February 1, 2007

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 05000237/2006011;
05000249/2006011

Dear Mr. Crane:

On December 31, 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 January 17, 2007, 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, one self-revealed and two NRC identified 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 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;
License Nos. DPR-19; DPR-25

Enclosure:
Inspection Report 05000237/2006011; 05000249/2006011
w/Attachment: Supplemental Information

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;
License Nos:	DPR-19; DPR-25
Report No:	05000237/2006011; 05000249/2006011
Licensee:	Exelon Generation Company
Facility:	Dresden Nuclear Power Station, Units 2 and 3
Location:	Morris, IL 60450
Dates:	October 1 through December 31, 2006
Inspectors:	C. Phillips, Senior Resident Inspector M. Sheikh, Resident Inspector D. Melendez-Colon, Reactor Engineer J. McGhee, Reactor Engineer J. Jandovitz, Reactor Inspector W. Slawinski, Senior Radiation Specialist J. Cassidy, Radiation Specialist R. Schulz, Illinois Emergency Management Agency
Approved by:	M. Ring, Chief Branch 1 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000237/2006011; 05000249/2006011; 10/01/2006 - 12/31/2006; Exelon Generation Company, Dresden Nuclear Power Station, Units 2 and 3; fire protection, operability evaluations, and event followup.

This report covers a 3-month period of baseline resident inspection, routine inspections by regional inspectors; and Temporary Instruction 2515/169, "Mitigating Systems Performance Index Verification." The inspection was conducted by Region III inspectors and the resident inspectors. Three Green findings, involving three 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 on November 20, 2006, when during the Unit 3 hotwell drain down to support Unit 3 startup, operators allowed the 2/3 condensate storage tank (CST) to overflow into the Unit 2 turbine building equipment drain system, which eventually overflowed onto the condensate/condensate booster pump room floor. Corrective actions by the licensee included implementing a standard pre-job brief to address risk and contingencies for this infrequently performed evolution. Additional actions involved revising procedures to include appropriate actions if the 2/3 CST high water level alarm is exceeded and limitations and precautions associated with proper 2/3 CST level, hotwell level, and emergency reject operations.

The finding was considered more than minor because it could be reasonably viewed as a precursor to a significant event. The finding was determined to be of very low safety significance because the event did not affect credited items for shutdown safety. 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)

Cornerstone: Barrier Integrity

Green. A performance deficiency involving a non-cited violation of the Dresden Nuclear Power Station Renewed Facility Operating License was identified by the inspectors due to the licensee's failure to develop a pre-fire plan. Specifically, on November 17, 2006, the inspectors identified that the licensee failed to develop a pre-fire plan for Fire Zone 8.2.6.A, elevation 534'. The licensee has since developed a pre-fire plan for the Fire Zone 8.2.6.A, Elevation 534'.

This finding was considered more than minor because it involved the Barrier Integrity attribute of procedural quality for the control room ventilation system because the failure to develop a pre-fire plan for Fire Zone 8.2.6.A could have adversely impacted the fire brigade's ability to fight a fire. The finding was related to the performance of the fire brigade and was not suitable for SDP evaluation. Therefore, the finding was reviewed by NRC management and determined to be of very low safety significance because no safe shutdown equipment was located in this fire zone. (Section 1R05)

Cornerstone: Mitigation Systems

Green. On October 16, 2006, a performance deficiency involving a non-cited violation of TS Section 5.5.13 having very low safety significance was identified by the inspectors for failure to comply with the TS requirement when visible corrosion on the Unit 2 125 Vdc safety-related battery inter-cell and terminal connections was identified. Specifically, the licensee failed to identify, document or take battery connections resistance measurements on battery cell terminations containing visible corrosion. Upon discovery, the licensee's corrective actions included: cleaning identified corroded inter-cells, reinforcing the expectation that sufficient documentation of corrective actions was to be documented, and taking and recording connection resistance measurements.

The finding was considered more than minor because the failure to ensure that the Unit 2 125 Vdc safety-related battery was being maintained in accordance with applicable procedures to comply with the TS requirements, could result in unacceptable battery terminal connection resistance and decreased battery capacity, rendering the DC system incapable of performing its intended safety function. The finding was determined to be of very low safety significance using the SDP Phase 1 screening worksheet. This finding has a cross-cutting aspect in the area of human performance (work practices) because the licensee did not effectively communicate expectations regarding procedural compliance and personnel did not follow procedures. (Section 1R15)

B. Licensee-Identified Violations

Cornerstone: Occupational Radiation Safety

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

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).

- The unit operated at 760 MWe to 790 MWe between November 3, 2006, and November 16, 2006, due to an extraction steam leak to the 2D1 high pressure feed-water heater. On November 16, 2006, the extraction steam piping that had failed caused a condenser tube leak exceeding 1gpm and the licensee shut the unit down for a maintenance outage to repair the extraction steam piping and the condenser tubes, inspect the electromatic relief valves (ERVs), and replace the 2B reactor recirculation pump seal. The unit was returned to full power on November 27,2006.

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

- On November 10, 2006, power was reduced to 815 MWe for control rod exercising, and returned to full power the same day.
- On October 21, 2006, power was reduced to 829 MWe for rod pattern adjustments, and returned to full power the same day.
- On November 1, 2006, power was reduced to 880 MWe to install a new clean-up demineralizer bed, and returned to full power the same day.
- On November 3, 2006, a Refuel Outage began. The unit was returned to full power on November 24, 2006.
- On December 15, 2006, power was reduced to 660 MWe, when a solenoid valve on the 3A feed-water regulating valve was replaced due to valve oscillations. The unit was returned to full power the same day.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather (71111.01)

a. Inspection Scope

The inspectors assessed the station's readiness for cold weather conditions by conducting detailed inspections of the circulating water ice melt valve and the condensate storage tank heat tracing system. The inspectors chose the ice melt valve for inspection due to its importance in preventing the freezing of water for the service water, circulating water, fire water, and emergency diesel generator cooling water systems. The heat tracing system was chosen because of its importance in maintaining the operability of safety-related piping exposed to extreme temperature conditions. The inspectors reviewed the Updated Safety Analysis Report, and seasonal readiness and adverse weather procedures to determine the operational requirements of the ice melt valve and heat tracing systems during cold weather conditions. The inspectors compared this information to the licensee's seasonal readiness open items list, and open maintenance work requests to ensure that none of the items on these lists impacted the ability of the ice melt valve and heat tracing system to perform their intended functions. The inspectors performed a review of previously initiated issue reports related to cold weather conditions and performed a plant walkdown to ensure that the items documented in the issue reports had been appropriately corrected.

This represented one inspection sample.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04Q)

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 (UFSAR). 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 reactor building closed cooling water system; and
- Unit 3 core spray system.

This represented two inspection 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 2 turbine building, 469'-6" elevation, condensate pump room, Fire Zone 8.2.1.A;
- Unit 2/3 emergency swing diesel generator, 517' elevation, Fire Zone 9.0.C;
- Unit 2/3 crib house, elevations 490', 509', and 517', Fire Zone 11.3;
- Unit 2/3 control room, 'B' heating, ventilation, air conditioning room, elevation 534', Fire Zone 8.2.6.A;
- Unit 3 turbine building, low pressure heater bays, north turbine cavity, elevation 517', Fire Zone 8.2.5.D;
- Unit 3 turbine building, high pressure heaters/steam lines, elevation 517', Fire Zone 8.2.5.E;
- Unit 3 turbine building, low pressure heater bays, elevation 538', Fire Zone 8.2.6.E;
- Unit 3 turbine building, 469'-6" elevation, condensate pump room, Fire Zone 8.2.1.B; and
- Unit 2 battery room, Fire Zone 7.0.A.1.

This represented nine inspection samples.

b. Findings

Introduction: The inspectors identified a non-cited violation (NCV) of the Dresden Nuclear Power Station Renewed Facility Operating License having very low safety significance (Green) for the licensee's failure to develop a pre-fire plan for Fire Zone 8.2.6.A, elevation 534'.

Description: On November 16, 2006, during a fire protection inspection of Fire Zone 8.2.6.A, the inspectors reviewed Dresden Station Fire Protection Reports along with associated pre-fire plans. The inspectors questioned the Fire Marshall to determine if Fire Zone 8.2.6.A had a pre-fire plan associated with it. The Fire Marshall searched the pre-fire plan database and could not find one specifically addressing Fire Zone 8.2.6.A. Fire Zone 8.2.6.A encompasses the main control room train 'B' heating, ventilation, and air conditioning (HVAC) equipment room. Train 'B' HVAC is a safety related train, it supports the control room emergency ventilation system and the control room emergency ventilation air conditioning (AC) system. In addition, the train 'B' HVAC equipment room is part of the control room emergency zone. Within this zone, plant operators are adequately protected against the effects of accidental radioactive gas releases. This zone also allows the control room to be maintained as the center from which emergency teams can safely operate during a design basis radiological release.

As described below in the Enforcement section, pre-fire plans shall be developed for the safety-related areas in the plant. The inspectors determined that Fire Zone 8.2.6.A, elevation 534', was a safety related area and should have had a pre-fire plan. The licensee entered this issue into the station's corrective action program as IR 559569, “NRC inspector questioned station’s fire pre-plans.” By the end of the inspection the station Fire Marshall developed a pre-fire plan for Fire Zone 8.2.6.A, elevation 534' and provided a copy to the inspectors.

Analysis: The inspectors determined that the failure to develop a pre-fire plan for Fire Zone 8.2.6.A, elevation 534', was a performance deficiency warranting a significance evaluation. Using Inspection Manual Chapter (IMC) 0612, Appendix B, “Issue Screening,” issued on November 2, 2006, the inspectors determined that this finding was more than minor because it involved the Barrier Integrity attribute of procedural quality for the control room ventilation system because the failure to develop a pre-fire plan for Fire zone 8.2.6.A could have adversely impacted the fire brigade’s ability to fight a fire. As such, this finding impacted the Barrier Integrity objective to provide reasonable assurance that physical design barriers protect the public from radio nuclide releases caused by accidents or events. Although Fire Zone 8.2.6.A did not have a pre-fire plan associated with it, no safe shutdown equipment was located in this fire zone.

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 affected fire protection defense-in-depth strategies. However, as discussed by IMC 0609, Appendix A, Attachment 1, issues related to performance of the fire brigade are not included in IMC 0609, Appendix F, “Fire Protection SDP,” and require management review. Therefore, the finding was reviewed by NRC management, and was determined to be a finding of very low safety significance (Green) because no safe shutdown equipment was located in this fire zone.

Enforcement: The inspectors determined that the licensee’s failure to develop a pre-fire plan for Fire Zone 8.2.6.A, elevation 534', was a violation of Dresden Nuclear Power Station Renewed Operating License. License conditions 2.E and 3.G of the Unit 2 and Unit 3 Dresden Nuclear Power Station Renewed Facility Operating Licenses state, in part, that “The licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the UFSAR for the facility....” Section 9.5.1, “Fire Protection System,” of Dresden UFSAR states that “The design bases, system descriptions, safety evaluations, inspection and testing requirements, NFPA [National Fire Protection Association] conformance reviews, personnel qualifications, and training are described in Reference 1.”

