005
-IR 00438878; 2A IA Dryer; dated January 5, 2006
-IR 00502751; 3A IAC Trip; dated June 22, 2006
-IR 00504450; Maintenance Rule Functional Failure for the 3A IAC; dated June 28, 2006
-IR 00505484; OOT Review Identifies Adverse Trend for 2B IAC TI (2-4741-211B); dated June 30, 2006
-IR 00505487; OOT Review Identifies Adverse Trend for 2B IAC TI (2-4741-212B); dated June 30, 2006

1R13 Maintenance Risk Assessments and Emergent Work Control
-OP-AA-102-104, “Pertinent information program,” Revision 0, attachment 2
-Shift Manager Narrative Logs, dated July 31, 2006
-WC-AA-101, On-line Work Control, Revision 12

-OP-AA-108-107-1001, Station Response to Grid Capacity Conditions, Revision 2
-OP-AA-108-107, Switchyard Control, Revision 2
-OP-AA108-111-1001, Severe Weather and Natural Disaster Guidelines. Revision 2
-OP-AA-111-101, Operating Narrative Logs and Records, Revision 5
-Appendix X, Technical Specifications Action Statement Initiated Surveillances, Revision 26
-DAN 902(3)-8 C-2, Revision 16; Actuation of Transformer Switches
-DOP 6400-15, Revision 04; TR 32 Load Tap Changer Operation
-DOA 6100-03, Revision 09; Aux Power Transformer Trouble
-DOA 6500-08, Revision 11; Unit 3 4KV Emergency Bus Degraded Voltage
-DOA 6500-12, Revision 14; Low Switchyard Voltage
-IR 515089, July 31, 2006; Unplanned Entries for Inoperable Off-site Power
-IR 516570, August 3, 2006; Training/Clarifications for Assessing Risk for Degraded Grid

1R15 Operability Evaluations
-Unit 3 Control Room Operator Logs for August 20, 2006
-DOP 03-01, "Power Changes," Revision 85
-DOS 0500-05, "Calculation of Core Thermal Power," Revision 22

1R17 Permanent Plant Modifications
-EC 349539, Revision 0 through 5; Replacement of TR 32 Transformer
-EC 361917, Revision 0; Refine Minimum Switchyard Voltage for TR32 in Automatic
-DOP 6400-15, Revision 04; TR 32 Load Tap Changer Operation
-DOA 6100-03, Revision 09; Aux Power Transformer Trouble
-DOA 6500-08, Revision 11; Unit 3 4KV Emergency Bus Degraded Voltage
-DOA 6500-11, Revision 05; 4KV Bus Overvoltage
-DOA 6500-12, Revision 13; Low Switchyard Voltage
-DOA 6500-12, Revision 13; Low Switchyard Voltage
-IR 475027, April 5, 2006; RES AUX TR 32 TROUBLE
-IR 481708, April 21, 2006; TR 86 Load Tap Changer Automatic Functional
-IR 483186, April 25, 2006; Improper Lug on the TR 32 SPR Control Circuit
-IR 483594, April 26, 2006; TR 32 SPR Troubleshooting Results
-IR 498533, June 9, 2006; RES AUX TR 32 TROUBLE alarm received
-IR 515089, July 31, 2006; Unplanned Entries for Inoperable Off-site Power

1R19 Post-Maintenance Testing
-IR 505537, "Channel A Reactor ½ Scram Received from OPRM 2 Power Loss"
-WO 99047366-01 "Perform PM inspection on 480v breaker MCC 29-1 cubicle E2 [2B core spray pump downstream injection valve 2-1402-25B]"
-DOS 1600-32, "Secondary Containment Leak Rate Test," Revision 04

1R22 Surveillance Testing
-Quick Human Performance Investigation Report IR 514789,"Mispositioning of Control Rod During Single Notch Timing"
-DRE05-0054, "Tornado Depressurization Analysis for the Design of the Dryer Enclosure as Part of the Steam Dryer Replacement Project," Revision 1
-DRE05-0052, "Steam Dryer Load Drop Analysis Outside the Reactor Building for the Steam Dryer Replacement Project," Revision 0

1EP4 Emergency Action Level and Emergency Plan Changes
-Dresden Station Annex to the Exelon Standardized Emergency Plan; Revision 20

