October 27, 2004

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 05000237/2004010;
05000249/2004010

Dear Mr. Crane:

On September 30, 2004, the NRC completed an inspection at your Dresden Nuclear Power Station, Units 2 and 3. The enclosed report presents the inspection findings which were discussed with Mr. D. Bost and other members of your staff on October 8, 2004.

The inspection examined activities conducted under your license as they relate to safety and to 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, five NRC identified findings and four self-revealed findings of very low safety significance were identified. Seven of these findings were determined to involve 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 seven findings as Non-Cited Violations, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. Additionally, a licensee-identified violation which was determined to be of very low safety significance is listed in Section 4OA7 of this report.

If you contest any Non-Cited Violation in this report, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-001; with copies to the Regional Administrator, Region III; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-001; and the NRC Resident Inspector at the Dresden facility.

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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 05000237/2004010; 05000249/2004010
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
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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:	05000237/2004010; 05000249/2004010
Licensee:	Exelon Generation Company
Facility:	Dresden Nuclear Power Station, Units 2 and 3
Location:	6500 North Dresden Road Morris, IL 60450
Dates:	July 1, 2004, through September 30, 2004
Inspectors:	C. Phillips, Senior Resident Inspector M. Sheikh, Resident Inspector P. Pelke, Reactor Engineer W. Slawinski, Senior Radiation Specialist B. Palagi, Senior Operations Engineer J. Neurauter, Reactor Engineer B. Dickson, Senior Resident Inspector, Clinton D. Eskins, Resident Inspector, LaSalle R. Schulz, Illinois Emergency Management Agency
Approved by:	M. Ring, Chief Branch 1 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000237/2004010; IR 05000249/2004010, 07/01/2004 - 09/30/2004, Exelon Generation Company, Dresden Nuclear Power Station, Units 2 and 3; Flood Protection, Outage Activities, Radiological Environmental Monitoring and Radioactive Material Control Program, and Event Followup.

This report covers a 3-month period of baseline resident inspection and announced baseline inspection of radiation safety. The inspection was conducted by Region III inspectors and the resident inspectors. The inspection identified nine Green findings, seven of which involved Non-Cited Violations. 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 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. Inspector Identified Findings

Cornerstone: Initiating Events

Green. A self-revealed finding of very low safety significance was identified involving several performance issues which resulted in the initiation of a Unit 2 manual scram on April 24, 2004, due to failure of the 2A recirculation pump motor. The performance issues included an inadequate process for rewinding the 2A recirculation pump motor when it was installed in 1999, an inadequate evaluation of the testing of the motor before installation, and the failure to perform post maintenance testing of the reactor building closed cooling water system piping to identify leakage. This failure resulted in the deposit of a conductive substance inside the motor. The licensee identified a number of corrective actions including replacing the 2A recirculation pump motor and revising Exelon Nuclear Engineering Standard NES-EIC-40.01 to include enhanced testing requirements.

The finding was more than minor because it affected the initiating events cornerstone objective to limit the likelihood of an initiating event. The finding was determined to be of very low safety significance because all equipment and systems operated as designed during the scram. (Section 4OA3)

Cornerstone: Mitigating Systems

Green. A finding of very low significance was identified by the inspectors on June 5, 2004, involving a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." The abnormal operating procedure instructions for response to external flooding, and surveillance test procedure for the diesel driven pump necessary to provide make-up to the isolation condenser for response to external flooding, were not adequate for the circumstances. The licensee planned to change the surveillance test procedure and perform a full flow test of the pump in the near future. The licensee planned to review the abnormal operating procedure and revise the procedure as appropriate.

This finding was more than minor because it affected the equipment performance and procedure quality attributes of the mitigating systems cornerstone, and affected the cornerstone objective of ensuring the reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The issue was of very low safety significance based on the low initiating event probability, and because of the slow onset of the flooding and the reduced decay heat in the reactor core at the time recovery actions would be necessary, the licensee would be able to reasonably perform recovery actions that would prevent core damage. (Section 1R06)

Green. A finding of very low significance was identified on July 1, 2004, by the inspectors involving a Non-Cited Violation of Technical Specification 3.3.1.1. The licensee failed to take adequate corrective actions to prevent recurrence of inoperable condenser low vacuum reactor protection system switches, failed to recognize the switches were inoperable, and failed to enter the appropriate Technical Specification Limiting Condition for Operation when the 3C and 2A turbine main condenser low vacuum reactor protection system scram channels were inoperable. The primary cause of the violation was related to the cross-cutting area of Problem Identification and Resolution.

The finding was more than minor because it affected the mitigating systems cornerstone objective by affecting the reliability of the reactor protection system. The finding was determined to be of very low safety significance (Green) because one inoperable channel would not prevent the reactor to scram on low condenser vacuum. Corrective actions by the licensee included installing temporary vent valves on the 3C and 2A sensing lines, enhancing operations training materials, revising the operations's procedure, and performing internal and external condenser walkdowns during the next outage on Unit 2 and Unit 3. (Section 4OA3)

Green. A finding of very low safety significance was identified by the inspectors involving a Non-Cited Violation of Technical Specification 5.4.1. Operators failed to lock manual feedwater isolation Valve 2-220-57B when returning the valve to service. This valve was downstream of where the high pressure core injection (HPCI) system taps into the feedwater line. The inspectors identified this issue during the drywell closeout after the maintenance outage on September 23, 2004. The operators were counseled and the licensee will require out-of-service checklists to be brought into the drywell in the future. The primary cause of this violation was related to the cross-cutting issue of Human Performance.

This issue was more than minor because it was repetitive. Other valves were found unlocked inside the drywell by the inspectors during the drywell close out after the last Unit 2 refueling outage in November 2003. The issue was of very low safety significance because the valve was in the correct position. (Section 1R20)

Green. A self-revealed finding of very low safety significance involving a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified. Inadequate procedural guidance resulted in the failure of electricians to properly set the open torque switch bypass on Valve 2-1301-3, “Isolation Condenser Outboard Condensate Return Valve,” on October 8, 1999. This resulted in the failure of the valve to open during an event that occurred on April 24, 2004. The licensee counseled the individuals and revised the maintenance procedure.

This finding was more than minor because it involved the equipment performance attributes of the mitigating systems cornerstone and affected the cornerstone objective of availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. This issue was of very low safety significance in that the isolation condenser was only being used for pressure control at the time of the event and other methods of pressure control were available, and in addition, the licensee could have manually opened the valve if necessary. (Section 4OA3)

Cornerstone: Barrier Integrity

Green. A finding of very low safety significance was identified by the inspectors involving a Non-Cited Violation of 10 CFR 50.65, “Maintenance Rule,” requirements. The licensee failed to identify that the number of functional failures for the reactor building ventilation system had exceeded the established performance criteria and did not move the reactor building ventilation system into the a(1) category. Once identified, the reactor building ventilation system was moved into the a(1) category on October 8, 2004. The licensee had not yet determined system goals or established corrective actions by the close of the inspection period. The primary cause of the violation was related to the cross-cutting area of Problem Identification and Resolution in that functional failures of the system were not properly entered into the corrective action program.

This issue was more than minor because it involved the design control and barrier performance attributes of the barrier integrity cornerstone; and affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. The issue was of very low safety significance because the licensee was still able to maintain secondary containment. (Section 1R12)

Green. A self-revealed finding of very low safety significance involving a Non-Cited Violation of Technical Specification 3.7.4 was identified on April 28, 2004. The licensee failed to correctly restore the control room emergency ventilation system to operable status following maintenance. This left the control room emergency ventilation system inoperable for greater than its Technical Specification allowed outage time. This finding was self-revealed when the system did not operate properly several days later during a routine system realignment. As corrective action, the licensee revised a procedure to give better guidance on how to remove the temporary modification.

The issue was more than minor because it affected the Barrier Integrity Cornerstone attributes of design and configuration control and the cornerstone objective of protecting persons in the control room from radionuclide releases caused by accidents or events. The issue was of very low safety significance due to the short duration of the condition of the system. (Section 4OA3)

Green. A finding of very low safety significance was identified on August 3, 2004, by the inspectors during the walkdown of a corrective action for a previous event. The licensee had an abnormal operating procedure requirement to have tools and equipment staged to install a temporary modification to keep the control room emergency ventilation system dampers open in the event of an accident. The equipment necessary to install the temporary modification was in various stages of disarray. Some equipment was not labeled and some necessary tools were missing. The licensee identified a number of corrective actions including properly packaging the necessary tools and equipment, revising procedures, and initiating a training request to ensure operations personnel are properly trained in the use of the tools and equipment.

The finding was more than minor because it affected the Barrier Integrity Cornerstone attributes of configuration control and the cornerstone objective of protecting persons in the control room from radionuclide releases caused by accidents or events. The issue was of very low safety significance due to it only impacting the radiological barrier function of the control room emergency ventilation system. This was not a violation of regulatory requirements. (Section 40A3)

Cornerstone: Public Radiation Safety

Green. A self-revealed finding of very low safety significance involving a Non-Cited Violation of 10 CFR 20.1501 was identified on October 16, 2003, following a gatehouse radiation monitor alarm at the Braidwood Nuclear Station upon detecting a discrete radioactive particle (DRP) on a worker’s boot. The DRP was attributed to the worker’s activities at the Dresden facility approximately 1 year earlier. The DRP was not identified at the Dresden Station due to an inadequate radiation survey of the worker following a personnel contamination monitor alarm and also because of limitations with the radiation monitoring instrumentation used at the licensee’s egress to the radiologically controlled area (RCA).

Corrective actions for this finding included tailgate training to radiation protection staff that respond to contamination monitor alarms, improvements to automated radiation monitoring capabilities at the main RCA egress, and actions to enhance gamma-sensitivity of those automated radiation monitors located in alternate egress areas and at the protected area gatehouse.

The finding was more than minor because it was associated with the “Program and Process” and “Human Performance” attributes of the Public Radiation Safety Cornerstone, and affected the cornerstone objective that ensures adequate protection of public health and safety from exposure to radioactive materials that are released into the public domain. The issue represents a finding of very low safety significance because public radiation exposure resulting from the problem was not greater than 0.005 rem total effective dose equivalent, the licensee did not have greater than five radioactive material control occurrences in the previous eight quarters and the dose to the involved worker was approximately one percent of the regulatory (10 CFR 20.1201) occupational dose limits for adults. An associated Non-Cited Violation of 10 CFR 20.1501 was identified for the failure to conduct an adequate survey to ensure proper control of radioactive material as required by 10 CFR Part 20, Subpart I, "Storage and Control of Licensed Material" (Section 2PS3).

B. <u>Licensee-Identified Violation</u>

A violation of very low safety significance which was identified by the licensee has been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. This violation 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 (100 percent of rated electrical capacity).

- On August 13, 2004, a controlled shutdown began for a forced outage for the purpose of repairing a crack on the generator footing. The unit was returned online August 21, 2004. The unit did not achieve full power because of vibration on the #9 turbine bearing.
- On September 18, 2004, a controlled shutdown began for a forced outage for the purpose of repairing the main turbine generator. The unit was returned online September 25, 2004.

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

- On several occasions throughout the inspection period, load was reduced to perform control rod adjustments, with the unit returning to full load during the same day.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R04 Equipment Alignment (71111.04Q&S)

.1 Partial System Walkdowns

a. Inspection Scope

The inspectors selected a redundant or backup system to an out-of-service or degraded train, reviewed documents to determine correct system lineup, and verified critical portions of the system configuration. Instrumentation valve configurations and appropriate meter indications were also observed. The inspectors observed various support system parameters to determine the operational status. 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:

- 3A Core Spray System;
- Unit 2, Division II, Low Pressure Coolant Injection System;
- Unit 2/3, A Train, Standby Gas Treatment System; and
- Unit 3 High Pressure Coolant Injection System.

b. Findings

No findings of significance were identified.