Section 9.5.10, “References,” of Dresden UFSAR, reference 1, lists “Dresden Units 2 and 3 Fire Protection Reports,” Volumes 1 through 5, and “Fire Protection Program Documentation Package,” Volumes 1 through 13, as the documents to follow for compliance with the fire protection program.

Section 2.5.4, “Fire Fighting Strategies,” of Dresden Station Units 2 and 3 Fire Protection Reports, Volume 1, “Updated Fire Hazards Analysis,” specifies that “Pre-fire plans are provided for all safety-related areas of the plant.” In addition, Dresden Station Units 2 and 3 Fire Protection Program Documentation Package, Volume 12, “References,” also specifies “Pre-fire plans have been developed for the safety-related areas in the plant.”

Also, in procedure OP-AA-201-008, Revision 1, “Pre-Fire Plans,” Paragraph 4.1.1, “Main Body,” the licensee stated, “A pre-fire plan shall be established for all safety related areas, areas representing a hazard to safety-related equipment, and insured buildings.”

Contrary to the above, the licensee failed to develop a pre-fire plan for Fire Zone 8.2.6.A, elevation 534'. This failure could have adversely impacted the fire brigade’s ability to fight a fire. This issue was entered in the licensee’s corrective action program as IR 559569. Corrective actions by the licensee included the development of a pre-fire plan for Fire Zone 8.2.6.A, elevation 534'. 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 a Non-Cited Violation, consistent with Section VI.A.1 or the NRC Enforcement Policy. **(NCV 05000237/2006011-01; 05000249/2006011-01)**

1R06 Flood Protection (71111.06)

a. Inspection Scope

The inspectors reviewed the UFSAR flood analysis documents and reviewed the licensee’s procedures for external flooding. The inspectors reviewed the licensee’s procedures for external flooding for ensuring proper safe shutdown of the plant, and reviewed the licensee’s previously implemented corrective actions for deficiencies associated with flood protection. Currently, the licensee’s approach to flood protection is to let the flood waters in and provide cooling flow to both Units Isolation Condensers via a diesel driven pump. The inspectors had outstanding questions that were open in regard to the diesel driven pump that were documented in URI 05000237, 249/2006010-04, “Full Flow Testing of the Diesel Driven Flooding Pump at Design Conditions.” The inspectors were still waiting for the evaluation of the results of the licensee’s testing of the pumps at the end of the inspection period.

In order to complete the sample the inspectors reviewed UFSAR, Section 9.2.5.3.1, “Dam Failure During Normal Plant Operation, Section 9.2.5.3.2, “Dam Failure Coincident with a LOCA [loss of coolant accident],” and Section 9.2.5.3.3, “Seismically Induced Dresden Dam Failure Coupled With a Small Break LOCA,” regarding licensee actions following a failure of the Dresden Lock and Dam. In addition, the inspectors reviewed DOA 0010-01, “Dresden Lock and Dam Failure,” Revision 20; Design Analysis No. DRE03-0026, “Analysis of the Intake Canal, CCSW Heat Exchanger, and Temporary Pumps Following A Dam Failure and 1" LOCA,” Revision 0A; WO 813176, “D2 SA PM Emergency Diesel Pump (CCSW pump) Operation;” and DOS 1500-19,”Operation of The Dresden Lock and Dam Failure CCSW Emergency Pump,” Revision 0.

This represented one inspection sample.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07A)

a. Inspection Scope

The inspectors verified readiness and availability of the 3B Low Pressure Coolant Injection (LPCI) heat exchanger (Hx) (3-1503-B) by performing the following activities:

- Observed in-progress eddy current testing and reviewed post-testing eddy current results;
- Observed the condition of heat exchanger end bell and tube sheet during walkdown;
- Reviewed the results of tube inspections and outage work packages to determine whether maintenance was performed in accordance with the licensee's maintenance program for heat exchangers and reviewed issue reports to verify that deficiencies were identified and incorporated into the licensee's corrective action program;
- Reviewed the evaluation and corrective actions for IR 556633, "3B LPCI Hx Finds Unexpected Amount of Clam Shells;" and
- Verified the heat exchanger was properly classified under the Maintenance Rule and identified issues received appropriate program reviews.

This inspection represented the completion of one Annual sample.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI) Activities (71111.08)

.1 Piping Systems ISI

a. Inspection Scope

From November 7, 2004, through November 9, 2006, the inspector conducted a review of the implementation of the licensee's ISI program for monitoring degradation of the reactor coolant system (RCS) boundary and the risk significant Unit 3 piping system boundaries. The inspector selected the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI required examinations and Code components in order of risk priority as identified in Section 71111.08-03 of the inspection procedure, based upon the ISI activities available for review during the onsite inspection period.

The inspector observed the following two types of nondestructive examination activities to evaluate compliance with the ASME Boiler and Pressure Vessel Code requirements and to verify that indications and defects were dispositioned in accordance with the ASME Code.

- Ultrasonic examination (UT) of a Unit 3 Main Steam (MS) System Nozzle 3D Inner Radius, no. N3D-1;

- UT of a Unit 3 MS System Nozzle 3D Nozzle to Vessel weld, no. N3D-2;

- UT of a Unit 3 High Pressure Coolant Injection (HPCI) elbow to pipe weld, No. 24-4; and

- Visual Examination (VT) of Unit 3 LPCI System Support, No. M-1200D-1003.

The inspector reviewed information from the licensee concerning recordable indications that were accepted by the licensee for continued service to verify that the licensee's acceptance for continued service was in accordance with the ASME Code. The inspector concluded there were no indications exceeding Code acceptance criteria identified from ISI exams conducted since the beginning of the previous refueling outage.

The inspector reviewed a pressure boundary weld repair on a Code Class 2 portion of the Unit 3 Reactor Coolant System to determine if the welding acceptance and preservice examinations (e.g., pressure testing, visual, dye penetrant, and weld procedure qualification tensile tests and bend tests) were performed in accordance with ASME Code Sections III, V, IX, and XI requirements. Specifically, the inspectors reviewed a Class 2 pressure boundary seal weld conducted on Stand by Liquid Control (SBLC) Valve, 3-1101-43A, last outage.

The inspectors performed a review of ISI related problems that were identified by the licensee and entered into the corrective action program, conducted interviews with licensee staff, and reviewed licensee corrective action records to determine if:

- the licensee had described the scope of the ISI related problems;
- the licensee had established an appropriate threshold for identifying issues;
- the licensee had evaluated industry generic issues related to ISI and pressure boundary integrity; and
- the licensee implemented appropriate corrective actions.

The inspectors performed these reviews to ensure compliance with 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements. The corrective action documents reviewed by the inspectors are listed in the attachment to this report.

The reviews as discussed above counted as one inspection sample.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11Q)

a. Inspection Scope

The licensee did not perform evaluations of operating crews in the simulator this quarter because of the U3 refueling outage. The inspectors observed an evaluation of three senior reactor operators performing evaluations of simulated work in the plant. The operators were to observe a simulation of racking out a 4 KV breaker using procedure DOP 6500, “4 KV Ground and Test Devices,” Revision 22. The senior reactor operators were to observe pre-prepared deficiencies of which they had no previous knowledge, and coach the individuals performing the work. The inspectors observed the licensee’s evaluators to ensure that no inappropriate cues were provided by the evaluators while assessing the operators' performance. The licensee identified several weaknesses in this area and wrote IR 553886, “45 percent Failure Rate on Ops Coaching OBE [out of the box evaluation].

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 systems in accordance with 10 CFR 50.65, Maintenance Rule. The following systems were 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 2 containment cooling service water; and
- Unit 3 containment cooling service water.

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 two inspection samples.

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:

- 'B' control room ventilation out-of-service for planned maintenance;
- Bus 24-1 cubicle cleaning; and
- Unit 3 low pressure coolant injection swing Bus relay calibration and set point change.

This represented three inspection samples.

b. Findings

No findings of significance were identified.

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:

- IR 506780,"Target Rock SRV Wrong Bore Size;"
- IR 551735, "Cable Deficiencies Identified;"
- IR 553418, "DOS 0500-06 Surveillance Failed;"
- IR 556509, "Unit 2 EDG;"
- IR 560013, "Clearances on Electromatic Relief Pilot Valves were found out of tolerance;" and
- Quad Cities IR 530544, "Failure To Meet Surveillance Requirement 3.8.4.2 Concerning Battery Corrosion."

This represented six inspection samples.

b. Findings

(1) Failure to Comply with Technical Specification (TS) 5.5.13 for 125 Vdc Battery Terminal Connection Corrosion and Resistance Measurements

Introduction: The inspectors identified a non-cited violation of TS 5.5.13, Battery Monitoring and Maintenance Program, having very low safety significance (Green) for the licensee's failure to comply with the yearly surveillance requirements to verify the adequacy of Unit 2 125 Vdc safety-related battery electrical connections. Also, the inspectors identified several time periods where the licensee failed to identify, document or take battery connection resistance measurements on battery cell terminations containing visible corrosion.

Description: On September 15, 2006, a Component Design Bases Inspection was completed at Quad Cities Nuclear Power Station, Units 1 and 2. The team identified several issues regarding the 125 Vdc safety-related battery inter-cell and cable to plate electrical connections. The inspectors reviewed Quad Cities IR 530544 associated with this issue and as a result, the inspectors reviewed several selected DC system components' completed surveillance test results to assess licensee performance in handling identified battery cell corrosion. The inspectors sampled monthly, quarterly, and yearly surveillances; reviewed previous IRs and interviewed personnel. The inspectors noted the following issues:

(a) Review of Unit 2 yearly 125 Vdc battery surveillance:

The inspectors reviewed completed Work Order (WO) 847835-1, "Unit 2 Annual TS 125 Vdc Main Station Battery Surveillance." The WO included Dresden Electrical Surveillance (DES) 8300-49, "Yearly Unit 2 (3) Safety Related 125V and 250V Station Battery Inspection," Revision 0.

The WO documented the following deficiencies on DES 8300-49, Attachment B, "Discrepancies and Corrective Actions:"

- Multiple inter-cell connections have discoloration;
- Corrosion found on inter-cell connections 4, 10, 17, 22, 27, 28, 29, 38, 45, 46, 47, 48, 50, 54, 56;
- Majority of inter-cell readings above 120 percent of baseline.

The corrective actions detailed for these discrepancies only stated "corrected" when more details should have been documented.

DES 8300-49, step G.2, Limitation and Actions, stated to list discrepancies and corrective actions on Attachment B.

DES 8300-49, step H.3.a, Acceptance Criteria (AC), stated to record the corrective actions performed on Attachment B.

DES 8300-49, step I.10, provided guidance on battery inspection in accordance with IEEE Standard 450-1995, "IEEE Recommended Practice for Maintenance, Testing, and

Replacement of Vented Lead-Acid Batteries for Stationary Applications," specifically; step I.10.f instructed electrical maintenance staff to clean connections if corrosion is evident, take resistance readings and record as found micro-ohms on Attachment B. This was an AC, however, corrective actions required by DES 8900-49 and the IEEE such as taking resistance readings and documenting on attachment B were not performed to meet the AC. The licensee initiated IR 573281 to capture the inspectors concerns.