2OS1 Access Control to Radiologically Significant Areas
-RP-AA-460; Controls for High and Very High Radiation Areas; Revision 10
-RP-DR-460-1001; Additional High Radiation Exposure Controls; Revision 0
-RP-DR-ALR-001; Steam Sensitive Area Entries; Revision 3
-RWP 10006211 (and associated ALARA files); Unit 2 and 3 Radwaste System Maintenance; Revision 2
-RP-AA-210; Dosimetry Issue, Usage, and Control; Revision 6
-RWP 10006170 (and associated ALARA files); Unit 2 Shutdown Cooling System Maintenance; Revision 2
-RWP 10006171; Unit 2 Reactor Water Cleanup System Maintenance; Revision 2
-IR 00366249; Unexpected ED Dose Rate Alarm Received; dated August 24, 2005
-Historical (2001 - 2005) Radiation Survey Data for Off Gas Condenser Rooms
-RWP 10004875 (with attachments); Unit 3 Steam Sensitive Areas - Activities at Power; Revision 1
-IR 00482205; Turnstile Not Locked Properly; dated April 23, 2006
-IR 00394351; Locked High Radiation Door Hinges Need Repair; dated November 3, 2005
-Nuclear Oversight Field Observation Reports; NOSPA-DR-05-4Q, 05-3Q, 06-1Q and 06-2Q ; Various RP Attributes; dated between September 19, 2005 and May 19, 2006
-Focus Area Self-Assessment Report No. 380066; Radiological Access Control; dated June 23, 2006
-RP-DR-463; Quarterly High, Locked High and Very High Radiation Posting and Door Checks; Revision 1
-High, Locked High and Very High Radiation Area Boundary, and Posting Surveillance Records; January - June 2006
-RP-AA-220; Bioassay Program; Revision 3
-RP-AA-222; Methods for Estimating Internal Exposure From In-Vivo and In-Vitro Bioassay Data; Revision 1
-Internal Dose Investigation Results Summary from D3R18 and D2R19 Refuel Outages, Associated Whole Body Count Data and Dose Calculations; November 2004 - 2005
-DFP-0800-39; Control of Material/Equipment Hanging in Units 2 and 3 Spent Fuel Pools; Revision 14
-Dresden Units 2 and 3 Spent Fuel Pool (non-fuel) Inventory; dated April 2006

2OS3 Radiation Monitoring Instrumentation and Protective Equipment
-EP-AA-1000; Exelon Nuclear Emergency Plan - Part II; Revision 16
-Respiratory Protection Qualification Matrices for Chemistry, Radiation Protection, Operations and Maintenance Staffs; 2005 and 2006
-RP-DR-826; MSA Self-Contained Breathing Apparatus Inspection; Revision 9
-SCBA Monthly Inspection, Monthly Functional Test and Annual Regulator Flow Test Records; Selected Records for 2005 - June 2006
-RP-DR-827; Use of the Eagle Breathing Air Compressor System; Revision 1
-Mine Safety Appliance (MSA) Training Certificates for Various Contractor Staff; dated August 20, 2004

2PS3 <u>Radiological Environmental Monitoring and Radioactive Material Control Programs</u>

-Offsite Dose Calculation Manual; Chapters 5 - 6 and Appendices A - C (Revision 2); Chapter 11 (Revision 2); Chapter 12 (Revision 5); and Appendix F (Revision 2)
-Dresden Nuclear Power Station Annual Radiological Environmental Operating Report for 2004 (issued May 13, 2005) and for 2005 (issued May 5, 2006)
-Environmental Inc. Midwest Laboratory, Sampling Procedures Manual; Revision 10
-CY-AA-170-100; Radiological Environmental Monitoring Program; Revision 1
-CY-AA-170-1000; Radiological Environmental Monitoring Program and Meteorological Program Implementation; Revision 2
-RP-AA-503; Unconditional Release Survey Method; Revision 0
-TID-2004-003; Unconditional Release Detection Thresholds and Dose Consequences; Revision 0
-Focus Area Self-Assessment Report; Radiological Environmental Monitoring Program; dated July 28, 2006
-Chemistry, Radwaste, Effluent and Environmental Monitoring Audit Report; Audit No. NOSA-DRE-06-04; dated May 3, 2006
-ODCM, REMP, Effluent and Environmental Monitoring Audit Report; Audit No. NOSA-DRE-05-08; dated December 7, 2005
-Field Rotameter (Serial Numbers 95W012433 & 91W506166) Quarterly Flow Verifications for January 2005 - July 2006
-Master Rotameter (Serial Numbers 91W513308) Calibration Certificates; dated August 2, 2005, August 8, 2005 and July 25, 2006
-Annual Maintenance and Monthly Flow Checks for Environmental Air Sample Pumps; Pump Nos. 445, 455, 486, 457, 454, 458, 446, 443, 452, 431, 453, 451, 429, 444, and 462; January 2005 - July 2006
-Efficiency Calibrations and LLD Determinations for Gamma Spectroscopy Systems (detectors used for volumetric solids and liquids); various dates between January 2003 and April 2005
-Harza Consulting Engineers and Scientists; Dresden Station Site Groundwater Study; dated July 1991; and Dresden Groundwater Study; dated January 1995
-The RETEC Group; Final Draft Groundwater Tritium Investigation Report, Dresden Generating Station; dated October 19, 2005
-Sundance Environmental and Energy Specialists, Ltd.; Hydrogeology and Groundwater Investigation at the Dresden Nuclear Power Station near Morris, Illinois; dated June 30, 2005
-Murray and Trettel, Inc. Monthly Reports on the Meteorological Monitoring Program at the Dresden Nuclear Station; January 2005 - June 2006
-Krueger Tower Inc.; Inspection of Dresden Meteorological Tower; dated April 28, 2006
-Dresden Assessment Report-2006-01 (and associated sample analyses data and dose assessment); Unit 2 Isolation Condenser Actuation; dated July 26, 2006
-IR 425093; Evaluation of Met Tower Inspections; dated November 17, 2005
-IR 444695; Met Tower Wind Speed Computer Points Not Working; dated January 23, 2006
-IR 0351236; Air Sampler Scheduled To Be Out-of-Service Over 3 Years; dated June 30, 2004
-IR 348779; Monitor Almost Shipped with Source; dated June 29, 2005
-IR 00532766; Potential ODCM Violation; dated August 17, 2006
-Focus Area Self-Assessment Report; Radioactive Material Control; dated June 30, 2006
-AR 00540522; Duplicate Tritium Analysis Do Not Agree with NRC; dated October 05, 2006