.2 Complete System Walkdown

a. Inspection Scope

The inspectors performed one complete semiannual walkdown of the Unit 3 control rod drive system. The inspectors reviewed the electrical and mechanical system checklist and drawings to ensure all vital components in this system were energized. The inspectors reviewed outstanding work orders associated with the system to determine whether there were any deficiencies that could affect the ability of the system to perform its safety-related function. The inspectors also reviewed all temporary modifications and operator workarounds to verify the operational impact on the system. The inspectors reviewed licensee condition reports (CRs) and issue reports (IRs), to verify past issues that had been identified and their corrective actions.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Routine Inspection (Quarterly)

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. 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 reactor building, elevation 589' isolation condenser area, (Fire Zone 1.1.2.5.A);
- Unit 3 reactor building, elevation 589' isolation condenser area, (Fire Zone 1.1.1.5.A);
- Unit 2 turbine building, elevation 469'-6" condensate pumps (Fire Zone 8.2.1.A);
- Unit 2 turbine building, elevation 495' containment cooling service water pumps (Fire Zone 8.2.2.A);
- Unit 2 turbine building, elevation 517' diesel generator (Fire Zone 9.0A);
- Unit 3 turbine building, elevation 495' containment cooling service water pumps (Fire Zone 8.2.2.B);
- Unit 3 turbine building, elevation 469'-6" condensate pumps (Fire Zone 8.2.1.B);
- Unit 2 reactor building, isocondenser pipe chase (3 valve room), elevation 545'-6" (Fire Zone 1.1.2.5.c); and
- Unit 3 reactor building, elevation 545'-6" (Fire Zone 1.1.1.3).

b. Findings

No findings of significance were identified.

.2 Weekly Fire Marshal Walkdown

a. Inspection Scope

The inspectors accompanied the Unit 3 field supervisor during a weekly fire marshal walkdown of the Unit 2 and Unit 3 reactor and turbine buildings. The inspectors observed that the field supervisor checked hot work activities in progress, general fire protection housekeeping, fire protection equipment was not blocked and appeared to be in good working order, fire doors and dampers were in good condition, and emergency lights appeared to be in good working order.

b. Findings

No findings of significance were identified.

1R06 Flood Protection (71111.06)

a. Inspection Scope

The inspectors reviewed the Updated Final Safety Analysis Report 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.

b. Findings

Introduction: The inspectors identified a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," having very low safety significance (Green) for the failure to develop adequate surveillance test and operating procedures for equipment that was the sole source of makeup water to the isolation condensers for both units during a design basis flood.

Description: Dresden Updated Final Safety Analysis Report (UFSAR) Section 2.4.3 stated that the NRC concluded in Systematic Evaluation Program (SEP) Topic II-3.B that a flow of 490,000 cubic feet per second in the Illinois River would result in a still water flood elevation of 525 feet. Adding wave runup to the stillwater flood elevation yields a site probable maximum flood (PMF) elevation of 528 feet. This is about 11 feet above grade. Water at this height puts all emergency core cooling system equipment under water. Therefore, the sole sources of decay heat removal for both units would be the isolation condensers. After initiation, make up water eventually needs to be added to the shell side of the isolation condenser.

The UFSAR Section 3.4.1.1, "External Flood Protection Measures," stated that if forecasted flood levels exceed 517 feet, 150 gallon per minute emergency makeup pumps are connected to the fire system. The UFSAR did not say why the pumps were to be connected. Procedure DOA 0010-04, “Floods,” Revision 16, stated that a portable diesel driven pump will be brought into the reactor building, hoisted into the air, and connected to the fire main system to supply makeup water to the isolation condensers. The inspectors reviewed the 10 CFR 50.59 screening performed for the procedure change that went from four gasoline powered pumps to one diesel driven pump. The 10 CFR 50.59 screening did not address the change in the number and type of pumps. This is an unresolved item pending NRC review of the licensee’s planned corrective action to perform a 10 CFR 50.59 evaluation and subsequent revision to UFSAR Section 3.4.1.1. **(URI 05000237/2004010-01; 05000249/2004010-01)**

Calculation DRE99-0035, “Capacity and Discharge Head For Portable Isolation Condenser Make-up Pump To Be Used During Flood Conditions,” Revision 2, conservatively assumes that the portable make-up pump to the isolation condensers would be required to remove the decay heat 6 hours after reactor shutdown is reached. The calculation concludes that, in order to supply sufficient makeup flow, the portable pump must be able to supply 174 gallons per minute per reactor at a discharge head of 263 feet. The vendor manual DRE VTIP MANL GC43-001, “Godwin Pumps HL80 M Dri-Prime Pump Operating and Maintenance Manual,” Revision 3, shows in Figure 6 that the pump can produce this amount of flow at the required head, but only at a speed of 2400 rpm.

The NRC identified on June 5, 2002, that this pump was not routinely tested; and therefore, its capability to perform its function during a flood was suspect. This was identified in CR 111005. Assignment 5 from CR 111005 was to “evaluate the preventative maintenance and surveillances that should be performed relating to the emergency diesel pump and other activities associated with DOA 0010-04.” This was completed on September 10, 2002. The licensee developed and implemented DOS 1300-04, “Operation Of The Isolation Condenser External Flood Emergency Make-up Pump,” on April 24, 2003. Per the surveillance test, DOS 1300-04, Revision 2, Step I.2.g, the pump is run at a speed of 1800 rpm. Therefore the surveillance test does not evaluate the ability of the pump to run at the speed (2400 rpm) necessary for the pump to perform its function.

Page 9 of Calculation DRE99-0035 states that the timetable for the licensing basis flood is given in Technical Evaluation Report (TER) C5257-421, “Hydrological Considerations,” dated May 7, 1982. The report concludes that the flood waters will rise from 509 feet to 517 feet in 7 hours assuming dam gates open to 16 feet. The Technical Requirements Manual and Procedure DOA 0010-04 both require that both units shutdown when the river elevation reaches 509 feet. Therefore, the amount of water needed for the isolations condensers is based on decay heat 6 hours after shutdown.

As mentioned above, the required amount of flow to the isolation condenser to remove decay heat 6 hours after shutdown would require a diesel driven pump speed of 2400 gpm. Procedure DOA 0010-04 has no mention as to what speed the diesel driven pump should or can be run. Procedure DOA 0010-04, Step D.14.f stated, “adjust the throttle to increase flow.” The inspectors conducted an interview with members of the non-licensed operator training staff that conducted training on the diesel driven pump to be used during a flood. The inspectors asked if given the flood scenario, at what speed would you operate the pump? One trainer stated that given the surveillance (DOS 1300-04) has the operator test the pump at 1800 rpm he would run the pump at that speed. The inspectors asked if no water was flowing at that speed what would you do? Both trainers responded that the pump would be secured and a valve lineup would be performed.

Procedure DOA 0040-02, "Localized Flooding In Plant," Revision 15, Step A.1.a, stated that for the isolation condenser external flood emergency make-up pump, "The maximum vertical suction lift from water source to pump impeller cannot be in excess of 27 feet." The inspectors requested that system engineering personnel check the length of the suction hose. The system engineer informed the inspectors that the suction hose was 30 feet long. Procedure DOA 0010-04, Step D.9.i, states, "Using the Rx Building Crane, raise the emergency make-up pump to between 12 and 15 feet above the floor;" and Step D.17, states, "<u>WHEN</u> water recedes to below EL 518 ft, <u>THEN</u> relocate suction hose from diesel-driven emergency make-up pump to draw water from the nearest ECCS [emergency core cooling system] corner room." The inspectors verified that the distance between the location of the pump impeller when the pump is hoisted into the air and the ECCS corner room was greater than 30 feet. This means there would be no suction source for the pump once the water started to recede.

Technical Evaluation Report (TER) C5257-421 was prepared for the NRC by the Franklin Research Center to evaluate the effects of flooding on Dresden Unit 2. This TER points out that "normal reactor cooling procedures will not ensue immediately following the time flood waters drop below elevation 509, consequently the operation of the gasoline-driven pumps will be required for a significant period of time, i.e., more than 3 days." The licensee did not estimate how long the diesel driven pump may be needed; and therefore, did not estimate how much fuel for the diesel driven pump needed to be pumped into barrels and staged in the reactor building in advance. The licensee does not know how long the flooding might impact the site and how much fuel might be needed to be staged in advance.

<u>Analysis</u>: The inspectors determined that the failure to implement adequate surveillance test and abnormal operating procedures that provided instructions to ensure an adequate supply of make-up water to the isolation condenser during flood conditions to prevent core damage was a performance deficiency warranting a significance evaluation in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," issued on June 20, 2003. The inspectors determined that the finding was more than minor because it (1) involved the equipment performance and procedure quality attributes of the mitigating systems cornerstone and (2) affected the cornerstone objective of ensuring the reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors also determined that the failure to implement an adequate surveillance test procedure after it was identified that there was no test procedure in 2002, also affected the cross-cutting area of Problem Identification and Resolution.

The inspectors determined that the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Appendix A, dated

September 10, 2004, because the finding was associated with the reliability of a mitigating system. The inspectors concluded that the diesel driven make-up pump would be a mitigating system in the case of the PMF. For the Phase 1 screening, the inspectors answered “No” to the first four questions under the mitigating systems column. The inspectors then went to the Phase 1 worksheet for Seismic, Fire, Flooding, and Severe Weather Criteria. Question 1 was answered “Yes.” Question 2.c was answered “Yes.” Returning to Question 5 under the mitigating systems on the Phase 1 screening sheet this question was answered “Yes” and referred to the Regional Office for a Phase 3 analysis. The Phase 3 analysis performed by the Senior Reactor Analyst (SRA) concluded that the safety significance of this finding based on the change in core damage frequency to be Green. The Phase 3 analysis reviewed the potential failure probabilities based on the procedure and equipment inadequacies. The SRA, in discussions with the licensee and the resident inspectors, determined that based on the low initiating event probability, and because of the slow onset of the flooding and the reduced decay heat in the reactor core at the time recovery actions would be necessary, the licensee would be able to reasonably perform recovery actions that would prevent core damage. A Green finding represents a finding of very low safety significance.

Enforcement: The inspectors identified that the licensee did not have procedures appropriate to the circumstances of external plant flooding on June 5, 2004. The failure to have procedures appropriate for the circumstances of a PMF was a violation of 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” which states in part, “Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances...” Contrary to the above, on June 5, 2004, (1) surveillance procedure DOS 1300-04, Revision 2, for the portable diesel driven isolation condenser make-up pump did not test the diesel at the speed (2400 rpm) necessary to deliver the minimum amount of flow to make up to the isolation condensers in the event of a flood; (2) abnormal operating procedure DOA 0010-04, Revision 16, did not specify the minimum required speed of the diesel driven make-up pump for the operators; (3) the suction hose from the pump was too short to successfully accomplish the step in procedure DOA 0040-02, Revision 15, that directed the operators to move the suction hose to the nearest corner room when the flood waters started to recede; and (4) none of the procedures specified how much fuel oil was necessary to be staged on the 545 foot level of the reactor building prior to the onset of flooding. The licensee planned to change the surveillance test procedure DOS 1300-04 to run the pump at 2400 rpm and perform a full flow test of the pump in the near future. The licensee planned to review DOA 0010-04 and revise the procedure as appropriate. Because this issue is of very low safety significance and has been entered into the licensee’s corrective action program (Issue Reports, 246038 and 261167), this violation is being treated as a Non-Cited Violation, consistent with Section VI.A., of the NRC Enforcement Policy. **(NCV 05000237/2004010-02; 05000249/2004010-02)**

1R11 Licensed Operator Requalification (71111.11Q)

a. Inspection Scope

The inspectors observed an evaluation of an operating crew on September 27, 2004. The scenario consisted of a recirculation flow controller failure, a reactor building closed cooling water pump trip, an instrument line break in the drywell which required flooding of the reactor pressure vessel, and a failure of a core spray pump. The inspectors verified that the operators were able to complete the tasks in accordance with applicable plant procedures and that the success criteria as established in the job performance measures were satisfied. The inspectors observed the licensee's evaluators to ensure that no inappropriate cues were provided by the evaluators while assessing the operators' performance. In addition, the inspectors verified that condition reports written regarding licensed operator requalification training were entered into the licensee's corrective action program with the appropriate significance characterization.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors reviewed the licensee's overall maintenance effectiveness for risk-significant mitigating systems. The inspectors also reviewed whether the licensee properly implemented the Maintenance Rule, 10 CFR 50.65, for the systems. Specifically, the inspectors determined whether:

- the systems were scoped in accordance with 10 CFR 50.65;
- performance problems constituted maintenance rule functional failures;
- the systems have been assigned the proper safety significance classification;
- the systems were properly classified as (a)(1) or (a)(2); and
- the goals and corrective actions for the systems were appropriate.