In addition, DES 8300-49, Step I.11.j, Note 2, provided guidance for the system engineer to be contacted for disposition/further actions if the baseline of inter-cell resistance readings and 20 percent above the baseline inter-cell resistance readings were exceeded; for this was an identified alert level in IEEE Standard 450-1995. IEEE Standard 450-1995 stated that if resistance readings obtained are more than 20 percent above the installation value or above a ceiling value established by the manufacturer, or if loose connections are noted, re-torque and retest. However, the system engineer was not contacted to perform an engineering evaluation when the majority of inter-cell resistance readings were found to be above 120 percent of baseline resistance readings, nor were any actions initiated or taken to re-torque and retest. The licensee initiated IR 573788 and 544728 to document the inspectors concerns in this area.

(b) Review of Issue Reports:

On May 31, 2006, the inspectors identified ten cells with corrosion on the battery cell posts related to the Unit 2 125 Vdc battery. The licensee initiated IR 495349, created actions to clean identified corrosion on the respective cells, and created actions to address the failure to catch battery corrosion issues during operation's staff daily rounds as expected from non-licensed operators. The inspectors noted that the licensee did issue a work request to clean the corroded cells, however; the licensee failed to take inter-cell resistance readings and record them. The inspectors reviewed five other IRs and noted similar deficiencies.

The inspectors concluded that electrical maintenance, engineering, and operation's personnel failed to follow battery related procedure requirements concerning identification of visible corrosion, failed to record required resistance measurements of inter-cell connections of affected cells several times when corrosion was identified, and failed to use the corrective action process to evaluate the 125 Vdc battery performance when it was identified that a majority of the cell inter-connections were above the baseline value plus 20 percent.

The inspectors determined that the licensee's failure to follow the annual surveillance test procedure and failure to identify corrosion on a safety-related battery system and monitor resistance of affected corroded cells resulted in the failure to comply with IEEE Standard 450-1995 and subsequently TS 5.5.13. Upon discovery, the licensee's corrective actions included: cleaning identified corroded cells, reinforcing the expectation that sufficient documentation of corrective actions was to be documented on Attachment B, and taking connection resistance measurements.

Analysis: The inspectors determined that the failure to follow battery procedure DES 8300-49 requirements to comply with TS 5.5.13 or verify that no visible corrosion existed at the safety-related 125 Vdc terminal connections or verify that battery connection resistance of corroded terminals was within acceptable limits was a performance deficiency and a finding warranting a significance evaluation. Corroded inter-cell connections and post connectors can fail when exposed to the design discharge current. As the battery ages, terminal post corrosion is a common problem that must be corrected by periodic checking and cleaning.

This finding was more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," dated September 30, 2005, Appendix B, "Issue Disposition Screening," in that the finding was associated with the attribute of equipment performance and affected the Mitigating Systems cornerstone objective of ensuring the availability and reliability of the DC power system to respond to initiating events to prevent undesirable consequences. Specifically, visible corrosion on terminal connections and failure to verify battery inter-cell and terminal connections resistance values could potentially result in unacceptable battery terminal connection resistance and decreased battery capacity, rendering the DC system incapable of performing its required safety function.

The inspectors evaluated the finding using IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," Phase 1 screening, dated November 22, 2005. The inspectors answered "No" to all questions under the Mitigating Systems column on page A1-9. Therefore, the issue screened out as having very low significance (Green). The inspectors concluded that the finding was not a design issue resulting in loss of function per Generic Letter 91-18, "Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability," did not represent an actual loss of a system's safety function, and did not affect external event mitigation.

The inspectors determined that the licensee failed to follow procedure requirements to comply with the TS requirement for battery monitoring and maintenance program which incorporated the IEEE Standard 450-1995. However, this did not result in battery inoperability based on subsequent resistance measurement results. The inspectors determined that the DC system would have performed its design function as determined by the licensee's evaluation.

The primary cause of this finding was related to the cross-cutting issue of human performance (work practices) because operations, engineering and maintenance personnel failed to follow procedure requirements during the performance of daily rounds and the annual surveillance test due to a mind set that corrosion did not impact battery operability.

Enforcement: Technical Specifications Requirements 5.5.13 provides guidance on restoration and maintenance of the batteries based on the recommendation of IEEE Standard 450-1995. DES 8300-49, step I.10, provided guidance on battery inspection in accordance with IEEE Standard 450-1995, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications," specifically step I.10.f instructed electrical maintenance staff to clean connections if corrosion is evident, take resistance readings and record as found micro-ohms on Attachment B. This was an acceptance criterion. In addition, DES 8300-49, step I.11.j, Note 2, provided guidance for the system engineer to be contacted for disposition/further actions if the baseline and 20 percent above baseline readings were exceeded, for this is an identified alert level in IEEE 450-1995. Also, IEEE 450-1995 stated that if resistance readings obtained are more than 20 percent above the installation value or above a ceiling value established by the manufacturer, or if loose connections are noted, re-torque and retest.

Contrary to the above, on October 16, 2006, the inspectors identified that the licensee did not take resistance readings and record measurements on the discrepancies and corrective actions sheet, Attachment B, to meet the acceptance criteria and subsequently meet the requirements of DES 8300-49 which implemented IEEE 450-1995 recommended actions. Also the licensee failed to identify corroded cells during monthly or quarterly surveillances until it was brought to their attention by the inspectors on May 31,2006; and finally, the licensee did not implement procedure requirements and subsequently IEEE Standard 450-1995 recommendations when the majority of the 125 Vdc battery inter-cell connections were above the 120 percent baseline values. However, because the violation was of very low safety significance, and the licensee entered the finding into their corrective action program as IRs 543839, 573281, 573788, and 544728. This violation is being treated as a non-cited violation consistent with Section VI.A.1 of the NRC Enforcement Policy.
(NCV 05000237/2006011-02)

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 TSs 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 00044901-01, “Dresden Unit 2 6 year Preventive Maintenance on 480VAC Breaker to the 250VDC Battery Charger;”
- WO 00910771, “Install EC [engineering change] 360423 - Upgrade Unit 3 SBLC [standby liquid control] Heat Tracing Controls;”
- WO 009653703-03, “Replace Unit 2 125 VDC Battery Cell 45;”
- WO 00920451-01, “Replace Degraded Voltage Time Delay Relay (Bus 24-1);” and
- WO 00749880-02, “Unit 3 LPCI Swing Bus Relay cal and set point change.”

This represented five inspection samples.

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities (71111.20)

.1 Unit 3 Refueling Outage

a. Inspection Scope

The licensee conducted a refueling outage on Unit 3 from November 3, 2006, through November 23, 2006. During the outage the licensee replaced the electromatic relief valve flanges, the contents of the standby liquid control tank with a more highly enriched form of sodium pentaborate, the 3A reactor recirculation system pump motor and the seals on the 3A and 3B pumps, performed a 12 year overhaul on the Unit 3 emergency diesel generator, overhauled numerous control rod drive hydraulic control units, performed eddy current testing on the 3B LPCI heat exchanger and the isolation condenser, and replaced the steam dryer.

The inspectors routinely reviewed the outage schedule and outage risk assessment to verify the licensee was correctly maintaining required equipment in service in accordance with the overall outage safety assessment. During the planned outage, the inspectors performed the following activities:

- attended control room operator and outage management turnover meetings to verify that the current shutdown risk status was well understood and communicated,
- performed walkdowns of containment to identify any indications of unidentified leakage,
- ensured that the control room operators adhered to the licensee’s TSs,
- performed walkdowns of the main control room to observe the alignment of systems important to shutdown risk,
- reviewed selected issues that the licensee entered into the corrective action program to verify that identified problems were being entered into the program with the appropriate characterization and significance,
- ensured that the licensee appropriately considered risk factors during the development and execution of planned activities,
- monitored the licensee’s troubleshooting efforts for emergent plant equipment issues,
- performed plant walkdowns to observe ongoing work activities,
- observed control rod withdrawals and initial transition to criticality,
- performed a walkdown of containment prior to closure to ensure that debris had not been left that could affect the performance of the containment sumps, and
- monitored mode switch changes and observed portions of power ascension.

This represented one inspection sample.

b. Findings

No findings of significance were identified.

.2 Unit 2 Maintenance Outage

a. Inspection Scope

The licensee conducted a maintenance outage on Unit 2 from November 16, 2006, through November 26, 2006. During the outage the licensee replaced the seals on the 2A and 2B recirculation pumps, repaired the 2D1 feedwater heater extraction steam piping and repaired numerous condenser circulating water tubes. The inspectors routinely reviewed the outage schedule and outage risk assessment to verify the licensee was correctly maintaining required equipment in service in accordance with the overall outage safety assessment. During the planned outage, the inspectors performed the following activities:

- attended control room operator and outage management turnover meetings to verify that the current shutdown risk status was well understood and communicated;
- performed walkdowns of containment to identify any indications of unidentified leakage;
- ensured that the control room operators adhered to the licensee's TSs;
- performed walkdowns of the main control room to observe the alignment of systems important to shutdown risk;
- reviewed selected issues that the licensee entered into the corrective action program to verify that identified problems were being entered into the program with the appropriate characterization and significance;
- ensured that the licensee appropriately considered risk factors during the development and execution of planned activities;
- monitored the licensee's troubleshooting efforts for emergent plant equipment issues;
- performed plant walkdowns to observe ongoing work activities;
- observed control rod withdrawals and initial transition to criticality;
- performed a walkdown of containment prior to closure to ensure that debris had not been left that could affect the performance of the containment sumps; and
- monitored mode switch changes and observed portions of power ascension.

This represented one inspection sample.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

.1 Routine Inspections

a. Inspection Scope

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 TSs.

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:

- DOS 7100-06, "In-Service Testing Seat Leakage Test HPCI [high pressure coolant injection] Inlet to FW [feed water] Check Valve," Revision 02;
- DOS 2300-04, "HPCI [high pressure coolant injection] Testable Check Valve Manual Full Stroke Operability Test," Revision 11;
- DOS 7000-22, "Local Leak Rate Testing of Primary Containment Isolation Valves," Revision 4, Test of Unit 3 Containment Ventilation System Valves (3-1601-24, 3-1601-63 & Slip-on Flange) ;
- DOS 7000-08, "Local Leak Rate Testing of Primary Containment Isolation Valves, "Revision 4, Test of Unit 3 Reactor Pressure Vessel Head Cooling Valves (3-205-24 & Blind Flange);
- DOS 0250-07, "Electromatic Relief Valve Testing With The Reactor Depressurized," Revision 01; and
- Unit 2 DOP 2000-24, "Drywell Sump Operation," Revision 13.