4OA1 <u>Performance Indicator Verification</u>

-LS-AA-2150; Attachment 1; Monthly PI Data Elements for RETS/ODCM Radiological Effluent Occurrences; May 2005 - July 2006

-Quarterly Summary Data of Dresden Station Units 2/3 Doses Resulting from Airborne Releases and Aquatic Effluents; 3rd Quarter 2005 - 2nd Quarter 2006
-IR 455274; Unexpected ED Alarm; dated February 17, 2006
-IR 473188; Unit 3 RWCU Pump Room Gate Locking Mechanism; dated March 31, 2006
-ED Alarm and ED Transaction Reports; Selected Data for July 2005 - June 2006
-IR Database (RP Department Generated or Assigned); July 2005 - June 2006

4OA5 Other Activities
-IR 357297; Deficiency Found During Crane Inspection But No IR/WR; dated July 27, 2005
-IR 352760; Feed Breaker For 2/3 Reactor Building Crane Fails PM; dated July 13, 2005
-IR 359550; Relay Setting Order Revised Time Delay But Not Response Time; dated August 3, 2005
-IR 368573; 613 Rx Crane Does Not Travel East/West; dated August 31, 2005
-IR 436303; Manual AK-25 BKR Not on a PM Schedule; dated December 22, 2005
-IR 454218; Per IR 279886 RX Bldg Crane Hook Leveling Issues; dated February 15, 2006
-IR 444102; January 2006 Safety Committee Open Issues; dated January 20, 2006
-IR 456785; Scaffold Moved to cause An Interference with Rx Bdg Crane; dated February 21, 2006
-IR 479268; Rail Clamp Missing; dated April 16, 2006
-IR 500186; Rx Bldg Crane and New Fuel Jib Vrane Mod; dated June 15, 2006
-IR 506230; Unit 2 Automatically Scrammed on MISV Closure
-IR 506230; Root Cause Report, Dresden Unit 2 Scrammed Due To Main Steam Isolation Valve (MSIV) Closure Resulting From A Failed Tube Connection On The Pneumatic Supply To The 1A MSIV Solenoid Manifold Pilot Block
-IR 509389; 2/3 Reactor Building Bridge Crane Remote Would Not Work; dated July 14, 2006
-Certificate of Conformance 1108II, 2Tp Slings, dated June 27, 2006
-EC (DCP) 342007; Install New Zone and Hoist Controls for Reactor Building Crane
-EC (DCP) 359740; Resolve Two Reactor Building Crane Safety Concerns- (1) Interference with Overhead Tanks Galleries Bracing Angles and (2) Interference with Duct on the West Side of Reactor Building
-EC (DCP) 354453; Install Interlock System Between Reactor Building Crane and Fuel Storage Vault Jib Crane
-Procedure DES 5800-02; Overhead Crane Annual Electrical Inspection; Revision 2
-Procedure MA-AA-716-022; Control of Heavy Loads Program; Revision 0
-Procedure DFP; Operation of 2/3 Reactor Building 125/9 Ton Crane; Revision 21
-Procedure DOS 0800-06; Unit 2/3 Reactor Building Crane Operation In Restricted Modes Test; Revision 15
-Procedure MA-DR-MM-5-58003; Visual Inspection And Preventive Maintenance of Unit 1, Unit 2, Unit 3, and Unit 2/3 Overhead and Gantry Cranes; Revision 2
-Procedure DMP 5800-18; Load Handling of Heavy Loads and Lifting Devices; Revision 17
-Evaluation; 72.48 Screening/Evaluation 580; Change Nominal lid thickness; dated June 20, 2002
-Screening 72.48-338 (Dresden); MPC Lid Thickness and Associated Component Change; Revision 4
-WO 603192; D2/3 An Com Hi-Track Water Jacket Relief Valve Testing; dated June 30, 2004
-WO 725639; D2/3 An Com Hi-Track Water Jacket Relief Valve 2/3-0899-1B TE; dated April 7, 2005
-WO 800841; D2/3 An Com Test of U 2/3 RB 125 Ton Lift Yoke, dated June 29, 2006
-WO 915228; D2/3 An Con Test of HI-TRAC Trunnions; dated June 28, 2006