The above aspects were evaluated using the maintenance rule program. The inspectors also verified that the licensee was appropriately tracking reliability and/or unavailability for the systems.

The inspectors reviewed the following systems:

- Unit 3 Control Rod Drive System;
- Reactor Building Containment Cooling Service Water; and
- Reactor Building Ventilation.

b. Findings

Introduction: The inspectors identified a Non-Cited Violation (NCV) of 10 CFR 50.65 having very low safety significance (Green) for failing to adequately implement the maintenance rule. The licensee failed to move the reactor building ventilation system to category (a)(1) monitoring of 10 CFR 50.65 after exceeding the performance criteria established for reliability of four functional failures in 2 years.

Description: The licensee was not able to maintain a .25-inch vacuum in the secondary containment relative to atmosphere on June 19, 2002, June 24, 2002, July 31, 2002, October 24, 2003, March 7, 2004, April 24, 2004, and July 22, 2004. On each date the differential pressure was positive for less than the Technical Specification (TS) 3.6.4.1.1 Limiting Condition For Operation time. The licensee's maintenance rule program considered a positive pressure in the secondary containment as a functional failure of the reactor building ventilation system. These events were not properly addressed by the licensee's corrective action program. The July 31, 2002 event was documented in the operator's logs but no condition report was written. The April 24, 2004 event was entered into the corrective action program as CR 216750, and was coded as a maintenance rule functional failure but was not counted as a maintenance rule functional failure.

The licensee had not established a repair or replacement program for the reactor building ventilation exhaust fan back-draft damper actuator springs. The springs provided the motive force for the back-draft dampers. The damper vendor recommended replacement of the springs within 10 years. The actuator springs were 18 years old. As a result of spring relaxation, the Unit 2A and 2C exhaust fan back-draft dampers would only open about 5 percent, resulting in a failure to maintain the .25 inch vacuum.

The licensee's performance criteria for the reactor building ventilation system, established in the maintenance rule database program, stipulated that more than four functional failures in 2 years required transfer of the reactor building ventilation system to 10 CFR 50.65 Section (a)(1) monitoring from Section (a)(2). The licensee had five functional failures between July 31, 2002, and July 22, 2004, and six functional failures between June 19, 2002, and April 24, 2004. The licensee did not move the reactor building ventilation system to (a)(1).

Analysis: The inspectors determined that the failure to correctly enter the events into the corrective action program which resulted in the failure to move the reactor building ventilation system from 10 CFR 50.65 Section (a)(2) to (a)(1) was a performance deficiency warranting a significance evaluation in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," issued on June 20, 2003. The inspectors determined that the finding was more than minor because it: (1) involved the design control and barrier performance attributes of the barrier integrity cornerstone; and (2) affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. The inspectors determined that the finding also affected the cross-cutting area of Problem Identification and Resolution.

The inspectors determined that the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situation," dated September 10, 2004, because the finding was associated with a degraded reactor building containment barrier. For the Phase 1 screening, the inspectors answered “Yes” to Question 1 under the Containment Barriers Cornerstone column because the finding only represents a degradation of the radiological barrier function of the reactor building. The finding screened as Green.

Enforcement: Section (a)(2) of 10 CFR 50.65 stated, in part, that monitoring as specified in paragraph (a)(1) is not required where it has been demonstrated that the performance of a structure, system, or component is being effectively controlled through the performance of appropriate preventive maintenance, such that the structure, system, or component remains capable of performing its intended function. The licensee’s performance criteria for the reactor building ventilation system, established in the maintenance rule database program, stipulated that more than four functional failures in 2 years required transfer of the reactor building ventilation system to 10 CFR 50.65 Section (a)(1) monitoring from Section (a)(2). The licensee had five functional failures between July 31, 2002 and July 22, 2004 and six functional failures between June 19, 2002 and April 24, 2004. The functional failures were due to inappropriate preventative maintenance. The licensee did not move the reactor building ventilation system to (a)(1) until the inspectors identified this issue. The licensee convened a maintenance rule evaluation panel on October 8, 2004, and moved the reactor building ventilation system to (a)(1). System goals have not yet been established. Because this violation was of very low safety significance and because it was entered into the corrective action program (Issue Report 256499), this violation is being treated as a Non-Cited Violation, consistent with Section VI.A of the NRC Enforcement Policy. **(NCV 05000237/2004010-03; 05000249/2004010-03)**

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

a. Inspection Scope

The inspectors evaluated the effectiveness of the risk assessments performed before maintenance activities were conducted on structures, systems, and components and verified how the licensee managed the risk. The inspectors evaluated whether the licensee had taken the necessary steps to plan and control emergent work activities. The inspectors also verified that equipment necessary to complete planned contingency actions was staged and available. The inspectors completed evaluations of maintenance activities on the:

- Unit 3, 250 VDC battery jumpering of cell 113;
- Unit 2 high pressure coolant injection room cooler fan emergent work;
- Planned maintenance of 3B core spray;
- Unit 2, Div I, LPCI, replace/regrease auxiliary contact in MCC 28-1, Valves 2-1501-3A/2-1501-18A/2-150138A;
- Unit 2/3, B train, standby gas treatment planned maintenance; and
- Planned maintenance of 3A core spray.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

.1 Routine Operability Evaluation (OE) Reviews

a. Inspection Scope

The inspectors reviewed operability evaluations to ensure that operability was properly justified and the component or system remained available, such that no unrecognized increase in risk occurred. The review included issues involving the operability of:

- Units 2 & 3 High Pressure Coolant Injection (HPCI) May Not be Available for Some Safe Shutdown Events Assumed in the Licensing Basis Due to Water Intrusion into the HPCI Steam Line (OE 04-002);
- Unit 3 Containment Cooling Service Water Back-Up Keep Fill Line is not Supported per Code Allowance (OE 04-006);
- Units 2 & 3 CR105X Dried Grease Could Degrade Operation of Contactor (OE 04-004);
- Units 2 & 3 Electromatic Relief Valve Discharge Flanges (OE 03-013, Rev. 1);
- IR 00245395, "NRC Concern With Reactor Level Density Error;"
- Low Unit 2 Switchyard Voltage on August 23, and September 14, 2004;
- 2(3)-0203-3A Target Rock Safety Relief Valve (OE04-014);
- CRD operability (IR 0257827);
- Secondary Containment, IRs 0238241, 0249994, and 0246489;
- Main Condenser Hood/Bay (OE 04-008, Revision 0); and
- Unit 2 & 3 “2A and 3C Condenser Bay Vacuum Indication/Switch Sometimes Indicates a Non-conservative Value after a Flow Reversal to East-to-West Flow” (OE 04-008, Revision 1).

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modification (7111.17A)

a. Inspection Scope

The inspectors reviewed one permanent plant modification to verify the design adequacy to ensure licensing and design bases were maintained, and to ensure functionality of interfacing structures, systems, and components. The modification reviewed included the following:

- Replacement of Standby Gas Treatment Solenoid Valve 2/3-7541-43B on AO-7510-B.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed post-maintenance test results to confirm that the tests were adequate for the scope of the maintenance completed and that the test data met the acceptance criteria. 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:

- Disassemble and Inspect 2B Core Spray Minimum Flow Stop Check Valve;
- Unit 3, 250 VDC Battery Jumpering of Cell 113;
- Testing of the 3B Core Spray Following 2Y EQ GE Pump Motor Maintenance;
- Unit 2/3, B train, Standby Gas Treatment, Testing of Valve Operator 2/3-7507-B Following Preventive Maintenance on Limitorque; and
- Inspect/Repair Unit 2 HPCI Discharge Testable Check Valve 2-2301-7.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

.1 Unit 2 Forced Maintenance Outage August 2004

a. Inspection Scope

On August 14, 2004, the licensee commenced a 6-day forced maintenance outage. The licensee identified a crack on a weld on one of the main turbine generator support footings. General Electric, while performing a review of Unit 2 main turbine vibrations, informed the station of a concern that the crack could propagate into the generator housing. Since the generator is cooled by hydrogen the result could have possibly led to an escape of hydrogen leading to the potential for a fire and/or explosion. General Electric recommended the shutdown of Unit 2 until the crack could be repaired.

The inspectors verified that the licensee effectively conducted the shutdown, managed elements of risk pertaining to reactivity control during and after the shutdown, and implemented decay heat removal system procedure requirements as applicable.

The inspectors performed the following activities daily:

- attended control room operator turnover meetings to verify that the current shutdown risk status was well understood and communicated;
- 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 its 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 licensee's troubleshooting efforts for emergent plant equipment issues;
- performed plant walkdowns to observe ongoing work activities;
- conducted in-office reviews of selected issues that the licensee entered into its corrective action program to verify that identified problems were being entered into the program with the appropriate characterization and significance;
- observed control rod withdrawals and initial transition to criticality; and
- monitored mode switch changes and observed portions of power ascension.

b. Findings

No findings of significance were identified.

.2 Unit 2 Maintenance Outage September 2004

During the Unit 2 outage in August 2004, the licensee attempted to reduce vibrations on the number 9 turbine bearing by realigning the main generator. That effort was unsuccessful. Unit 2 generator vibrations were still high at a lower power level than those seen previous to the August outage. Unit 2 was shutdown again on September 18, 2004, for a scheduled 26 day outage in order to send the generator offsite to be rewound. Testing during the shutdown convinced licensee management that the actual problem causing the high vibration was soft foot conditions (foundation not as firm as it should be) under three of the eight generator support structures. The licensee repaired the soft conditions and added balancing weights to the main generator rotor. The unit was restarted and returned to full power on September 26, 2004.

The inspectors performed the following activities daily:

- attended control room operator turnover meetings to verify that the current shutdown risk status was well understood and communicated;
- performed walkdowns of the main control room to observe the alignment of systems important to safe/shutdown risk condition;
- monitored licensee's troubleshooting efforts for emergent plant equipment issues, specifically the failure of 2-2301-7, high pressure coolant injection (HPCI) testable discharge check valve to pass its cold shutdown surveillance;
- performed a drywell closeout walkdown;
- observed control rod withdrawals and initial transition to criticality; and
- monitored mode switch changes and observed portions of power ascension.

b. Finding

Introduction: The inspectors identified a Non-Cited Violation (NCV) of TS 5.4.1 having very low safety significance (Green) for failing to lock open Valve 2-0220-57B, Feedwater Manual Isolation Valve.

Description: The inspectors identified during a drywell closeout of Unit 2, on September 23, 2004, that Valve 2-0220-57B was not locked open as required by out-of-service checklist 31383 performed on September 22, 2004. The licensee had taken 2-0220-57B out-of-service closed during the outage to perform corrective maintenance on the 2-2301-7, high pressure coolant injection (HPCI) discharge check valve. The inspectors verified that 2-220-57B was open, but the locking device was wrapped around the valve yoke and not the valve handwheel. Valve 2-220-57B was in the flow path for HPCI injection and B train of feedwater.