This represented a total of six inspection samples, of which one was In-Service Testing, two were Isolation Valves, one was Reactor Coolant System leakage detection (DOP 2000-24), and two were Routine Surveillance tests.

b. Findings

(1) Surveillance Procedures Can Not Be Performed as Described

Introduction: The inspectors identified an Unresolved Item involving TS 5.4.1 regarding the performance of HU-AA-104-101, "Procedure Use and Adherence," Revision 1. Step 4.1.1 of HU-AA-104-101, states to follow the procedure exactly as written and step 4.2.1.1 of HU-AA-104-101 states that if a procedure can not be performed as written, a Procedure Change Request (PCR) or other appropriate action should be initiated and that the procedure should be revised prior to use. Specifically, the licensee identified that DOS 2300-04, "High Pressure Coolant Injection (HPCI) Testable Check Valve Manual Full Stroke Operability Test," Revision 11, and that DOS 7100-06, "Seat Leakage Testing of Valves 2(3)-2301-7, 2(3)-220-59, and 2(3)-1201-158," Revision 02, could not be performed as described.

Description: On November 18, 2006, the inspectors reviewed the completed surveillance test document related to manually stroking the HPCI testable check valve 3-2301-7 using DOS 2300-04, Revision 11. The inspectors noted that during these activities the workers identified that DOS 2300-04, step I.9.j could not be performed as written. Step I.9.j stated to retest 2(3)-2301-7 per step 0 thru 0 above and no step 0 existed. These errors existed when the surveillance was performed. The workers hand-corrected the steps, to state, "retest 2(3)-2301-7 per step I.1 thru I.8 above." The inspectors had three issues with this action:

1.) Was the change made to the procedure within the requirements for a procedure change made in the field?

2.) Was the change made to the procedure technically correct?

3.) Did the change made to the test procedure result in pre-conditioning the test results?

This was an Unresolved Item **(URI 05000249/2006011-03)** pending answers to the above questions.

In addition, after completion of the surveillance test, an issue report was not documented nor was a procedure change request created prior to the next use. The inspectors raised the concern to the licensee staff.

The inspectors identified other minor issues and communicated them to the licensee. The licensee agreed with the inspectors and entered these issues into their Corrective Action Program as IRs 559810, 576208, and 576209.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

.1 Plant Walkdowns and Radiation Work Permit (RWP) Reviews

a. Inspection Scope

The inspectors selectively reviewed the licensee's access controls and survey data for a variety of ongoing refueling outage work areas located within radiation areas, high radiation areas and locked high radiation areas in the plant to determine if the radiological controls, postings and barricades were adequate. These areas included the Unit 3 drywell, the Unit 3 reactor water cleanup system pump room and general areas throughout the Unit 2 and 3 Reactor Buildings including the refuel floor. The inspectors also walked down and surveyed (using an NRC survey meter) selected areas in the Unit 2 and 3 Reactor, Turbine and Radwaste Buildings to determine if radiological conditions were consistent with area postings and controls.

These reviews represented one inspection sample.

b. Findings

No findings of significance were identified.

2OS2 As-Low-As-Is-Reasonably-Achievable (ALARA) Planning and Controls (71121.02)

.1 Inspection Planning

a. Inspection Scope

The inspectors reviewed plant collective refueling outage exposure history, current exposure trends for the Unit 3 refueling outage (D3R19) and ongoing outage activities in order to assess current dose performance and exposure challenges. This included determining the plant's current 3-year rolling average for collective exposure in order to provide a perspective of significance for any resulting inspection finding assessment.

The inspectors reviewed the overall D3R19 work and the associated exposure (dose) projections, time/labor estimates and historical dose data focusing on the following work activities which were likely to result in the highest personnel collective exposures or were otherwise radiologically significant activities:

- Drywell Main Steam Safety and Electromatic Valve Maintenance Activities;
- Drywell In-Service-Inspection Activities;
- Drywell Recirculation Pump and Motor Maintenance;
- Drywell Nuclear Instrumentation Maintenance; and
- Reactor Disassembly /Reassembly.

The inspectors determined site specific trends in collective dose based on plant historical exposure for similar work activities and source term data. The inspectors reviewed procedures associated with maintaining occupational exposures ALARA and evaluated those processes used for D3R19 to develop dose projections including time/labor estimates, and to track work activity specific exposures.

These reviews represented four inspection samples.

b. Findings

No findings of significance were identified.

.2 Radiological Work Planning

a. Inspection Scope

The inspectors obtained the licensee's list of D3R19 refueling outage work ranked by estimated exposure, and based on dose performance through the first-half of the outage reviewed the following radiologically significant D3R19 work activities:

- Reactor Disassembly/Reassembly (RWP 10006807);
- Drywell Main Steam Safety, Electromatic and Target Rock Valve Maintenance Activities (RWP 10006770);
- Drywell In-Service-Inspection Activities (RWP 10006781);
- Drywell Recirculation Pump and Motor Maintenance (RWP 10006780);
- Turbine and Generator Maintenance Activities (RWP 10006801);

- Reactor In-Vessel Inspections (RWP 10006809); and
- Main Condenser Maintenance (RWP 10006800).

For each of the activities listed above, the inspectors reviewed the RWP, the ALARA Plan including specific task plan time/labor estimates and associated total effective dose equivalent (TEDE) ALARA evaluations (i.e., respirator evaluations), as applicable. The reviews were performed in order to determine if the licensee had established radiological engineering controls and dose mitigation criteria that were based on sound radiation protection principles in order to achieve occupational exposures that were ALARA. This also involved determining that the licensee had reasonably grouped the radiological work into activities that were based on historical precedence, industry norms, and/or special circumstances.

The inspectors compared the exposure results achieved through approximately 3-quarters of the 17-day refueling outage including the person-rem expended with the doses projected in the licensee's ALARA planning for the above listed work activities and for other selected outage activities. Reasons for inconsistencies between intended (projected) and actual work activity doses as well as time/labor differences were examined to determine if the activities were planned reasonably well and to ensure the licensee was cognizant of and evaluated any work planning deficiencies. The impact of moisture carryover problems and the licensee's actions to address those issues were reviewed to determine whether the licensee took appropriate actions consistent with the requirements to maintain doses ALARA.

The interfaces between radiation protection and maintenance groups were reviewed to identify potential interface problems. The integration of ALARA requirements into work procedures and RWP documents was evaluated to determine whether the licensee's radiological job planning would reduce dose.

The inspectors compared the person-hour estimates provided by maintenance planning and contractor craft groups to the radiation protection ALARA staff with the actual work activity time expenditures in order to evaluate the accuracy of these time estimates.

The inspectors reviewed the shielding applications completed for the refueling outage and their associated effectiveness, including the shielding designed for the old steam dryer which is being temporarily stored in the dryer/separator pit. The inspectors also reviewed the work activity planning for the outage to determine if it included consideration of the benefits of dose rate reductions through system flushing, water filled components, and job sequencing in order to maximize its ALARA effectiveness.

Work-In-Progress ALARA Reports were reviewed by the inspectors for those outage jobs that approached or exceeded their respective dose estimates or that were otherwise generated to document problems, to identify changes in work scope or to document variances in estimated versus actual doses. These reports were reviewed to determine if the licensee could identify problems at an early stage and address them adequately as the work progressed.

These reviews represented nine inspection samples.

b. Findings

No findings of significance were identified.

.3 Verification of Dose Estimates and Exposure Tracking Systems

a. Inspection Scope

The inspectors reviewed the licensee's assumptions and basis for its collective refueling outage exposure estimate and for individual outage job estimates, and evaluated the methodology and practices for projecting work activity specific exposures. This included evaluating both dose rate and time/labor estimates for adequacy compared to historical station specific or industry data.

The inspectors reviewed the licensee's process for adjusting outage exposure estimates when unexpected changes in scope, emergent work or other unanticipated problems were encountered which could significantly impact worker exposures. This included determining that adjustments to estimated exposure (intended dose) were based on sound radiation protection and ALARA principles and not adjusted to account for failures to effectively plan or control the work. Several outage jobs that had exceeded or could exceed the SDP collective dose thresholds (5 person rem criterion and greater than 50 percent of the planned dose) were evaluated to determine the extent of any ALARA issues. The frequency and scope of collective dose adjustments was also reviewed to evaluate the adequacy of the original ALARA planning.

The licensee's exposure tracking system was examined to determine whether the level of exposure tracking detail, exposure report timeliness, and exposure report distribution was sufficient to support control of outage work exposures. Radiation work permits were reviewed to determine if they covered an excessive number of work activities to ensure they allowed work activity specific exposure trends to be detected and controlled. During the conduct of exposure significant work, the inspectors evaluated if licensee management was aware of the exposure status of the work and would intervene if exposure trends increased significantly beyond exposure estimates.

These reviews represented three inspection samples.

b. Findings

No findings of significance were identified.

.4 Job Site Inspections and ALARA Controls

a. Inspection Scope

The inspectors observed a variety of ongoing outage work activities including in-vessel visual inspections, drywell safety relief valve maintenance, drywell in-service inspections, and reactor water cleanup system valve maintenance to assess the adequacy of the ALARA initiatives and the job specific radiological controls.

The licensee's use of ALARA controls for these work activities was evaluated to determine whether:

- The licensee developed and effectively used engineering controls to achieve dose reductions and to verify that the controls were consistent with the licensee's ALARA reviews.
- Workers were cognizant of work area radiological conditions, utilized low dose waiting areas and that radiological oversight of work was adequate.

These reviews represented two inspection samples.

b. Findings

No findings of significance were identified.

.5 Radiation Worker and Radiation Protection Technician Performance

a. Inspection Scope

Radiation worker and radiation protection technician performance was assessed by the inspectors during observed work activities performed in radiation and high radiation areas focusing on work activities in the Unit 3 drywell. The inspectors determined whether workers demonstrated the ALARA philosophy in practice by being familiar with the work activity scope, the tools to be used for the job, by utilizing low dose waiting areas and to determine if workers had knowledge of the radiological conditions and adhered to the ALARA requirements for the work activity. Job support and the communications provided by the radiation protection staff were also evaluated by the inspectors.

This review represented one inspection sample.

b. Findings

No findings of significance were identified.

.6 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed the results of an outage readiness self-assessment and the results of Nuclear Oversight Department field observations and an audit of the radiation protection program to assess the licensee's ability to identify and correct problems.

The inspectors determine if identified problems were entered into the corrective action program for resolution, and that they had been properly characterized, prioritized, and were being addressed.

Corrective action assignment reports (s) generated during the refueling outage that were related to the radiation protection program were reviewed by the inspectors and licensee staff members were interviewed to assess whether follow-up activities were being conducted in a timely manner commensurate with their importance to safety and risk using the following criteria:

- 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 effective corrective actions.

These reviews represented three inspection samples.

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety

2PS1 Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems (71122.01)

.1 Radiological Impact of Radwaste Equipment Leak

a. Inspection Scope

The inspectors reviewed the in-plant radiological controls and evaluated the potential for any environmental impact associated with a process stream leak in the radwaste demineralizer system which the licensee identified in July 2006. The leakage flooded the radwaste demineralizer vault (demineralizer tank room) located in the radwaste building basement with several feet deep of radioactively contaminated sludge-like slurry. The licensee had similar leakage issues with the radwaste demineralizer system and with the concentrator waste system in the 1990s. As a result of these previous problems, the concentrator waste tank vault which is located adjacent to the radwaste demineralizer vault was also flooded with several feet deep of now hardened sludge.