-WO 916065; D2/3 An Com Test of MPC Lift Cleat (2/3-08209-2A); dated June 27, 2006
-WO 916066; D2/3 An Com Test of MPC Lift Cleat (2/3-08209-2B); dated June 27, 2006
-NF-AB-760, “Reactivity Anomaly Determination,” Revision 0
-Nuclear Engineers Logs for September 5, 2006

LIST OF ACRONYMS USED

ADAMS	Agencywide Documents Access and Management System
AREOR	Annual Radiological Environmental Operating Report
ATI	Action Tracking Items
CFR	Code of Federal Regulations
CR	Condition Report
CRD	Control Rod Drive
DIS	Dresden Instrument Surveillance
DOP	Dresden operating Procedure
DOS	Dresden Operating Surveillance
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
EC	Engineering Change
ECP	Estimated Critical Prediction
ED	Electronic Dosimetry
EDG	Emergency Diesel Generator
EP	Emergency Preparedness
gpm	gallons per minute
HRA	High Radiation Area
IEMA	Illinois Emergency Management Agency
ISFSI	Independent Spent Fuel Storage Installation
IMC	Inspection Manual Chapter
IR	Inspection / Issue Report
LER	Licensee Event Report
LHRA	Locked High Radiation Area
LLD	Lower Limit of Detection
MPC	Multi-Purpose Canister
MSIV	Main Steam Isolation Valve
MWe	megawatts electrical
NCV	Non-Cited Violation
NUREG	Nuclear Regulatory Guide
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NSO	Nuclear Station Operator
ODCM	Offsite Dose Calculation Manual
OE	Operability Evaluations
PARS	Publicly Available Records
PI	Performance Indicator
REMP	Radiological Environmental Monitoring Program
RETS	Radiological Effluent Technical Specifications
RP	Radiation Protection
RPT	Radiation Protection Technician
RWP	Radiation Work Permit
SCBA	Self-Contained Breathing Apparatus
SDP	Significance Determination Process
TLD	Thermoluminescent Dosimeter
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report

URI	Unresolved item
VHRA	Very High Radiation Area
WO	Work Order

Attachment 2
Confirmatory Measurements Comparison Criteria

The NRC applied the comparison criteria contained in NRC Inspection Procedure (IP) 84750, "Radioactive Waste Treatment, and Effluent and Environmental Monitoring," dated March 15, 1994, to determine if the licensee's measurement results were in statistical agreement with the NRC measurement results. For the purposes of this comparison, the NRC result is divided by its associated uncertainty to obtain the resolution. (Note: For purposes of this process, the uncertainty is defined as the relative standard deviation, one sigma, of the NRC's contract laboratory's analysis.) The licensee's result is then divided by the corresponding NRC result to obtain the ratio (licensee result/NRC). The licensee's measurement is in agreement if the value of the ratio fall within the limits shown in the following table for the corresponding resolution.

Resolution	**Acceptance Range (Licensee Result/NRC Result)**
<4	Technical Judgement[1]
4-7	0.5-2.0
8-15	0.6-1.66
16-50	0.75-1.33
51-200	0.80-1.25
>200	0.85-1.18

For analyses that are below the minimum detectable concentration (either for the licensee or NRC's contract laboratory), the measurements are determined to be in agreement if both are below the minimum detectable concentration or if one has an uncertainty that is within the minimum detectable concentration.

[1]The inspectors used technical judgement in reviewing results having a relative 1 sigma uncertainty greater than 25 percent (i.e., resolution less than 4). In these cases, the values were typically very close to the laboratory's detection capabilities, and greater variability was expected. Consequently, these sample comparisons were made based on the inspectors' qualitative review of the analytical results.