Analysis: The inspectors determined that failing to lock the valve open was an operator performance deficiency warranting a significance evaluation in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, " Issue Disposition Screening," issued on June 20, 2003. The inspectors determined that the finding was more than minor because it affected the configuration control and human performance attributes of the Mitigating Systems cornerstone; and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. In addition, this was not the first time the NRC identified drywell valves that were in the correct position but not locked per licensee procedure. During the Unit 2 refueling outage drywell close out on November 9, 2003, the inspectors identified that the two reactor head vent valves 2-0299-59 and 2-0299-60 were closed but not locked closed per licensee procedure. This was documented in Condition Report 185823. Regarding Valve 2-220-57B, licensee management personnel stated that, when interviewed, the operators stated that the valve was difficult to open and when they finished they forgot to lock the valve. The individuals did not have a copy of the clearance order checklist with them in the drywell in an effort to reduce dry active waste. However the operators were briefed prior to the task and the clearance checklist clearly stated the valve was to be locked. Therefore, this finding also affected the cross-cutting area of Human Performance.

The inspectors completed a significance determination of this issue using IMC 0609, "Significance Determination Process (SDP)," Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," dated September 10, 2004. For Phase I screening the inspectors answered "No" to Question 1 under Mitigating Systems Cornerstone because the finding did not result in a loss of function for either the feedwater or HPCI systems. The finding screened as Green.

Enforcement: Technical Specification 5.4.1 required, in part, that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, Revision 2, Appendix A, February 1978, Paragraph 1.c recommends procedures for equipment control (e.g., locking and tagging). One of the equipment control procedures that implemented this TS was OP-MW-109-101, "Clearance and Tagging," Revision 2. Step 10.4.1.5.A, stated, "PLACE equipment in the positions/conditions specified on the clearance checklist." Clearance Checklist 31383 Step 18, required valve 2-220-57B to be locked in the open position. The operators opened but did not lock the valve in the required position. The valve was in the open but unlocked position for less than 24 hours. The non-licensed operators were temporarily removed from shift duties and a fact finding review was scheduled for completion after the end of the inspection period. The individuals were counseled. The licensee planned to require carrying clearance checklists into the drywell in the future. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program (Issue Report 256029), this violation is being treated as a Non-Cited Violation, consistent with Section VI.A of the NRC Enforcement Policy. **(NCV 05000237/2004010-04)**

1R22 Surveillance Testing (71111.22)

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 Event, Mitigating Systems, and Barrier Integrity Cornerstones:

- DIS 0500-06, Rev. 20, Condenser Low Vacuum Pressure Switches Channel Calibration and Channel Functional Test;
- DOS 0700-05, Rev. 09, Intermediate Range Monitor Upscale and Inoperative Functional Testing;
- Emergency Relief Valve Pressure Switches; and
- Unit 2/3, B Train, Standby Gas Treatment, Preventative Maintenance Surveillance on Limitorque Valve Operator 2/3-7504-B.

1R23 Temporary Modification (71111.23)

a. Inspection Scope

The inspectors screened three active temporary modifications and assessed the effect of the temporary modifications on safety-related systems. The inspectors also determined if the installation was consistent with system design:

- Temporary Configuration Change Package No. 347984, Revision 1, “Install Temporary Recorder to Monitor Unit 2 Steam Dryer Parameters;”
- Control room ventilation system as called for in DOA 5750-01; and
- Temporary Configuration Change Package No. 349028, Revision 0, “Install Bleeder Valve on the Unit 3 ‘C’ Condenser Hood Low Vacuum Sensing Line to Prevent Moisture Buildup.”

b. Findings

No findings of significance were identified.

1EP6 Drill and Training Evaluations (71114.06)

September 27, 2004, Emergency Preparedness Performance Indicator Drill

a. Inspection Scope

The inspectors observed station personnel during a licensee only participation emergency preparedness drill exercise on September 27, 2004, to determine the effectiveness of drill participants and the adequacy of the licensee's critique in identifying weaknesses and failures. The drill scenario involved failure of the master recirculation flow controller, trip of the 2A reactor building closed cooling water pump, a break in an instrument line in the drywell, and failure of the core spray pump.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Public Radiation Safety

2PS3 Radiological Environmental Monitoring Program (REMP) And Radioactive Material Control Program (71122.03)

.1 Inspection Planning - Reviews of Radiological Environmental Monitoring Reports and Data

a. Inspection Scope

The inspectors reviewed the 2002 and 2003 Annual Radiological Environmental Operating Reports, the results of monthly radiological environmental monitoring analyses for January through May 2004, and the most recent licensee assessment results to verify that the REMP was implemented as required by TSs and the Offsite Dose Calculation Manual (ODCM). The inspectors reviewed the radiological environmental reports for changes to the ODCM with respect to environmental monitoring, commitments in terms of sampling locations, monitoring and measurement frequencies, land use census, the sample analysis vendor's inter-laboratory comparison program, and analysis of radiological environmental sample data. The inspectors reviewed the ODCM to identify the environmental monitoring stations and evaluated the locations of these stations and the types of samples collected from each to determine if they were consistent with the ODCM and NRC guidance in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light Water Cooled Nuclear Power Plants," and in Regulatory Guide 4.8, "Environmental TSs for Nuclear Power Plants." The inspectors reviewed the Updated Final Safety Analysis Report (UFSAR) for information regarding the monitoring program and the Emergency Response Plan for information regarding meteorological monitoring instrumentation to determine whether the environmental monitoring program was developed consistent with its design basis.

The inspectors reviewed the scope of the licensee's audit program to verify that it met the requirements of 10 CFR 20.1101(c).

These reviews represented one inspection sample.

b. Findings

No findings of significance were identified.

.2 Onsite Inspection Activities

a. Inspection Scope

The inspectors walked-down all eight "indicator" environmental air sampling stations, the sole "control" station, both operable "special" air sampling stations, and approximately 25 percent of the thermoluminescence dosimeter (TLD) monitoring stations. The walkdowns were performed to determine whether these environmental stations were located as described in the ODCM, to assess equipment material condition and operability, and to verify that environmental station orientation relative to plant effluent release points, vegetation growth control, and equipment configuration allowed for the collection of representative samples.

The inspectors accompanied the REMP contract technician and observed the collection and change-out of air particulate and charcoal cartridges at each air sampling station and observed the collection of surface water samples to determine whether appropriate practices were used to ensure sample integrity and to verify that sampling techniques were in accordance with the licensee's procedures.

The meteorological tower was walked down by the inspectors to verify it was adequately sited and that instrumentation was installed consistent with Regulatory Guide 1.23, "Meteorological Programs in Support of Nuclear Power Plants." The inspectors verified that the meteorological instruments were operable, calibrated, and maintained in accordance with the Emergency Response Plan, the guidance provided in NRC Safety Guide 23, and applicable licensee procedures. The inspectors compared real-time data collected at the meteorological tower versus the time-averaged data transmitted to the control room to verify data integrity.

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

The inspectors reviewed sampler station modifications since the last inspection and/or significant changes made by the licensee to the ODCM as dictated by the 2002 or 2003 land use census. The inspectors reviewed technical justifications for changed sampling locations. The inspectors verified that the licensee performed the reviews required to ensure that the changes did not affect its ability to monitor the impacts of radioactive effluent releases on the environment.

The inspectors reviewed the calibration and maintenance records for all indicator, control and special environmental air samplers, focusing on the air flow meter and particulate filter/charcoal cartridge components. Additionally, records of the most recent full calibration for each of the two field rotameters and for the master rotameter used by the licensee to measure and validate air sample pump flow rates was reviewed to ensure traceability to the National Institute of Standards and Technology. As the licensee does not conduct analyses of REMP samples on-site and utilizes a vendor laboratory to provide analytical services, the inspectors did not review licensee calibration records for environmental sample radiation measurement instrumentation (i.e., count room equipment) or quality control charts.

The inspectors reviewed the results of the REMP sample vendor's quality control program including the inter-laboratory comparison program to verify the adequacy of the vendor's program and the corrective actions for any identified deficiencies. The inspectors reviewed the LLD values achieved by the vendor laboratory for all REMP required sample media to verify that analytical detection capabilities met ODCM requirements for each environmentally monitored pathway. The inspectors reviewed the report of the last quality assurance audit of the radiological environmental monitoring program to determine whether the licensee met its TS/ODCM requirements.

These reviews represented six inspection samples.

b. Findings

No findings of significance were identified.

.3 Unrestricted Release of Material from the Radiologically Controlled Area (RCA)

a. Inspection Scope

The inspectors observed locations where the licensee typically monitors potentially contaminated material and individuals leaving the RCA, and evaluated the procedures and practices used for control, survey, and release of materials and workers from these areas. The inspectors questioned several radiation protection staff responsible for the performance of personnel surveying and releasing material for unrestricted use to assess their knowledge of procedures and protocols and to verify that release surveys are performed appropriately.

The inspectors assessed the radiation monitoring instrumentation used for both the unrestricted release of workers and material/equipment from the RCA, to determine if it was appropriate for the radiation types present and was calibrated with radiation sources consistent with the plant's nuclide mix. The inspectors reviewed the licensee's criteria for the survey and release of potentially contaminated material and workers to verify that there was guidance on how to respond to an alarm which indicates the potential presence of licensed radioactive material. The inspectors reviewed the licensee's radiation survey equipment to ensure the radiation detection sensitivities were consistent with the NRC guidance for surface contamination contained in Circular 81-07, "Control of Radioactively Contaminated Material," and Information Notice 85-92, "Survey of Wastes Before Disposal from Nuclear Reactor Facilities," and with Health Physics

Positions (position-221) in NUREG/CR-5569 for volumetrically contaminated material. The inspectors reviewed the licensee's program to determine if it adequately identified and evaluated the impact of difficult-to-detect radionuclides (i.e., those that decay via electron capture) and accounted for those nuclides during routine unrestricted release surveys. The inspectors reviewed the licensee's procedures and records to verify that the radiation detection instrumentation was used at its typical sensitivity level based on appropriate counting parameters (i.e., counting times and background radiation levels). The inspectors verified that the licensee had not established a "release limit" by altering the instrument's typical sensitivity through such methods as raising the energy discriminator level or locating the instrument in a high radiation background area.

Additionally, the inspectors reviewed the circumstances associated with the unconditional release of a worker's contaminated boot in September/October 2002, that was subsequently identified when the worker attempted to leave the Braidwood Nuclear Station approximately 1 year later wearing those boots. Specifically, the inspectors reviewed the licensee's condition evaluation of the incident, reviewed radiation protection (RP) procedures governing the unconditional release program, and discussed the incident with RP staff. The inspectors also independently performed a dose assessment to verify the adequacy of the occupational dose assigned to the worker that wore the contaminated boot.

These reviews represented two inspection samples.

b. Findings

Introduction: A self-revealed finding of very low safety significance which involved a Non-Cited Violation was identified for the failure to conduct an adequate radiological survey of a worker's footwear following a contamination monitor alarm, prior to its unconditional release from the RCA. As a result, the licensee failed to detect a discrete radioactive particle (DRP) embedded in a boot worn by a worker which allowed the contaminated boot to be released unconditionally without any radiological restrictions.

Description: On October 16, 2003, a contract worker alarmed the portal radiation monitor at the gatehouse upon attempting to depart the Braidwood Nuclear Station. The cause of the alarm was later determined to be a DRP comprised primarily of cobalt-60 that was embedded (fixed) on the upper exterior portion of the individual's work boot. The individual had not entered any RCAs since starting work at Braidwood 3 days earlier and had not worked at a nuclear plant since October 2002. In September and October 2002, the individual worked at the Dresden Station during their refueling outage and was involved in reactor assembly/disassembly. The boots were worn exclusively for work at nuclear plants and remained at the individual's residence when not being used.