The inspectors evaluated the potential for any environmental impact (seepage out from the vaults to the environment) and to assess compliance with 10 CFR 20.1501. Additionally, since the leak is ongoing and the demineralizer system continues to be used to process radwaste, the interim compensatory measures being taken by the licensee to ensure the level of the sludge slurry is maintained at a constant level and remains significantly below the level of the groundwater table was reviewed. Specifically, the inspectors reviewed the licensee’s environmental impact evaluation which considered the height of the groundwater table, the level of flooding in the vaults and the resultant hydrostatic pressure gradients which demonstrated the lack of any seepage to the environment. The inspectors also reviewed the in-plant radiological controls in areas surrounding the open demineralizer vault and surrounding areas inside the radwaste building to determine if the posting and the radiological access controls satisfied the high and locked high radiation area requirements of 10 CFR 20 and the licensee's TSs. Additionally, the inspectors discussed with the licensee its plans and timetable for remediation of both the radwaste demineralizer and concentrator waste vaults.

These reviews represented one inspection sample.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Reactor Safety and Radiation Safety Strategic Areas

a. Inspection Scope

The inspectors sampled licensee submittals for the performance indicators (PIs) listed below for the periods indicated. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in Revision 4 of Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," were used. The following PIs were reviewed:

Cornerstone: Barrier Integrity

- Reactor Coolant System Specific Activity

The inspectors reviewed Chemistry Department records including isotopic analyses for 2005 through October 2006, for Units 2 and 3, to determine if the greatest dose equivalent iodine (DEI) values determined during steady state operations corresponded to the values reported to the NRC. The inspectors also reviewed selected DEI calculations including the application of dose conversion factors as specified in plant TSs. Additionally, the inspectors accompanied a chemistry technician and observed the collection and preparation of a reactor coolant system sample to evaluate compliance with the licensee's sampling procedure protocols. Further, sample analyses and calculation methods were discussed with chemistry staff to determine their adequacy relative to TSs, licensee procedures and industry guidelines.

These reviews represented two inspection samples.

Cornerstone: Occupational Radiation Safety

- Occupational Exposure Control Effectiveness

The inspectors reviewed the licensee's assessment of the PI for occupational radiation safety, to determine if indicator related data was adequately assessed and reported for the period July 2005 through July 2006. To assess the adequacy of the licensee's PI data collection and analyses, the inspectors discussed with radiation protection staff the scope and breadth of its data review and the results of those reviews. The inspectors independently reviewed electronic dosimetry dose rate and accumulated dose alarm reports, the dose assignments for any intakes that occurred during the time period reviewed and the licensee's AR database along with individual ARs generated during the period reviewed to verify there were no potentially unrecognized occurrences. The inspectors also conducted walkdowns of accessible locked high radiation area entrances to verify the adequacy of controls in place for these areas.

These reviews represented one inspection sample.

b. Findings

No findings of significance were identified. However, the inspectors reviewed the adequacy of the licensee's evaluation of abnormal radiological restricted area exit electronic dosimetry transaction records. Specifically, the records for a condition identified as "Electronic Dosimetry Digi Reset" were reviewed. Based on the licensee's assessment, each "Digi Reset" condition represented an event that results in a non-functioning dosimeter for a period of time ranging up to approximately 15-minutes. During the digi reset condition, the dosimeter does not function and consequently does not continuously integrate the radiation dose rate in the area and would not alarm when a preset integrated dose was received. Therefore, this condition could represent an occurrence in the Occupational Radiation Safety PI as defined in the Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline." At the time of this inspection, the licensee had not fully determined the extent of the issue, had not identified each specific instance when this problem occurred as necessary to assess compliance with TS High and Locked High Radiation Area requirements. The licensee planned to perform additional investigations to determine the specific instances when the "Digi Reset" problem occurred, to quantify the duration that the dosimeter was not functioning, the amount of dose that was not integrated, and complete its evaluation for compliance with the requirements specified in TS 5.7 "Administrative Controls for High Radiation Areas." Pending completion of the licensee's evaluation and further NRC review of this generic matter, the issue is categorized as an Unresolved Item (URI). **(URI 05000237/2006011-04; 05000249/2006011-04)**

.2 (Closed) Temporary Instruction 2515/169: Mitigating Systems Performance Index Verification

a. Inspection Scope

During the inspection period, the inspectors performed activities associated with verifying that the licensee had correctly implemented the Mitigating Systems Performance Index (MSPI) guidance for reporting the unavailability and unreliability of the following monitored safety systems.

- High Pressure Coolant Injection (Units 2 and 3);
- Isolation Condenser (Units 2 and 3);
- Emergency Diesel Generators (Units 2, 3, and 2/3);

- Low Pressure Core Injection Torus Cooling Mode (Units 2 and 3); and
- Component Cooling Water (Units 2 and 3).

The inspection included a review of the licensee's operations logs, issue reports, and maintenance rule information to determine the baseline unavailability of each of the monitored systems. This information was then compared to the baseline unavailability provided in the licensee's MSPI Bases Document.

After determining the baseline unavailability, the inspectors used the information in the MSPI Bases Document to determine those surveillance activities which the licensee credited as short duration activities or activities that could be recovered by operator action. In each case, the inspectors reviewed the surveillance procedure and historical completion times to ensure that each surveillance activity met the requirements provided in Nuclear Energy Institute Document 99-02, Revision 4, "Regulatory Assessment Performance Indicator Guideline." Lastly, the inspectors reviewed a sample of control room operating logs, issue reports, and maintenance rule information from 2005 and 2006 to determine the actual planned and unplanned unavailability for each system. This information was used to tabulate the actual failure data for each monitored system or component. The inspectors then compared their tabulated unavailability and unreliability information to the data reported by the licensee as part of the Consolidated Data Entry process. The inspectors also confirmed that none of the monitored systems performance limits were exceeded.

b. <u>Findings</u>

No findings of significance were identified. While performing activities associated with verifying the baseline planned unavailability, the inspectors identified numerous mathematical errors within the MSPI Basis Document data. Discussions with licensee personnel resulted in the recalculation of baseline data for the EDGs twice. The final baseline numbers, however, did not result in an MSPI color change. The licensee planned on resolving the MSPI Bases Document issues described above by January 2007. The inspectors planned to review the updated MSPI Bases Document during the 2007 Performance Indicator Review.

Based upon the sample selected, the inspectors determined that the licensee had accurately documented and reported the actual MSPI unreliability data for each monitored component in 2005 and 2006. As a result, no significant errors were identified which resulted in an MSPI color change.

4OA2 <u>Identification and Resolution of Problems</u> (71152)

.1 <u>Routine Quarterly Review</u>

As discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify that they were being entered into the licensee's corrective action system at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. In addition, in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's corrective action program. This review was accomplished by reviewing daily issue reports and attending daily issue report review meetings.

This represents one routine quarterly review.

.2 <u>In-depth Review</u>

<u>Review of Licensee's Response to the Unit 2/3 "B" Standby Gas Treatment System Failure (SBGT)</u>

On September 22, 2006, operations personnel attempted to start the 2/3 "B" SBGT train to support venting the U3 containment. The SBGT train flow started low (2700 scfm) and started to trend down and the heater turned off. Operators executed a manual system shutdown. The problem was identified as the "B" train flow controller which was replaced. The train was returned to operability the same day.

a. <u>Effectiveness of Problem Identification</u>

(1) <u>Inspection Scope</u>

The inspectors reviewed the information provided in IR 534548 and observed the troubleshooting effort first hand 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) <u>Issues</u>

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

b. <u>Prioritization and Evaluation of Issues</u>

(1) <u>Inspection Scope</u>

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

(2) <u>Issues</u>

The licensee had identified the age of the controllers as an issue in 2004 when reviewing preventive maintenance templates. The preventive maintenance template review determined that the controller needed to be replaced every 15 years. The controller was already over 25 years old. The recommended replacement for this controller after the preventive maintenance review was September 2005. The actual replacement date was scheduled for the week of September 17, 2006. However, because the parts were not ordered and the seismic qualification of the parts was not completed in a timely manner, the work was rescheduled for September 2007. Currently the replacement of the "A" train SBGT is scheduled for September 2007. Had the controller preventive maintenance been properly prioritized and completed as originally scheduled the failure would not have occurred. The parts had been received and the seismic qualification was completed prior to installation after the failure on September 22, 2006.

c. Effectiveness of Corrective Actions

(1) Inspection Scope

The inspectors reviewed the licensee's corrective actions which resulted from IR 534548 to determine if the IR addressed generic implications and that corrective actions were appropriate.

(2) Issues

The inspectors reviewed the licensee's actions and concluded that the actions were incomplete. The licensee determined that the cause of the failure was age related since it had previously been identified that the controllers needed to be replaced due to their age. The licensee saw no value in performing an analysis as to why the controller failed. The inspector concluded that the root cause of the controller failure was not the age of the controller but the failure to adequately plan and perform the preventive maintenance already identified by the licensee as necessary. The licensee performed no determination of cause and took no corrective actions to determine why the scheduled maintenance had not been performed as planned. In addition, since no root cause was done on the "B" train controller, it was not determined if a possible issue existed with the "A" train controller.

This represented one inspection sample as an in-depth review.

.3 Semiannual Review for Trends

a. Inspection Scope

As required by Inspection Procedure 71152, Identification and Resolution of Problems, the inspectors performed a review of the licensee's corrective action program (CAP) and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspector's review was focused on fire protection, emergency preparedness (EP) and foreign material exclusion (FME) and consisted of a 6 month period form July 2006 through December 2006. The inspectors reviewed multiple issue reports (IRs) generated during the time period, in an attempt to identify potential trends. The screening was accomplished as follows:

- IRs dealing with company policies, administrative issues, and other minor issues were eliminated as being outside the scope of this inspection;
- The IRs were sorted into categories involving same equipment problems, repetitive issues, reoccurring departmental problem/challenges and repeated entries into TSs. The IRs were then screened for potential common cause issues and considered for potential trends;

- The inspectors removed a group of IRs that discussed strictly programmatic problems because the inspection requirement was primarily for equipment problems and human performance issues;
- The inspectors also removed a group of IRs where their review indicated that duplicate IRs had been written for the same event or failure;
- The remaining groups, considered potential unidentified trends, were provided to the licensee for discussion in case there was extenuating information that the inspectors were not aware of; and
- Groups of IRs remaining after all of the above screening were considered trends which the licensee had failed to identify.
- The inspectors then were able to make an assessment by comparing the trends identified by the licensee to those trends identified by the NRC.

In addition, the inspectors reviewed all of the nuclear oversight assessments and audits conducted during July 2006 to December 2006.

This represented one inspection sample as a semiannual review for trends.

b. Findings

There were no findings of significance identified. The inspectors determined that, within the areas reviewed, licensee employees were writing IRs at an appropriate threshold, and that employees at all levels of the organization were writing IRs. The inspectors determined that the licensee had identified issues adequately and appropriately entered them into the corrective action program.