An investigation by Dresden RP staff revealed that the individual had contaminated his boots twice while at Dresden Station (September 18 and October 25, 2002) but was allowed to leave the site with the boots after they were decontaminated and hand surveyed by RP staff, and the worker subsequently cleared the personal contamination monitor (PCM) at the RCA egress. The boots had remained at the station during the individual's employment tenure at Dresden and were hand carried through the gatehouse portal monitors on the worker's final day onsite on October 25, 2002. Due to the location and the quantity (90 nanocurie maximum) of radioactive material on the worker's boot and the type of radiation monitors used at Dresden's RCA and plant egress locations, the DRP was probably not detected by these automated monitors.

Based on the licensee's evaluation and the inspector's independent assessment of the problem, the contaminated boot escaped detection by the licensee primarily because the RP staff failed to conduct a thorough survey (hand-held instrument frisk) on one or both of those occasions when the worker alarmed the automated PCM at the RCA egress. Following a survey (frisk) of the worker and decontamination of the worker's boots by the RP staff, the worker was allowed to leave after successfully clearing a PCM. The embedded DRP probably escaped detection by the PCM due to the type of detectors used at the RCA egress and the location of the particle relative to the detector geometry. The PCMs located at the contractor RCA egress used by the worker were sensitive to beta-emitting radioactive material only and lacked gamma-sensitivity. The DRP likely escaped gatehouse portal monitor detection because the worker hand-carried the boots through the monitor in a less than optimal detection configuration.

<u>Analysis</u>: The inspectors determined that the licensee failed to conduct an adequate survey following PCM alarms. As a result, the licensee did not detect a DRP that was embedded on the worker's boot. This failure represents a performance deficiency. The inspectors determined that the issue was associated with the "Program and Process" and "Human Performance" attributes of the Public Radiation Safety Cornerstone and affected the cornerstone objective to ensure adequate protection of the public health and safety from exposure to radioactive materials released into the public domain. Also, the issue involved an occurrence in the licensee's radioactive material control program that was contrary to NRC regulations. Therefore, the occurrence represents a more than minor issue which was evaluated using the significance determination process (SDP) for the Public Radiation Safety Cornerstone.

The inspectors determined that the licensee failed to prevent the inadvertent release and loss of control of licensed material outside the protected area that could have potentially caused radiation dose to the public. Utilizing Manual Chapter (MC) 0609, Appendix D, "Public Radiation Safety SDP," the finding involved radioactive material control but did not involve transportation, public radiation exposure was not greater than 0.005 rem total effective dose equivalent (TEDE) and the licensee did not have more than five radioactive material control occurrences in the previous eight quarters. Based on both the licensee's and the inspectors' independent dose assessments, the DRP did not result in a TEDE dose (as defined in 10 CFR Part 20) to either the worker or a member of the public greater than one mrem. Therefore, consistent with Section VI of Appendix D to MC 0609, the finding is not suitable for SDP evaluation. The extremity (foot) dose determined for the worker was conservatively calculated at 500 mrem or one percent of the occupational dose limits of 10 CFR 20.1201. Based on the lack of public dose consequence and the magnitude of the occupational dose to the worker, the finding is determined to be of very low safety significance.

<u>Enforcement</u>: Title 10 CFR 20.1501 requires, in part, that surveys be made as necessary to comply with the requirements of 10 CFR Part 20 to evaluate the quantities of radioactive material, the magnitude of radiation levels, and the potential radiological hazards. Subpart I of 10 CFR Part 20, "Storage and Control of Licensed Material," and

specifically 10 CFR 20.1802 require that licensed material in an unrestricted area that is not in storage be controlled. However, between September 18 and October 25, 2002, the licensee failed to conduct adequate followup surveys of a worker that alarmed PCMs upon attempting to leave the RCA. As a result, a DRP was released into the public domain where it remained uncontrolled until detected at the Braidwood Station approximately 1 year later. The failure to conduct adequate surveys following PCM alarms is a violation of 10 CFR 20.1501 which led to a violation of 10 CFR 20.1802. The finding is not suitable for SDP evaluation, but has been reviewed by NRC management and is determined to be a Green finding of very low safety significance.

The licensee performed an apparent cause evaluation of this event, assessed the dose to the worker and implemented adequate corrective action. These corrective actions included tailgate training to radiation protection staff that respond to contamination monitor alarms, improvements to RCA automated radiation monitors used at the main RCA egress location via the installation of new gamma-sensitive monitors, and actions to enhance gamma-sensitivity of those radiation monitors located at alternate RCA egresses and at the gatehouse. Since the licensee documented this issue in its corrective action program (condition report (CR) 181692) and because the violation is of very low safety significance, its is being treated as a Non-Cited Violation.
(NCV 05000237/2004010-05; 05000249/2004010-05).

.4 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed the licensee’s self-assessments, audits and Special Reports, as applicable, related to the radiological environmental monitoring and radioactive material control programs since the last inspection to determine if identified problems were entered into the corrective action program for resolution. The inspectors also verified that the licensee's self-assessment and/or audit program was capable of identifying repetitive deficiencies or significant individual deficiencies in problem identification and resolution.

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

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

These reviews represented one inspection sample.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA1 Performance Indicator Verification (71151)

.1 Initiating Events and Mitigating Systems

a. Inspection Scope

The inspectors reviewed a sample of plant records and data against the reported Performance Indicators in order to determine the accuracy of the indicators:

Unit 2:

- Heat Removal (Low Pressure Coolant injection/Containment Cooling Service Water), July 2003 - 2004
- High Pressure Coolant Injection, September 2003 - 2004
- Emergency AC Power System, April 2003 - September 2004
- Safety System Functional Failure, July 2003 - September 2004

Unit 3:

- Heat Removal (Low Pressure Coolant injection/Containment Cooling Service Water), July 2003 - 2004
- High Pressure Coolant Injection, September 2003 - 2004
- Emergency AC Power System, April 2003 - September 2004
- Safety System Functional Failure, July 2003 - September 2004

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety

.2 Radiation Safety Strategic Area

a. Inspection Scope

The inspectors sampled licensee submittals for the performance indicator (PI) listed below for the period indicated. To verify the accuracy of the PI data reported during that period, PI definitions and guidance contained in Revision 2 of Nuclear Energy Institute Document 99-02, “Regulatory Assessment Performance Indicator Guideline,” were used. The following PI was reviewed:

- Radiological Effluent TS/Offsite Dose Calculation Manual Radiological Effluent Occurrence PI

The inspectors reviewed the licensee's assessment of this PI by reviewing CRs generated during approximately the 18 months preceding the inspection to identify any potential occurrences such as unmonitored or improperly calculated effluent releases that could have impacted offsite dose. Also, the inspectors evaluated the licensee's methods for determining offsite dose from radiological effluents and reviewed monthly PI data elements for the April 2003 through June 2004 period to verify that data was recorded and verified as required by the licensee's PI procedure.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

a. Inspection Scope

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. Minor issues entered into the licensee's corrective action system as a result of inspectors' observations are generally denoted in the report. 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 condition reports and attending daily condition report review meetings.

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up (71153)

a. Inspection Scope

The inspectors reviewed licensee event reports (LERs) to ensure that issues documented in these reports were adequately addressed in the licensee's corrective action program. The inspectors also interviewed plant personnel and reviewed operating procedures to ensure that generic issues were captured appropriately.

The inspectors reviewed the Updated Final Safety Analysis Report and other documents to verify the statements contained in the LERs.

r. Findings

.1 (Closed) LER 50-237;249/2004-003-00: Unit 2 and 3 Control Room Emergency Ventilation System Inoperable Due To Damper Failure to Close

Introduction: Two Green findings were identified during the review of LER 50-237;249/2004-003. The first was a Green self-revealing finding involving the failure to restore the control room emergency ventilation (CREV) system to operable status following maintenance. This resulted in a NCV of TS 3.7.4 , CREV System, Action Statement A.1. This finding was considered to be self-revealing because it was discovered when the system did not operate properly during routine realignment from train A to train B.

The second was a Green NRC identified finding involving failure to control parts and equipment necessary for the performance of a post accident procedure to maintain control room habitability in the event of a failure of train B of control room ventilation. This finding was considered to be NRC identified because, when the inspector was doing an in-plant walk-down of an operating procedure that had been modified as a corrective action for this LER, the inspector found that necessary parts were not being controlled as called for in the procedure.

Discussion: The Unit 2 and 3 control room heating ventilation and air conditioning (HVAC) system has two trains, A and B. Normally, control room ventilation is supplied by train A. The CREV system is used to protect control room personnel in the event of high outside airborne radioactive contaminants, outside toxic gases, and to purge smoke from the control room in the event of an internal fire. Following a loss of coolant accident (LOCA), the normal outside air intake is isolated and outside air is supplied thought an air filtration unit (AFU) containing high efficiency particulate filters and charcoal adsorbers. Analysis has shown that as long as the control room outside air supply is switched to the AFU within 40 minutes of a LOCA, control room operator doses will remain within limits.

On April 19, 2004, a temporary modification was installed on the CREV system to allow operation of train A HVAC while preventive maintenance (PM) was being performed on a 480 Volt breaker. While the PM was being performed, control power to solenoid valves that supply pneumatic air to position CREV isolation dampers would be lost. Without control power the CREV dampers would fail closed and control room ventilation would be unavailable. The temporary modification provided pneumatic jumpers, (plugs for the solenoid valve exhaust ports and temporary tubing that bypassed the solenoid valves) to maintain the CREV isolation dampers open. With the temporary modification in place the CREV system was declared inoperable on a 7 day time clock.

On April 22, 2004, with the breaker PM completed, a work order to remove the temporary modification was completed, and the CREV system was tested and declared operable. On April 28, 2004, a routine realignment of the control room HVAC from train A to B was attempted. During this realignment the HVAC damper for the unfiltered air supply should have closed but remained open. With the unfiltered air supply damper failed open, protection from external airborne radioactivity and toxic gas was unavailable. The CREV system was again declared inoperable and troubleshooting was initiated. It was found that the temporary modification that had been installed on April 19, 2004, had only been partially removed on April 22, 2004. The temporary tubing that bypassed the solenoid valves had been removed but the plugs in the solenoid valve exhaust ports had been left installed. Therefore, the CREV system had been inoperable for greater than 7 days from April 19 until April 28, 2004, in violation of TS 3.7.4.

On August 3, 2004, the inspectors conducted an in-plant walkdown of procedure DOA 5750-01, "VENTILATION SYSTEM FAILURE," Revision 37. This procedure contained instructions for the installation of the temporary modification for placement of the pneumatic jumpers to maintain the CREV isolation dampers open. This procedure was revised as a corrective action resulting from this LER. The inspectors walked down this procedure because it called for installation of the temporary modification, following a LOCA and failure of the train B air handling unit, to restore control room ventilation. Procedure DOA 5750-01 called for this to be completed within 40 minutes of the LOCA in order to maintain control room radiation dose within limits. The complexity of the procedure made the inspector question if it could be completed in the required time. The procedure is written assuming jumpers, pipe plugs, fittings, and wrenches are pre-staged in the control room Dresden Emergency Operating Procedures (DEOP) Cabinet. When asked, neither of the two control room Unit Supervisors were able to locate the pre-staged equipment for inspection. With the help of the NRC inspector the equipment was located in a miscellaneous equipment drawer of the DEOP Cabinet, the jumper tubes were lying loose and unmarked, the fittings were in two plastic bags labeled with black felt tip marker "Temp Alt 5-10-99," and no hexagonal-end wrench was available to install the required pipe plugs. The inspector concluded that had it been necessary to install the temporary modification and restore control room ventilation following a LOCA it may have taken more than the allotted 40 minutes.

<u>Analysis:</u> For the self-revealing finding involving the violation of TS 3.7.4, Action Statement A.1, the licensee determined that the root cause of the failure to completely remove the temporary modification from the CREV system was inadequate instructions in procedure DOA 5750-01, "Ventilation System Failure," in that guidance for removal was not included in the procedure. Using IMC 0612, Appendix B, "Issue Screening," the inspector determined that the failure to return the CREV system to operable status within its TS allowed outage time was a performance deficiency. The inspectors concluded that this issue was more than minor because it affected the Reactor Safety Barrier Integrity Cornerstone design control and configuration control attributes, and the objective of protecting persons in the control room from radionuclide releases caused by accidents or events.

Using IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," dated September 10, 2004, the inspector answered "No" to question one in the Containment Barriers column of the Phase 1 worksheet of the SDP worksheet in that not only was the radiological barrier function for the control room effected but also the barrier against toxic atmosphere. For the same reason question number one was answered "No," question two was answered "Yes" which required a Phase 3 SDP analysis.

The finding was referred to the SRA for a Phase 3 analysis. The analyst consulted with licensee risk specialists to gain an understanding of the potential sources for toxic gas intrusion into the control room. These sources included onsite sources, especially those that might be deployed on the roof areas near the outside air intake and chemical plants or refineries in the vicinity of the plant. The principal risk to core damage would be debilitation of the control room operators or a forced evacuation requiring alternate means to achieve a safe shutdown. Based on a qualitative judgement of the unlikelihood of toxic gas reaching the outside air intake in sufficient concentrations to affect operators in combination with the short exposure period of the finding (6 days), the analyst concluded that the finding was of very low risk significance (Green).

The second finding, that the licensee had failed to control parts and equipment necessary for the performance of a post accident procedure to maintain control room habitability, was also evaluated using IMC 0612, Appendix B, “Issue Screening.” The inspectors concluded that this issue was more than minor because it also affected the Reactor Safety Barrier Integrity Cornerstone design control and configuration control attributes, and the objective of protecting persons in the control room from radionuclide releases caused by accidents or events. Using IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” the finding screened as very low safety significance (Green). This was due to the inspector answering “Yes” to question number one in the Containment Barriers column of the Phase 1 worksheet of the SDP worksheet, because the uncontrolled equipment would have only been used to provide the radiological barrier function for the control room following an accident.

Enforcement: The failure to restore the CREV system to operable status following maintenance, resulted in a violation of TS 3.7.4. Action Statements B.1 and B.2 require that if restoration of the CREV system to operable status is not complete in 7 days, the plant is to be in Mode 3 within 12 hours and in Mode 4 within 36 hours. Contrary to the above, the licensee failed to properly remove a temporary modification to the CREV isolation dampers, which resulted in the CREV system being inoperable for 9 days from April 19, 2004, to April 28, 2004. Believing the system had been restored to an operable status on April 22, 2004, the licensee failed to take the appropriate actions dictated by T.S. 3.7.4, Action Statements B.1 and B.2. The licensee revised the procedure to give better guidance on how to remove the temporary modification. Because this issue is of very low safety significance and has been entered into the licensee’s corrective action program as Condition Report 217741, this violation is being treated as a Non-Cited Violation, consistent with Section VI.A., of the NRC Enforcement Policy.
(NCV 05000237/2004010-06; 05000249/2004010-06)

The failure to control parts and equipment needed for the performance of a post accident procedure to maintain control room habitability in the event of the failure of train B of control room ventilation was not considered a violation of regulatory requirements. This was because, although procedures call for the use of CREV train A in the event of a train B failure, the finding involved non-safety related equipment. The licensee entered this issue into the station’s corrective action program as CR 242461. The licensee identified a number of corrective actions including properly packaging the necessary parts and tools, revising procedures DOA 5750-01 and 5750-04, and initiating a training request to ensure operations personnel are properly trained in the use of the

DOA air jumpers. This issue was considered a finding of very low significance. **(FIN 05000237/2004010-07; 05000249/2004010-07)**

.2 (Closed) LER 50-237/2004-004-00 and LER 50-237/2004-004-01: Unit 2 Manual Scram Due To The Trip Of A Reactor Recirculation Pump

Introduction: A Green self-revealed finding was identified involving several performance issues which resulted in the initiation of a manual scram. The performance issues included an inadequate process for rewinding the 2A recirculation pump motor when it was installed in 1999, an inadequate evaluation of the testing of the motor before installation, and the failure to perform post maintenance testing of the reactor building closed cooling water (RBCCW) system piping to identify leakage. This failure resulted in the deposit of a conductive substance inside the motor.

Description: On April 24, 2004, with Unit 2 at 66 percent power, the 2A reactor recirculation pump tripped when a short between phases A and B of the motor winding occurred. Operators initiated a manual scram in accordance with Dresden Abnormal Procedure DOA 202-01, "Recirculation (RECIRC) Pump Trip - One Or Both Pumps," Revision 25. The plant was in a region of the reactor's power to flow map that required an immediate manual scram. All plant systems responded normally to the scram.

The licensee initiated a root cause investigation which identified that on October 10, 1999, during the Unit 2 refueling outage, the 2A reactor recirculation pump motor was replaced. This motor was originally sent to the vendor for rewind and testing in March of 1997.

In June 1999, during the testing of the rewound spare motor, inadequately sealed spots at the series connections location at the lower end of the motor were identified. A repair was initiated and the test was completed satisfactorily. However, the licensee did not evaluate or determined the cause at that time of the deficiency. The licensee's root cause report stated that this condition may have indicated a potential weak spot that would have required a full repair.

During the replacement of the motor, the RBCCW piping was reconnected to the rewound motor. However, post maintenance testing was not performed to ensure the piping and fittings did not leak. Following the refuel outage, in October 2001, the licensee identified a RBCCW water leak in an elbow in a lower air intake of the 2A recirculation pump motor. The leak was from a loose pipefitting. This condition had resulted in the presence of electrically conductive surface deposits and moisture intrusion onto the motor winding. Subsequently, this deficiency contributed to the failure of the connection points insulation and the motor failure.

A failure analysis was conducted by the vendor which concluded that the motor failure was attributed to the presence of moisture, conductive surface contamination, and weak spots in the insulation of the series connections due to a voltage and mechanical stress concentration point at the end of the conductor.

Based on the licensee's determination of the root cause and contributing causes, one Green finding was identified involving performance issues which resulted in the initiation of a manual scram. The performance issues included an inadequate process for rewinding the 2A recirculation pump motor when it was installed in 1999, an inadequate evaluation of the testing of the motor before installation, and the failure to perform post maintenance testing of the RBCCW system piping to identify leakage. This failure resulted in the deposit of conductive substance inside the motor.

Analysis: Using IMC 0612, Appendix B, "Issue Screening," the inspectors determined that this finding was more than minor because it affected the initiating events cornerstone objective to limit the likelihood of an initiating event. The inspectors completed a significance determination of this issue using IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." The inspectors answered "No" to all questions in the initiating event column of the Phase 1 Screening Worksheet; and therefore, concluded that the issue was of very low safety significance (Green). (**FIN 05000237/2004010-08)**

Enforcement: No violation of NRC requirements occurred because the finding involved non-safety related equipment. The licensee entered this issue into the station's corrective action program as CR 217570. The licensee identified a number of corrective actions including replacing the failed reactor recirculation pump motor and revising Exelon Nuclear Engineering Standard NES-EIC-40.01, Revision 1, "Large Motor (>2kv) Repair Requirements," to include enhanced testing requirements

.3 (Closed) LER 50-237;249/2004-003-00: Unit 3 Scram Due to Loss of Offsite Power and Subsequent Inoperability of the Standby Gas Treatment System for Units 2 and 3

On May 5, 2004, at 1327 hours, with Unit 3 at 100 percent power, an automatic scram occurred due to a main generator load reject when a loss of offsite power occurred. All systems initially responded to the scram as expected except the standby gas treatment system was unable to maintain the secondary containment at the TS surveillance requirement limit of greater than or equal to 0.25 inches of vacuum water gauge. An Unusual Event for the loss of offsite power was declared at 1342 hours and terminated at 1601 hours on May 5, 2004. Additionally, during restoration of offsite electrical power to Bus 33, the emergency diesel generator 2/3 output breaker tripped.

This event was reviewed during an NRC Special Inspection conducted on May 6 through May 14, 2004, and documented in Special Inspection Report 05000249/2004009 issued on June 21, 2004. The report documents three self-revealed findings of very low safety significance (Green). Two of the findings were determined to be violations of NRC requirements. The first finding, not associated with a violation of NRC requirements, was related to inadequate preventive and corrective maintenance performed on switchyard circuit breaker 8-15 which caused the 'C' phase of the breaker to not open when operated on May 5, 2004. The second finding, associated with a violation of NRC requirements, dealt with inadequate procedures for restoration of offsite power to safety related busses. The third finding, associated with a violation of NRC requirements dealt with an inoperable secondary containment when the opposite unit's drywell purge fans were in operation.

Since the Special Inspection, the licensee completed W.O. 698088 on August 18, 2004, which implemented the ABB Product Advisory for the remaining gas SF6 circuit breaker 6-7. In August 2004, the system engineer verified that ABB has provided Exelon Energy Distribution all applicable product advisories. Dresden Operations issued Standing Order 04-06 for switchyard activity control which discusses the proper management of work occurring in the Dresden switchyards.

Exelon Nuclear has established an executive committee, which includes representatives from Exelon Energy Delivery, to enhance the reliability of its nuclear station switchyards. A significant outcome of this initiative will be the development of preventative maintenance templates for the various switchyard components. The licensee expects full implementation of the templates at Dresden by June 30, 2005. In addition, an Exelon liaison position has been established for industry switchyard issues.

The cause of the emergency diesel generator output breaker trip was still under investigation at the time of the LER submittal. The final corrective actions for the breaker trip will be described in a supplemental report scheduled to be submitted no later than October 30, 2004.

.4 <u>(Closed) LER 50-249/2004-004-00</u>: Unit 3 Shutdown due to Inoperable Source Range Monitor

On May 8, 2004, at 0229 hours, with Unit 3 subcritical during startup in Mode 2, source range monitor (SRM) 24 was declared inoperable due to erratic indications. SRM 22 was already inoperable due to the inability to fully insert the detector into the core region. This resulted in the plant being unable to meet the TS 3.3.1.2, "SRM Instrumentation," requirement of three operable SRMs in Mode 2. A reactor shutdown was commenced at 0450 hours. All control rods were fully inserted in the reverse order in which they had been withdrawn, and the plant entered Mode 3 at 0535 hours.

The licencee determined that SRM 22 had a faulty full-in limit switch. The limit switch was replaced. SRM 24 was inoperable due to an oxide buildup on the connectors. The connectors were disconnected and reconnected numerous times, stopping the erratic indication during troubleshooting. A work request was written to thoroughly clean all the connectors during the next refueling outage. Engineering will evaluate the need for a preventive maintenance item to periodically clean the SRM connectors and evaluate the potential of refurbishing the limit switch and a replacement frequency. No findings were identified.

.5 <u>(Closed) LER 50-237/2004-002-00:</u> Unit 2 SCRAM Due to Main Steam Isolation Valve Closure and Subsequent Inoperability of the Isolation Condenser

<u>Introduction</u>: A violation of 10 CFR 50, Appendix B, Criterion V, of very low safety significance (Green) was self-revealed during the recovery from a reactor scram on April 24, 2004. Licensee maintenance personnel failed to correctly set the open torque switch bypass setting for motor operated Valve (MOV) 2-1301-3, "Isolation Condenser Outboard Condensate Return Valve," on October 8, 1999. This resulted in the failure of Valve 2-1301-3 to open when an operator tried to manually operate the valve from the control room.