.4 (Closed) Unresolved Item (URI) 05000237/2006010-05; 05000249/2006010-05, "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"

a. Inspection Scope

The inspectors reviewed requirements regarding use and adherence to procedures and corrective actions implemented by the licensee regarding the performance of DOA 1900-01, step D.1.c and evaluated the adequacy of these actions.

b. Findings

Introduction: The inspectors identified a minor violation of TS 5.4.1 regarding the performance of HU-AA-104-101, "Procedure Use and Adherence," Revision 1. Step 4.2.1.1. of HU-AA-104-101 stated that if a procedure can not be performed as written, a Procedure Change Request (PCR) or other appropriate action should be initiated and that the procedure should be revised prior to use. On January 20, 2006, the licensee identified that DOA 1900-01, "Loss of Fuel Pool Cooling," step D.1.c. could not be performed under a loss of AC power coincident with a loss of coolant accident (LOCA) conditions and did not follow the instructions of HU-AA-104-101.

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. The re-start of the fuel pool cooling system is needed to ensure the assumption that the fuel pool temperature is 125°F or less in calculation DRE97-0214, Revision 1, “Reactor Building Post-LOCA Temperature Analysis” is valid. Because of the post-LOCA radiation levels it would not be desirable for an operator to enter the reactor building to manually close and open valves.

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

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 LOCA conditions. The compensatory action in place directed operations personnel to take actions to ensure DOA 1900-01, step D.1.c. was not used on either unit until a solution to the problem was implemented. Procedure HU-AA-104-101, “Procedure Use and Adherence,” Revision 1, provides guidance on what actions to take if a procedure can not be performed as written.

Analysis: The inspectors determined that the licencee’s failure to perform procedure HU-AA-104-101 was a minor performance deficiency. The failure to maintain adequate procedures could have adversely impacted the functionality of the spent fuel pool cooling system and the operator’s ability to restore fuel pool cooling under a loss of AC power coincident with LOCA conditions. The inspectors determined this performance deficiency to be minor because:

1) the procedure had not actually been performed and therefore there was no actual plant impact due to the failure to change this procedure; and

2) the inability to restart the fuel pool cooling pumps remotely was an item that had been discussed in the control room operators turnover for several months.

This issue was entered into the licensee’s corrective action program as IR 543739. Corrective actions by the licensee included a revision of DOA 1900-01 (Revision 15) to include a note directing operators to perform step D.1.c as soon as possible and before 2 hours have elapsed if conditions for personnel occupancy will or could degrade. Also, guidance is provided on how to promptly start a fuel pool cooling pump if the reactor building may become inaccessible. At the end of this inspection the licensee was still evaluating alternate methodologies for remote fuel pool cooling pump restart.

This URI is closed.

4OA3 Event Followup (71153)

.1 Review of Unit 2/3 Condensate Storage Tank Overflow Event

a. Inspection Scope

The inspectors interviewed operations personnel, reviewed procedures and control room logs, and assessed the information contained in the licensee's corrective action documents to determine the sequence of events which led to the overflowing of the 2/3 condensate storage tank (CST) into the Unit 2 turbine building equipment drain system (TBEDS), which eventually overflowed onto the condensate/condensate booster pump room floor and caused the licensee to enter DOA 0040-02, "Localized Flooding in Plant," Revision 20, for localized flooding.

This inspection represented the completion of one event response sample.

b. Findings

Introduction: A Green finding involving a non-cited violation of TS 5.4.1 was self revealed when during the Unit 3 hotwell drain down to support Unit 3 startup, the 2/3 condensate storage tank (CST) overflowed into the Unit 2 turbine building equipment drain system (TBEDS), which eventually overflowed onto the condensate/condensate booster pump room floor. Operators entered DOA 0040-02, "Localized Flooding in Plant," Revision 20, for Unit 2 localized flooding. The input was stopped shortly after receiving the condensate/condensate booster pump room high water level alarm and the flooding condition stopped. The finding was determined to be of very low safety significance because credited items for shutdown safety were not affected.

Description: On November 20, 2006, Unit 3 was rejecting condensate from the hotwell to the 2/3 CST per DOP 3300-02, "Condensate System Startup," Revision 37, to lower Unit 3 hotwell level in support of Unit 3 startup following D3R19. During this evolution, Unit 2 control room operators were monitoring 2/3 CST level and communicating the rising level to Unit 3 control room operators.

Approximately 10 minutes before reaching the 2/3 CST high-level alarm at 28 ft 6 in, a Unit 2 operator informed Unit 3 personnel to secure hotwell reject, based on expecting to receive the high-level alarm. Due to Unit 3 control room's delay to secure all hotwell reject and establishing alternate 2/3 CST draining lineups in a timely manner, the 2/3 CST overflowed, resulting in Unit 2 condensate/condensate booster pump room high water level. Unit 2 entered DOA 0040-02 for localized flooding.

Based on interviews conducted by the inspectors, it was apparent that there was little or no perceived risks during hotwell reject operations by the Unit 3 operators. Operators explained that they were unfamiliar with the variances between the reject flow rates and the 2/3 CST drain down flow rates during this infrequently performed evolution. The Unit 3 operators felt that once the emergency reject was secured, current CST draining in progress would be sufficient to lower CST level. Unit 3 operators continued with maximum condensate reject flow via the manual bypass valve for 43 minutes after being told the 2/3 CST high-level alarm actuated at 28 feet 6 inches. Operators did not fully understand the consequences of continuing normal reject flow at maximum flow upon receiving the 2/3 CST high-level alarm.

A formal pre-job brief was not conducted to discuss expected alarms, flow rates, and contingency plans to prevent overflow of 2/3 CST during hotwell reject. This evolution required coordination and monitoring by both Unit 2 and Unit 3 personnel. Contingency 2/3 CST draining lineups were not established until after the 2/3 CST high-level alarm was received.

Alarm procedures, DAN 902(3)-6 A-5, "Unit 2/3A Condensate Storage Tank Level HI/LO", Revision 16 and DAN 902(3)-6 A-6, "Unit 2/3B Condensate Storage Tank Level HI/LO", Revision 16 did not adequately address actions and/or contingencies to perform if the high level alarm of 28 feet 6 inches was exceeded. These DANs discussed verifying hotwell reject valves are operating properly, but did not provide guidance on securing reject flow during hotwell drain down for post outage recovery if the 2/3 CST high-level alarm is received.

In addition, DOP 3300-02 neither contained precautions or limitations associated with proper 2/3 CST level and hotwell reject maximum flow rates for hotwell draining in support of unit startup following outage periods, nor contained a note or caution statement related to the operation of the emergency reject valve and expected CST rate of level change.

Operators eventually opened a Unit 3 core spray suction valve to drain water from the CST to the Unit 3 torus to stop the flooding in the Unit 2 condensate pump room. This action was outside the DAN and DOA procedure requirements.

Finally, the 2/3 CST overflowed at a level of 29 feet 3 inches instead of the expected level of 29 feet 10 inches described in DAN 902(3)-6 A-6. The licensee had not performed a calibration of the level instrument by the end of the inspection period so it was unknown if the problem was due to instrument calibration or procedure error.

<u>Analysis</u>: The inspectors determined that the licensee's failure to include essential information in DAN 902(3)-6 A-5 and DAN 902(3)-6 A-6 regarding actions and/or contingencies to perform when the high water level alarm was exceeded was a performance deficiency impacting the Initiating Events cornerstone and warranted a significance evaluation. Using Inspection Manual Chapter (IMC) 0612, Appendix B, "Issue Screening," issued on November 2, 2006, the inspectors determined that this finding was more than minor because it could be reasonably viewed as a precursor to a significant event. The failure to maintain adequate procedures addressing alarm conditions caused the 2/3 CST to overflow into the Unit 2 TBEDS, which eventually overflowed onto the condensate/condensate booster pump room floor and caused the licensee to enter DOA 0040-02 for localized flooding. Unit 2 was shut down at the time of the flooding for a maintenance outage. This event did not affect credited items for shutdown safety. 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. Operators eventually had to take actions outside the procedure guidance to stop the flooding into the Unit 2 condensate pump room.

The inspectors completed a Phase 1 significance determination of this issue using IMC 0609, "Significance Determination Process," Appendix G, Checklist 8, dated May 25, 2004. The inspectors determined that each item on the checklist was being met. Therefore, no quantitative assessment was required and the finding screened as Green (very low safety significance).

Enforcement: The inspectors determined that the licensee's failure to include essential information in DAN 902(3)-6 A-5 and DAN 902(3)-6 A-6 regarding actions and/or contingencies to perform if the high water level alarm is exceeded was a violation of Dresden TS Section 5.4.1, "Procedures." Section 5.4 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 abnormal, off-normal, or alarm conditions are recommended in this regulatory guide.

Contrary to the above, on November 20, 2006, the licensee failed to include pertinent guidance regarding actions and/or contingencies to perform if the high water level alarm for the 2/3 CST is exceeded. This failure resulted in the overflowing of the 2/3 CST into the Unit 2 TBEDS, which eventually overflowed onto the condensate/condensate booster pump room floor. Unit 2 operators received the condensate/condensate booster pump room high water level alarm and entered DOA 0040-02 for localized flooding. This event was entered into the licensee's corrective action program as IR 560445. Corrective actions by the licensee included implementing a standard pre-job brief to address risk and contingencies for this infrequently performed evolution and revising procedures DAN 902(3)-6 A-5, DAN 902(3)-6 A-6 and DOP 3300-02 to include appropriate actions and/or contingencies if the 2/3 CST high water level alarm is exceeded and include limitations and precautions associated with proper 2/3 CST level and hotwell maximum and emergency reject operations. 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 a Non-Cited Violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. **(NCV 05000249/2006011-05)**

4OA5 Other Activities

.1 (Closed) Unresolved Item (URI) 05000237/2006002-03; 05000249/2006002-03, ERV As-Found Testing

During previous refueling outages, the licensee's maintenance practice was that each ERV actuator was refurbished, the pilot and main valves for two ERVs were replaced, and the pilot internals were replaced for the remaining two ERVs. Based on this level of maintenance activity with the valves, the licensee only performed as-left testing, and not as-found testing, on the ERVs to ensure the ERVs would remain operational for the cycle. In January 2004, the licensee submitted a TS amendment to discontinue in-situ testing of the ERVs and the NRC approved this amendment. This amendment submittal also described the licensee's continuing practice of not conducting as-found testing of the ERVs, based on the level of maintenance conducted on the valves.

The TS amendment modified the TS surveillance requirements for Specifications 3.4.3.2, 3.5.1.10, and 3.6.1.6.1 to provide for an alternative means of testing the ERVs. The amendment essentially allowed testing of the ERV actuators and verification of the ERVs' ability to open without cycling the valves with steam pressure while installed in the plant. In the NRC's evaluation of this submittal, the station's current practice of using overlapping tests to verify the functionality of the ERVs was determined to be acceptable and no objection was made to the station's practice of not conducting as-found testing.

The licensee's position was based on the verification of each valve's ability to open at the steam facility; as-left testing after refurbishment of all four ERV actuators, replacement of the pilot internals for two ERVs, installation of two new pilot and main valves for the other two ERVs; and the maintenance history coupled with the inspection of the ERVs during each refueling outage. Due to the discovery of ERV degradation related to extended power uprate operation experienced at the Quad Cities Station, the inspectors questioned whether the licensee's practice of not performing as-found testing of the ERVs was appropriate. The adequacy of the licensee's testing methodology for the ERVs was considered an unresolved item pending NRC review of this issue.

The inspectors reviewed the Safety Evaluation Report for Dresden TS Amendments 211 to Operating License DRP-19 and 203 to Operating License 203; and Quad Cities Inspection Report 05000254/2006009, 05000265/2006009; Units 1 & 2; Special Inspection for Vibration Induced ERV Failures. The issues that were identified at Quad Cities were with the valve actuators. Excessive vibrations resulted in significant wear to parts of the solenoid actuators that had the potential to prevent the ERV's from manually actuating. The issues did not impact the operation of the main valve itself. The licensee has performed in the past and continues to perform as found testing of the ERV actuators and pilot valve assemblies. The inspectors concluded that the testing performed on the ERVs was appropriate for the circumstances and was in compliance with NRC requirements. This item is closed.

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 January 17, 2007. 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 meeting was conducted for:

- Inservice Inspection (Inspection Procedure 71111.08), with Mr. D. Wozniak and other members of licensee management at the conclusion of the inspection on

November 9, 2006. The inspectors returned proprietary information reviewed during the inspection and the licensee confirmed that none of the potential report input discussed was considered proprietary.

- Occupational Radiation Safety ALARA program inspection during the licensee's Unit 3 refueling outage with Mr. S. Wozniak on November 17, 2006.

4OA7 Licensee-Identified Violation

The following violations of very low safety significance were identified by the licensee and are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy for being dispositioned as NCVs.

Cornerstone: Occupational Radiation Safety

Technical Specification 5.4.1 requires that written procedures be established and implemented for activities provided in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Procedures specified in Regulatory Guide 1.33 include radiation protection procedures for personnel monitoring, which are provided by licensee procedure RP-AA-460-1004, "Unescorted Access to and Conduct in Radiologically Controlled Areas," Revision 0. The procedure requires that workers wear electronic dosimetry at all times when in the Radiologically Controlled Area (RCA). Contrary to these requirements, on November 10, 2006, an individual performed activities on RWP 10006761 inside the RCA (drywell) without wearing an electronic dosimeter. This incident is documented in the licensee's corrective action program as AR 556162. This issue represents a finding of very low safety significance because it did not involve ALARA planning or work controls, there was no overexposure or substantial potential for an overexposure to the worker, nor was the licensee's ability to assess worker dose compromised.

Attachment: Supplemental Information

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. 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
J. Kish, ISI Coordinator
D. Leggett, Nuclear Oversight Manager
J. Miller, NDE Level III
P. O'Connor, Lead License Operator Requalification Training
M. Overstreet, Lead RP Supervisor
E. Rowley, Chemistry
C. Symonds, Training Director

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

05000237/2006011-01 05000249/2006011-01	NCV	Licensee's Failure to Develop a Pre-fire Plan for Fire Zone 8.2.6.A, Elevation 534' (Section 1R05)
05000237/2006001-02	NCV	Failure to Comply with TS 5.5.13 for 125 Vdc Battery Terminal Connection Corrosion and Resistance Measurements (Section 1R15)
05000249/2006011-03	URI	Was Procedure Change Made to Surveillance Test Procedure Correct (Section 1R22)
05000237/2006011-04;	URI	Impact of Nonfunctional Dosimeters on TS 5000249/2006011-04 Specification High Radiation Area Compliance (Section 4OA1)
05000249/2006011-05	NCV	Review of Unit 2/3 Condensate Storage Tank Overflow Event (Section 4OA3)

Closed

05000237/2006011-01 05000249/2006011-01	NCV	Licensee's Failure to Develop a Pre-fire Plan for Fire Zone 8.2.6.A, Elevation 534'
05000237/2006001-02	NCV	Failure to Comply with TS 5.5.13 for 125 Vdc Battery Terminal Connection Corrosion and Resistance Measurements
05000249/2006011-05	NCV	Review of Unit 2/3 Condensate Storage Tank Overflow Event
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 a Loss of Coolant Accident (LOCA) Conditions (Section 4OA2)
05000237/2006002-03	URI	ERV As-Found Testing 05000249/2006002-03 (Section 4OA5)

Discussed

05000237/2006010-04 05000249/2006010-04	URI	Full Flow Testing of the Diesel Driven Flooding Pump at Design Conditions (Section 1R06)

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
-DOP 3700 - M2/E2, "Unit 2 RBCCW System Checklist," Revision 10
-M20, "Diagram of Reactor Building Cooling Water Piping," Revision LI

1R05 Fire Protection
-Fire Pre-Plan U3TB-68, Rev. 6; Unit 3 Turbine Building, 469'6" Elev., Condensate Pumps
-Fire Pre-Plan U2/3DG-105, Rev. 6; Unit 2/3 Emergency Swing Diesel Generator, 517'

1R06 Flood Protection Measures
-DOA 0010-01, "Dresden Lock and Dam Failure," Revision 20
-UFSAR, Section 9.2.5.3.1, "Dam Failure During Normal Plant Operation, Section 9.2.5.3.2, "Dam Failure Coincident with a LOCA [loss of coolant accident]," and Section 9.2.5.3.3, "Seismically Induced Dresden Dam Failure Coupled With a Small Break LOCA
-Design Analysis No. DRE03-0026, "Analysis of the Intake Canal, CCSW Heat Exchanger, and Temporary Pumps Following A Dam Failure and 1" LOCA," Revision 0A
-WO 813176, "D2 SA PM Emergency Diesel Pump (CCSW pump) Operation"
-DOS 1500-19,"Operation of The Dresden Lock and Dam Failure CCSW Emergency Pump," Revision 0
-Quality Assurance Topical Report, Revision 77

1R07 Heat Sink Performance
-WO 750562, RFL PM Clean/Insp/Hydro/Eddy Current 'B' LPCI HX
-DMP 1500-03, Revision 25; Containment Cooling (LPCI) Heat Exchanger Maintenance
-ER-AA-335-1006, Revision 1; Heat Exchanger Electromagnetic Testing (ET) Methodology
-IR 556633, dated 11/11/2006; 3B LPCI HX Finds Unexpected Amount of Clam Shells

1R08 Inservice Inspection (ISI) Activities

Non-Destructive Examination (NDE)
-Exelon Procedure ER-AA-335-016; VT-3 Visual Examination of Component Supports, Attachments, and Interiors fo Reactor Vessels; Revision 3
-GE-PDI-UT-1; PDI Generic Procedure for the Ultrasonic Examination of Ferritic Pipe Welds; Revision 5
-GE-UT-300; Procedure for Manual Examination of Reactor Vessel Assembly Welds in accordance with PDI; Reversion 10
-GE-UT-311; Procedure for Manual Ultrasonic Examination of Nozzle Inner Radius, Bore and Selected Nozzle to Vessel Regions; Revision 15
-Visual Examination (VT-3) Data Sheet. D3R19-028; LPCI Support 3/1/1506-16M-1200D-1003; dated November 8, 2006
-Ultrasonic Examination Data Sheet D3R19-052; HPCI Elbow to Pipe Weld 3/2/2306-24-4; dated November 7, 2006.

-Ultrasonic Examination Data Sheet D3R19-002; RPV Nozzle Inner Radius, N3D-1; dated November 8, 2006
-Ultrasonic Examination Data Sheet D3R19-006; RPV Nozzle to Shell Weld, N3D-2; dated November 8, 2006

NDE Personnel Certifications
Bauman, K. E.
Erickson, S. R.
Ginder, T. M.
Hugh, T. E.
Riccardelli, M. E.

CORRECTIVE ACTION DOCUMENTS
-IR 00447649; Fitzpatrick Torus Support Cracking at HPCI Nozzle; dated January 30, 2006
-IR 00463115; Heat Exchanger Needs Weld Repair; dated March 7, 2006
-IR 00268140; Need to Perform NDE Exam of Stained D/W Penetration; dated October 28, 2004
-IR 00380061; FASA - Program Engineering Dual Credit Taken for ISI Inspection; dated September 30, 2005
-IR 00441661; NDE Liquid Penetrant Exam Reveals Several Linear Indications; dated January 20, 2006

CORRECTIVE ACTION DOCUMENTS BASED ON INSPECTOR ISSUES
-IR00555388; Review and Approval of UT Scan Plans, dated November 11, 2006.
-IR00562896; Illumination Requirements for Performing Visual Examinations, dated November 29, 2006.

DOCUMENTS RELATED TO ISI FLAW EVALUATION
-Visual Examination Data Sheet 04-379, snubber 3-0404A-13 s/n 37733; dated October 26, 2004
-Final Third 10-Year Interval Inservice Inspection (ISI) Summary Report, Letter from D. Bost to NRC; dated March 7, 2005

DOCUMENTS RELATED TO WELDED REPAIRS
-WO 00344167; Disassemble and Inspect 3-1101-43A; dated February 2, 2006
-Welding Procedure; WPS8-8-GTSM; Revision 1; dated August 20, 2003
-Welder Qualification Record; L. Esparza
-ASME Weld Data Record for WO 00344167-01; dated March 8, 2006
-ASME Section XI Repair/ Replacement Plan, 00344167 R/RP 3-06-002; dated January 19, 2006

1R11 Licensed Operator Requalification
-IR 553886, “45 percent Failure Rate On Ops Coaching OBE [Out of the Box Evaluation]

1R15 Operability Evaluations
-IR 518329 “Results of Actuation History Investigation of Larger Bore TR Valve”
-Adverse Condition Monitoring and Contingency Plan, “Unit 2 Degraded Thermal Performance,” Revision 02 dated November 3, 2006
-DOS 0500-06, “APRM Gain Adjustment Factor Verification,” Revision 26

-DAP 14-14, “Control Rod Sequences/Reactivity Maneuvers”
-Dresser Information Sketch No. OS333-0601, Revision 2
-Dresden 3-0203-3B ERV, “Failure and Apparent Cause”
-DOS 0250-07, “Electromatic Relief Valve Testing with the Reactor Depressurized,” Revision 1
-WO 977505, “Rebuild 2 ERV Pilot Valves For the D2M12 Outage,” Tasks 2, 3 ,5, and 6
-WO 977019, “Take ERV Pilot Clearance Measurements During D2M12,” Task 01

1R19 Post-Maintenance Testing
-WO 910771-03,”Install EC 360423 - Upgrade U3 SBLC [standby liquid control] Heat Tracing Controls”
-WO 910771-09, ”Install EC 360423 - Upgrade U3 SBLC [standby liquid control] B Heat Tracing Controls”
-WO 899748, “Standby Liquid Control Trace Functional Test”
-MA-DR-771-402, “Unit 2-4KV Tech Spec Undervoltage and Degraded Voltage Relay Routines,” Revision 05
-ER-AA-520, “Instrument Performance Trending,” Revision 3
-IR 487346, “Replace The Bus 24-1 Degraded Voltage Time Delay Relay”