Description: On October 8, 1999, two electrical maintenance personnel performed procedure DEP 0040-10, “MOV Votes Test Procedure,” Revision 13, in accordance with Work Order 97071725-01. Procedure DEP 0040-10, Attachment C, Step 30, stated, “Verify limit switch settings are correct and that on four rotor models the Open Torque Switch Bypass opens at 50 percent to 70 percent of the full open stroke.” Valve 2-1301-3 was a gate valve with a 12 inch full mechanical stroke. Because of the valve’s function, the licensee determined that the full electrical stroke should only be 1.65 inches. The electrical maintenance personnel set the open torque switch bypass at .975 inches per the procedure. However, Step E.6 stated, “Verify the limits are set per the MOV Set Point Binder using the VOTES trace and to adjust as necessary,” and Step G.5.F, stated, “Verify limit switch settings comply with the Set Point Binder, if adjustments have to be made refer to DEP 040-09, “Limitorque Valve Operator Maintenance.” Both of these steps were signed off as complete. The MOV Set Point Binder had the correct open torque switch bypass setting of 4.5 inches which was beyond the valve’s electrical open stroke. The licensee concluded that the procedure was inadequate since it contradicted itself and that human performance was a contributing cause. The inspectors agreed with the licensee’s conclusions. The procedure was misleading; however, had the technicians read the MOV Set Point Binder and saw the discrepancy the problem could have been resolved.

Analysis: The inspectors determined that the failure to adequately set the open torque switch bypass setting for Valve 2-1301-3 was a performance deficiency warranting a significance evaluation. The inspectors concluded that the finding was more than minor in accordance with IMC 0612, Appendix B, “Issue Screening,” dated June 20, 2003. The inspectors determined this was more than minor because it affected the configuration control, equipment performance, and procedure quality attributes of the mitigating systems cornerstone and the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences.

The inspectors completed a significance determination of this issue using IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” dated September 10, 2003. For the Phase 1 analysis the inspectors answered questions 1 and 2, “Yes” under the mitigating systems cornerstone column. This resulted in the required performance of a Phase 2 analysis. The inspectors reviewed worksheet tables for the following: Transients, Transients without Primary Heat Sink, Loss of Service Water, Loss of Instrument Air, Loss of An AC [alternating current] Bus, and Loss of Offsite Power. The inspectors assumed that 1) the isolation condenser was unavailable from the beginning of each transient, and 2) that the isolation condenser valve could not be recovered. Base on these two assumptions, using the counting rule, worksheet step 13 was greater than zero which initially had a risk significance of Yellow. The inspectors realized that the assumptions made to perform the Phase 2 were inaccurate. The evaluation was sent to the Regional Office in order to perform a Phase 3 analysis. The Phase 3 analysis performed by the Senior Reactor Analyst (SRA) concluded that the safety significance of this finding based on the change in core damage frequency was Green. The Phase 3 analysis reviewed the potential failure probabilities based on the procedure and equipment inadequacies.

Enforcement: The failure to have adequate instructions for work involving a safety-related valve resulted in a violation of 10 CFR 50, Appendix B, Criterion V. Code of Federal Regulations Title 10, Part 50, Appendix B, states in part, "Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances..." The setting of the open torque switch bypass on Valve 2-1301-3 was an activity affecting quality. Contrary to the above, on October 8, 1999, the open torque switch bypass was set improperly on Valve 2-1301-3 due to inadequate procedural guidance. Because this issue is of very low safety significance and has been entered into licensee's corrective action program (IR 216787), this violation is being treated as a Non-Cited Violation, consistent with Section VI.A., of the NRC Enforcement Policy. **(NCV 05000237/2004010-09)**

.6 (Closed) LER 50-237;249/2004-005-00: Units 2 and 3 Inoperable Turbine Condenser Vacuum - Low Switches

Introduction: The inspectors identified a NCV of TS 3.3.1.1 having a very low safety significance (Green) for failing to properly enter a required TS LCO when the 3C and 2A turbine main condenser low vacuum Reactor Protection System (RPS) scram channels were inoperable. This issue was considered to be NRC-identified because the licensee had failed to identify this deficiency without the inspectors' questions on July 1, 2004. The licensee looked into the issue further and identified that these switches had been inoperable outside their TS outage time multiple times in the past 2 years. Had it not been for the inspector's questioning, the licensee would not have found the problem at this particular time. Therefore, the finding is considered inspector identified.

Description: On April 5, and May 3, 2004, vacuum indication taken from the 3C main condenser hood was not indicating accurate values after circulating water flow was reversed. Subsequent trending activities by the system engineer revealed that the vacuum indication from the 2A condenser hood was also exhibiting bad vacuum indication on some occasions after flow reversals.

The low condenser vacuum pressure switches provided reactor scram signals to protect the reactor from loss of the main heat sink. Protection for the condenser itself was assured by closure of the turbine stop and bypass valves as vacuum decreases below a preset low level. Condition Report (CR) 218325 dated May 1, 2004, "Condenser Vacuum Indication Error Caused by Condensation," stated that the 3C condenser low vacuum reading was inaccurate. The error caused a measured vacuum that was higher than the actual value; and therefore, farther away from the trip setpoint. The cause of the inaccurate vacuum indication was attributed to improperly sloped sensing lines and water accumulation. The sensing lines connected vacuum instrumentation to the condenser hoods. The RPS trip switches utilized the same sensing lines as the control room indication and plant process computer points.

Operations personnel failed to recognize that the 3C switch was inoperable. At the time of discovery, the switch had already been inoperable for a time greater than the TS allowed completion time. At the time of discovery operations personnel took appropriate actions and restored the channel to an operable status within the TS LCO.

The inspectors discussed the Unit 3 main condenser vacuum sensing line operability evaluation with system engineering personnel and were able to obtain historical data that demonstrated that the 3C and 2A pressure switches were inoperable for greater than their allowable outage time at least four times in the past 2 years. The licensee agreed that periods of longer than the TS LCO allowed completion time were exceeded.

The licensee's root cause report stated that on some occasions, initial interaction between plant engineering and the operations personnel failed to identify vacuum indication instrumentation as inoperable. The licensee attributed these failures to weaknesses in the operations training materials with regard to the inter-relations between the RPS pressure switches and the control room recorder. The licensee also identified weaknesses in the operator rounds appendices and procedures. Specifically, the operations flow reversal procedure, DOP 4400-08, Revision 33, "Circulating Water System Flow Reversal," did not include information to tell operators what to observe relative to hood vacuum following a circulating water flow reversal. Finally, the root cause report identified a weakness in the corrective action program, in that previous corrective actions from Licensee Event Report (LER) 249/98006 for a similar issue were too narrowly focused and did not correct the problem.

Analysis: The inspectors determined that the failure to take adequate corrective actions to prevent recurrence of inoperable low vacuum RPS switches, failure to recognize the switches were inoperable, and failure to enter the appropriate TS LCO were performance deficiencies warranting a significance evaluation. The inspectors concluded that the finding was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated June 20, 2003. The inspectors determined that this finding was more than minor because it affected the mitigating systems cornerstone design control and equipment performance attributes, and the objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors completed a significance determination of this issue using IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," dated September 10, 2004. The inspectors answered "No" to all questions in the mitigating systems column of the Phase 1 Screening Worksheet; and therefore, concluded that the issue was of very low safety significance (Green). This finding was associated with the reactor safety cross-cutting attribute of Problem Identification and Resolution.

Enforcement: Technical Specification 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Condition A, stated that if one or more required channels were inoperable, place the channel in trip within 12 hours. Contrary to the above, the licensee failed to enter the TS required action within the 12-hour allowable outage time when the 3C and 2A turbine main condenser low vacuum RPS scram channels were inoperable, which caused an indicated vacuum value that was greater than the actual value; and therefore, farther away from the trip limit. This occurred on four separate occasions between 2002 and 2004. The licensee's corrective actions, as described in the root cause report, included installing temporary vent valves on the 3C and 2A sensing lines to continuously purge and clear condensation from the lines, enhancing operations training materials to include adequate level of detail on main condenser vacuum indications, revising the operations' procedure DOP 4400-08, Revision 33, "Circulating Water System Flow Reversal," and performing internal and external condenser

walkdowns during the next outage on Unit 2 and Unit 3 to determine the sensing line slope and to repair the sensing line slope, if necessary. Because this issue is of very low safety significance and has been entered into the licensee's corrective action program as Condition Report 234361, this violation is being treated as a Non-Cited Violation, consistent with Section VI.A., of the NRC Enforcement Policy.
(NCV 05000237/2004010-10; 05000249/2004010-10)

4OA4 Cross-Cutting Findings

.1 A finding described in 1R06 of this report had, as its primary cause, a Problem Identification and Resolution deficiency, in that weaknesses had been previously identified by the NRC with the external flooding surveillance and operating procedures.

.2 A finding described in Section 1R12 of this report had, as its primary cause, a Problem Identification and Resolution deficiency, in that two functional failures of the reactor building ventilation system occurred and were not properly entered into the corrective action program which resulted in the failure to move the reactor building ventilation system from 10 CFR 50.65 a(2) to a(1).

.3 A finding described in Section 1R20 of this report had, as its primary cause, a Human Performance deficiency, in that operators failed to perform a return to service clearance checklist as written. The operators failed to lock open the 2-220-57B main feedwater isolation valve when it was returned to service.

.4 A finding described in 40A3 of this report had, as its primary cause, a Problem Identification and Resolution deficiency, in that 1) the licensee failed to correct equipment deficiencies which resulted in one channel of main condenser low vacuum reactor trip signals being inoperable on both units after it had been identified as a problem in 1998; and 2) operations personnel failed to identify that the main condenser vacuum low pressure switches were inoperable on multiple occasions from 2002 to 2004.

4OA5 Other Activities

(Closed) Unresolved Item 50-237-97021-01 (DRS); 50-249-97021-01 (DRS): The inspector had 4 concerns: 1) licensee had inadequate analysis to support UFSAR statements that the Dresden station could be safely shutdown following a Dresden Dam failure coincident with a loss of coolant accident (LOCA) in one of the units; 2) licensee had inadequate analysis to support UFSAR statements that station could be safely shutdown following a Dresden Dam failure during normal operations using only Class I systems; 3) ultimate heat sink (UHS) inventory was less than assumed in the Systematic Evaluation Program (SEP) Safety Evaluation Report (SER); and 4) licensee had inadequate analysis supporting the assumed Service Water Pump (SWP) performance. These items were referred by Task Interface Agreement (TIA) for Office of Nuclear Reactor Regulation (NRR) review. The TIA response stated that the accident scenario of Concern 1 was beyond design basis and no supporting analysis was needed. Also, the Current Licensing Basis was codified in the Power Uprate SER of license amendment 191 to Unit 2 and license amendment 185 to Unit 3. The license amendments and SER were issued December 21, 2001. The Power Uprate SER is located in ADAMS at accession number ML0135401870. The Power Uprate SER did not require the Dresden Station to safely shutdown following a Dresden Dam failure coincident with a LOCA in one of the units. Therefore, Concern 1 is closed. For Concern 2, the Power Uprate SER stated that the licensee did not need to use only class I systems to shut down. Therefore Concern 2 is closed. For Concern 3, the TIA response stated that the licensee's planned action of using portable engine-driven pumps to pump water from the intake and discharge canals was acceptable to compensate for the reduced UHS inventory assumption since the planned actions were in the emergency procedure. The Power Uprate SER also stated this was acceptable. Therefore, Concern 3 is closed. For Concern 4, the TIA response stated that the SWPs could not be assumed to perform without a supporting analysis. The licensee informed the inspector that the SWPs would no longer be relied on to function after a dam failure and changed the UFSAR accordingly. Also, the Power Uprate SER stated that the licensee did not need to credit the SWP pumps for shutdown; therefore, Concern 4 is closed. This item is considered closed.

40A6 Meetings

Interim Exit Meetings

Interim exit meetings were conducted for:

- Public radiation safety inspection for radiological environmental monitoring and radioactive material control with Messrs. D. Bost and D. Wozniak on July 22, 2004. On July 28, 2004, the inspection results were further discussed in a telephone conversation with Mr. S. Taylor.

4OA7 Licensee Identified Violation

The following violation of very low safety significance (Green) was identified by the licensee and is a violation of NRC requirements which met the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as a Non-Cited Violation.