1R20 Refueling Outage
-MA-DR-771-402, “Unit 2-4KV Tech Spec Undervoltage and Degraded Voltage Relay Routines,” Revision 05
-ER-AA-520, “Instrument Performance Trending,” Revision 3
-IR 487346, “Replace The Bus 24-1 Degraded Voltage Time Delay Relay”

1R22 Surveillance Testing
-DOS 7000-08, Revision 04; Local Leak Rate Testing of Primary Containment Isolation Valves
-DOS 7000-22, Revision 01; Local Leak Rate Testing of Unit 2 (3) Reactor Pressure Vessel Head Cooling Valves
-DOS 7000-31, Revision 04; Local Leak Rate Testing of Unit 2 (3) Containment Ventilation System Valves
-WO 756804; 30 M/RFL TS LLRT Valve 1601-24, 63, 2599-5A, 5B & Flanges
-WO 756384; 30 M/RFL TS LLRT Valve 3-0205-27 & Blind Flange
-IR 554904 dated 11-07-2006; MSIV 3-0203-2A LLRT Results Exceed Tech Spec Limit <34 SCFH
-IR 555023 dated 11-08-2006; LLRT Flange Not Tightened on Installation

2OS1 Access Control to Radiologically Significant Areas; and
2OS2 ALARA Planning And Controls

-RP-AA-210; Dosimetry Issue, Usage and Control; Revision 6
-RP-AA-460; Controls for High and Very High Radiation Areas; Revision 10
-RP-AA-401; Operational ALARA Planning and Controls; Revision 4
-D3R19 Outage RWP Preparation Log and Associated Job Dose Projections and Daily RWP Doses; Dose Reports for November 13 - 17, 2006
-RWP 10006809 (Revision 0) and Associated ALARA Plan; D3R19 Refuel Floor IVVI Activities
-RWP 10006765 (Revision 1); Associated ALARA Plan and TEDE ALARA Evaluation; D3R19 Drywell Nuclear Instrumentation System Maintenance
-RWP 10006770 (Revision 0); Associated ALARA Plan and TEDE ALARA Evaluations; D3R19 Drywell Safety/Electromatic/Target Rock Valve Maintenance

-RWP 10006780 (Revision 2); Associated ALARA Plan and TEDE ALARA Evaluations; D3R19 Drywell ‘A’ Recirc Pump and Motor Maintenance
-RWP 10006781(Revision 1) and Associated ALARA Plan; Drywell In-Service-Inspection
-RWP 10006807(Revision 0); Associated ALARA Plan and TEDE ALARA Evaluations; Reactor Disassembly/Reassembly
-ALARA Work-In-Progress Reviews for RWP 10006809; D3R19 Refuel Floor Reactor In-Vessel Inspection Activities; dated November 13 and 15, 2006
-ALARA Work-In-Progress Reviews for RWP 10006770; Drywell Main Steam Valve Maintenance; undated ongoing reviews for period November 7 - 15, 2006
-ALARA Work-In-Progress Reviews for RWP 10006781; Drywell In-Service-Inspection; dated November 11 and 13, 2006
-ALARA Briefing Checklist; RWP No.10006789; Remove/Replace 3-1201-189; 8:00; dated November 14, 2006
-RWP 10006800 (Revision 0); Associated ALARA Plan, TEDE ALARA Evaluations and Recognized Risk Personnel Contamination Dose Assessment; D3R19 Main Condenser Maintenance, and Tube Staking
-RP-AA-401; Attachment 7; Work-In-Progress Review for RWP 10006800; dated November 12, 2006
-RWP 10006801 (Revision 1); Associated ALARA Plan, TEDE ALARA Evaluations and Recognized Risk Personnel Contamination Dose Assessment; D3R19 Main Condenser Maintenance Activities
-RP-AA-401; Attachment 7; Work-In-Progress Review for RWP 10006801; dated November 7, 2006
-RP-AA-401; Attachment 7; Work-In-Progress Review for RWP 10006801; dated November 8, 2006
-RP-AA-401; Attachment 7; Work-In-Progress Review for RWP 10006801; dated November 14, 2006
-Functional Area Self Assessment (FASA); Assignment Number 501387-08; ALARA Outage; dated October 11, 2006
-AR 559186; Inadequate Evaluation of ED Resets; dated November 16, 2006; NRC Identified
-AR 556162; Operator Violates RWP; dated November 10, 2006
-AR 556567; TLD Finger Ring Lost; dated November 11, 2006
-AR 555597; Contaminated Area Entered Without Protective Clothing; dated November 7, 2006
-AR 554713; RP Identifies Radworker Improvement Area; dated November 7, 2006
-AR 555545; NOS ID Individual Breaching Contaminated Area Boundary; dated November 9, 2006
-AR 554351; NOS ID Issue with Contamination Control; dated November 6, 2006
-AR 555932; White Hoses Stored in Plant with Open Ends; dated November 8, 2006
-AR 556661; NOS ID RP Posting Issue with HCUS; dated November 10, 2006
-AR 553605; Rad Worker Standards; dated November 4, 2006
-AR 555390; NRC Identified Deficiencies in HLA, dated November 8, 2006
-AR 551809; D3R19LL: NOS Identified 2-Minute Drill Enhancement at Drywell; dated November 7, 2006
-NOS Audit; NOSPA-DR-06-2Q; ALARA/Contamination Control; dated June 14, 2006
-NOS Audit; NOSPA-DR-06-3Q; Radiological Postings; dated September 1, 2006
-NOS Audit; NOSPA-DR-06-4Q; FASA Review for NRC Inspection; dated October 20, 2006
-Dresden Moisture Carryover Evaluation; dated February 16, 2005
-AR 555552; D3R19 Impact From Moisture Carryover; dated November 8, 2006

2PS1 Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems

-Evaluation For Water In Radwaste Demineralizer Vault to be Released into Groundwater, Associated History and Environmental Impact; Undated
-AR 520217; Unit 2 and 3 Waste Demin Vault Leaking During Processing; dated August 12, 2006
-AR 530124; Waste Demin Vault Filled with Water; dated September 8, 2006

4OA1 Performance Indicator Verification

-AR 455274; Unexpected ED Alarm; dated February 17, 2006
-AR 473188; Unit 3 RWCU Pump Room Gate Locking Mechanism; dated March 31, 2006
-ED Alarm and ED Transaction Reports; Selected data for July 2005 - July 2006
-AR Database for all Radiological Issues; July 2005 - July 2006

4OA2 Identification and Resolution of Problems

-IR 492690; The Fire Main has Developed a Leak near the IDNS Building
-IR 506500; Inspect Piping for Blockage
-IR 506520; Test Results Indicate Pipe Blockage
-IR 521691; Through Wall Leak on U1 Fire Header
-IR 529696; Through Wall Leak Identified in U3 Fire Header
-IR 555521; Fire Header Thru-Wall Leak Approx 1 Drop/Min.
-IR 499336; Gap between Ceramic Blanket Fire Blanket and Pen. Sleeve
-IR 499343; Cables for Valve 2-1402-24B Bypass Fire Break
-IR 499603; Ceramic Fiber has Fallen below the Ceiling
-IR 499608; Hose Clamp on Ceramic Fiber at Penetration Painted Over
-IR 499809; No Fire Barrier Installed in AEER Penetrations
-IR 499956; Fire Barrier F-117 # 8 is Degraded
-IR 500397; No Fire Barrier Installed in Penetrations F-128-24 & 18
-IR 500707; Conduit Missing at Penetration by Barrier
-IR 500713; Cables Run Through Pen without Fire Barrier Installed
-IR 500719; Pen with Electrical Cables has No Fire Barrier Installed
-IR 500842; Penetrations Fire Barriers Operable but Degraded
-IR 500844; Fire Barrier in Penetration F-156-12 Degraded
-IR 500872; Noncombustible Material Not Installed in Pen's as Req by FHA
-IR 503774; U3 HPCI RM N. Wall Pen Not Shown on Dwg No Fire Barrier Inst
-IR 504704; Unable to Check Penetrations for Fire Barriers
-IR 514838; Penetrations F-156-11 & 25 are Operable but Degraded
-IR 504507; Duty Team RPM Did Not Receive Call Out Page for EP Drill
-IR 511511; On-Duty ERO Members Not Responding to Pager Test
-IR 511521; Les Than 60 percent of Dresden ERO Responded to Pager Test
-IR 515781; Muster Attendance – OPS Communicator
-IR 518838; ERO Personnel Not in Attendance at EP Drill
-IR 522106; ERO On-Duty Personnel Not Responding to ERO Pager Test
-IR 522366; Poor ERO Team Response to 08/19/2006 Pager Test
-IR 524149; Did Not Receive GSEP Page on 8/26/06
-IR 524418; Announced ERO Pager Test Provides Poor Results
-IR 528133; Duty Personnel Not Responding to Pager Test

-IR 529598; NOS Identified a Continuing Negative Trend with EP Activities
-IR 545208; NOS Rates EP Yellow for 3rd Qtr 2006
-IR 521558; Combustible Loading Calculation Updates Not Processed
-IR 543230; Resin Barrels Found Without Transient Combustibles Permit
-IR 549473; NOS IDs Combustible Materials Stored in No Combustible Area
-IR 553453; No Transient Combustible Permits
-IR 559932; Transient Combustible Permit Not Posted at the Job Site
-IR 560020; Improperly Stored 55 Gallon Drums of Oil in the Plant
-IR 506171; U1 DFP Day TK Hi Level
-IR 508341; U1 DFP Day Tank Level Indication is Broken
-IR 514207; New Unit 1 DFP Does Not Meet Pump Curve Flow
-IR 507580; U1 DFP Failed to Start Following Overhaul
-IR 523542; U1 DFP Shutdown Due to Smoking Packing Again

LIST OF ACRONYMS USED

ADAMS	Agencywide Documents Access and Management System
AC	Acceptance Criteria
ALARA	As-Low-As-Is-Reasonably-Achievable
AR	Assignment Report
ASME	American Society of Mechanical Engineers
CFR	Code of Federal Regulations
D3R19	Nineteenth Refueling Outage for Dresden Unit 3
DEI	Dose Equivalent Iodine
DOP	Dresden operating Procedure
DOS	Dresden Operating Surveillance
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
Hx	Heat Exchanger
IEMA	Illinois Emergency Management Agency
IMC	Inspection Manual Chapter
IR	Inspection / Issue Report
ISI	Inservice Inspection
LLRT	Local Leak Rate Testing
LPCI	Low Pressure Coolant Injection
M/RFL	Monthly/Refuel
MWe	megawatts electrical
No.	Number
NDE	Nondestructive Examination
NCV	Non-Cited Violation
NUREG	Nuclear Regulatory Guide
NRC	Nuclear Regulatory Commission
PARS	Publicly Available Records
PI	Performance Indicator
PM	Preventive Maintenance Task
RFL	Refuel
RWP	Radiation Work Permit
SBLC	Stand By Liquid Control System
SCFH	Standard Cubic Feet per Hour
SDP	Significance Determination Process
TEDE	Total Effective Dose Equivalent
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved item
UT	Ultrasonic Examination
VT	Visual Examination
WO	Work Order