Cornerstone: Public Radiation Safety

Technical Specification 5.4.1 requires that written procedures be established, implemented and maintained covering the applicable procedures in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Procedures specified in Regulatory Guide 1.33 include those for radiation surveys and for contamination control, which are provided, in part, by licensee procedure RP-AA-503, "Unconditional Release Survey Method." The licensee's procedure requires that material or equipment not be released for unrestricted use unless it has no detectable licensed radioactive material. Contrary to this procedure, two separate incidents occurred in October 2003, when low level but detectable quantities of licensed radioactive material (contaminated materials) were identified by radiation protection (RP) staff outside the Radiologically Controlled Area (RCA) (but inside the protected area). These problems occurred because workers failed to understand requirements for the unconditional release of personal items and other materials from the RCA, exacerbated by inattentive radiation protection staff. The occurrences are documented in the licensee's corrective action program as CR 00181367 and CR 00184544. Corrective actions included tailgate training with RP staff, counseling of involved workers and plans to enhance plant and contractor staff training along with procedural revisions. These problems are of very low safety significance because the contamination levels on the items inadvertently released outside the RCA were very low and consequently of little to no dose consequence. The contaminated items remained within the licensee's protected area; and therefore, are not counted as "occurrences" as provided in NRC Manual Chapter 0609, Appendix D, "Public Radiation Safety Significance Determination Process."

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

D. Bost, Site Vice President
D. Wozniak, Plant Manager
H. Bush, Radiological Engineering Manager
R. Conklin, Radiation Protection Supervisor
J. Fox, Design Engineer
R. Gadbois, Operations Director
D. Galanis, Design Engineering Manager
V. Gengler, Dresden Site Security Director
J. Griffin, Regulatory Assurance - NRC Coordinator
J. Hansen, Regulatory Assurance Manager
R. Kalb, Chemistry ODCM Coordinator
T. Loch, Supervisor, Design Engineering
M. McGivern, System Engineer
D. Nestle, Radiation Protection Technical Manager
M. Overstreet, Radiation Protection Supervisor
R. Quick, Security Manager
N. Spooner, Site Maintenance Rule Coordinator
B. Surges, Operations Requalification Training Supervisor
B. Svaleson, Maintenance Director
S. Taylor, Radiation Protection Director

NRC

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

IEMA

R. Schulz, Illinois Emergency Management Agency

Contractor

A. Lewis, REMP Technician, Environmental Inc., Midwest Laboratory

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000237/2004010-01 05000249/2004010-01	URI	USFAR change 50.59 (1R06)

Opened and Closed

05000237/2004010-02 05000249/2004010-02	NCV	Source of Make-up Water (1R06)
05000237/2004010-03 05000249/2004010-03	NCV	The Licensee Did Not Move the Reactor Building Ventilation System Into the Maintenance Rule (a)(1) Category (1R12)
05000237/2004010-04	NCV	Operators Failed to Lock Valve in Unit 2 Drywell (1R20)
05000237/2004010-05 05000249/2004010-05	NCV	Failure to Perform an Adequate Radiological Survey Prior to the Unconditional Release of Material Outside the RCA (Section 2PS3)
05000237/2004010-06 05000249/2004010-06	NCV	The Licensee Failed to Correctly Restore the Control Room Emergency Ventilation System to Operable Status Following Maintenance (4OA3)
05000237/2004010-07 05000249/2004010-07	FIN	The Licensee Did Not Control Tools and Equipment Staged to Install a Temporary Modification to Keep the Control Room Emergency Ventilation System Dampers Open in the Event of an Accident (4OA3)
05000237/2004010-08	FIN	Performance Issues Which Resulted in the Initiation of a Manual Scram on Unit 2 Due to Failure of the 2A Recirculation Pump Motor (4OA3)
05000237/2004010-09	NCV	Improperly Set Open Torque Switch Bypass of the Isolation Condenser Outboard Condensate Return Valve (4OA3)
05000237/2004010-10 05000249/2004010-10	NCV	Failure to Prevent Recurrence of Inoperable Condenser Low Vacuum Reactor Protection System Switches (4OA3)

Closed

50-237;249/2004-003-00	LER	Unit 2 and 3 Control Room Emergency Ventilation System Inoperable Due to Damper Failure to Close

50-237/2004-004-00 and 50-237/2004-004-01	LER	Unit 2 Manual Scram Due to the Trip of a Reactor Recirculation Pump
50-237;249/2004-003-00	LER	Unit 3 Scram Due to Loss of Offsite Power and Subsequent Inoperability of the Standby Gas Treatment System for Units 2 and 3
50-249/2004-004-00	LER	Unit 3 Shutdown Due to Inoperable Source Range Monitor
50-237/2004-002-00	LER	Unit 2 Scram Due to Main Steam Isolation Valve Closure and Subsequent Inoperability of the Isolation Condenser
50-237;249/2004-005-00	LER	Units 2 and 3 Inoperable Turbine Condenser Vacuum - Low Switches
50-237/97021-01 (DRS); 50-249/97021-01 (DRS)	URI	UFSAR Dam Failure Discrepancies

Discussed

None.

LIST OF ACRONYMS USED

ABB	Asea, Brown, and Boveri
AFU	air filtration unit
CFR	Code of Federal Regulations
CR	Condition Report
CREV	control room emergency ventilation
DEOP	Dresden Emergency Operating Procedure
DIS	Dresden Instrument Surveillance
DOA	Dresden Abnormal Operating Procedure
DOP	Dresden Operating Procedure
DOS	Dresden Operating Surveillance
DRP	Division of Reactor Projects
DRP	discrete radioactive particle
DRS	Division of Reactor Safety
HPCI	high pressure coolant injection
HVA	heating, ventilation, and air conditioning
IEMA	Illinois Emergency Management Agency
IMC	Inspection Manual Chapter
IR	Issue Report
LCO	limiting condition for operation
LER	Licensee Event Report
LLD	lower limit of detection
LOCA	loss of coolant accident
MC	Manual Chapter
MOV	motor operated valve
MWe	megawatts electrical
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
OA	Other Activities
ODCM	Offsite Dose Calculation Manual
PCM	personnel contamination monitor
PM	preventive maintenance
PMF	probable maximum flood
PI	Performance Indicator
RBCCW	reactor building closed cooling water
REMP	Radiological Environmental Monitoring Program
RCA	radiologically controlled area
RP	radiation protection
RPS	reactor protection system
SDP	Significance Determination Process
SEP	Systematic Evaluation Program
SER	Safety Evaluation Report
SRA	Senior Reactor Analyst
SRM	source range monitor
SWP	service water pump
TEDE	total effective dose equivalent

TER	Technical Evaluation Report
TIA	Task Interface Agreement
TLD	thermoluminescence dosimeter
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
UHS	ultimate heat sink
URI	Unresolved Item
WO	Work Order

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 of 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 2300-01, “High Pressure Coolant Injection (HPCI) System Standby Operation,” Rev. 28
Unit 2/3, DOP 7500-M1/E1, “Unit 2/3 Standby Gas Treatment Valve Checklist,” Rev. 05
DOP 1500-M1, “Unit 2 Unit 2 LPCI and Containment Cooling Valve Checklist,” Rev. 36
M-26, Low Pressure Coolant Injection (LPCI) System Diagram
DOP 0300-M1/E1, “Unit 3 Control Rod Drive Hydraulic System Checklist,” Rev. 32
DOP 0300-M4, “Unit 3 Hydraulic Control Units East (Bank 3) Row 3 and 4,” Rev. 03
DOP 0300-M2, “Hydraulic Control Units West (Bank 1) Row 7 and 8,” Rev. 03
M-365, Diagram of Control Rod Drive Hydraulic Piping
DOP 1400-M1/E1, “Unit 3 Core Spray System,” Revision 17
CR 214447; ASCO solenoid valves 2(3)-399-548A(B) and 548A(B); April 12, 2004
CR 231734; Unplanned LCO entry for CRD M-14 coupling check; June 25, 2004
CR 229869; Unplanned LCO entries for HCU K-6; June 19, 2004
DOP 1400-M1/E1; “Unit 3 Core Spray System”; Revision 17
M-358; Diagram of Core Spray Piping; Revision CC

1R05 Fire Protection

IR# 169275; 2/3 EDG interlock door malfunctioned; July 28, 2004
Dresden Unit 2 Fire Pre-Plan U2TB-43
Dresden Unit 2 Fire Pre-Plan U2TB-37
Dresden Unit 2 Fire Pre-Plan U2TB-36
Dresden Unit 3 Fire Pre-Plan U3TB-68
Dresden Unit 3 Fire Pre-Plan U3TB-69
OP-AA-201-001, “Fire Marshall Tours,” Revision 2

1R11 Operator Requalification

LT017, “Simulator Exercise Guide,” Revision 01, dated September 2004
CR 0255880; Target rock valve degradation found during valve rebuild, September 22, 2004

1R06 Flood Protection

Document Based Instruction Guide; 12NL04; “Operation of the Isocondenser External Flood Emergency Make-up Pump;” Revision 00
AR 111005; NRC Identifies Weakness in External Flood Procedure; June 5, 2002

NRC Inspection Report 50-010/85013; 50-237/85030; 50-249/85026; dated November 15, 1985
AD-AA-101; "Floods;" Revision 7
LS05-82-06-069; "Safety Evaluation of Hydrology SEP Topics II-3.A; II-3.B; II-3.B.1; and II-3.C; June 21, 1982
SA-1334; Input for NRC Contingent Dresden Phase 3 SDP Analysis - External Flooding Frequency for Elevation 517'; Revision 0
CR 246038; DOA 0010-04 Flooding Procedure potentially inadequate; July 27, 2004
DOA 0010-04; "Floods;" Revision 16
DOA 0040-02; "Localized Flooding in Plant;" Revision 15
DOP 1300-03; "Manual Operation of the Isolation Condenser;" Revision 20
WO 668549; D2/3 QRT PM Emergency Diesel Pump (Flood pump) Operation; July 9, 2004
CC-AA-309-1001; "Capacity and Discharge Head for Portable Isolation Condenser Make-up Pump to be used During Flood Conditions;" Revision 0
UFSAR 2.4; "Hydrologic Engineering;" Revision 01A
UFSAR 3.4; "Water Level (Flood) Design;" Revision 4

1R13 Maintenance Risk Assessments and Emergent Work Control

Final Clear checklist # 30458
IR# 248839; 'B' SBGT heater current relay out of tolerance; August 31, 2004
IR# 248991; Incorrect grid location for the 2/3-7506B in DOP 7500-M1/E1; August 31, 2004
WO# 679078-01; Replace/regrease CR105X contacts; September 2, 2004
WO# 679077-01; Replace/regrease CR105X contacts; September 2, 2004
WO# 679080-01;Replace/regrease CR105X contacts; September 2, 2004
WO# 731568-03; Acoustic monitoring of piping to determine leaks outside, between B CST and west side of trackway interlock, and 2/3 diesel generator room
OP-MW-109-101, "Clearance Preparation and Approval Checklist," Rev. 2
WO# 724335; "U3 250V Station Battery Cell #113"
MA-DR-EM-6-83001; "Battery Cell Jumpering," Rev. 0
EC 343009; "Evaluation of Units 2&3 125V and 250V Battery Capacity with One (1) Cell Jumpered as a 2003 Summer Contingency"
EC 366236; "Evaluation For Jumpering Of One Cell On The Unit 3 250VDC Safety Related Battery"
EC 350740; "Evaluation Of Battery Cell Jumper Cart Molded Case Ckt Bkr (Unit 3) 250 Battery"
CR 243971; "NRC Identified Slight Corrosion On Two U3 125V Battery Posts;" August 12, 2004

1R12 Maintenance Effectiveness (71111.12)

CR 221610; Reactor Bldg dP low due to Degraded Purge Filter Housing; May 17, 2005
CR 219346; Following U3 scram, inadequate secondary containment dP; May 5, 2004
CR 208282; Loss of Rx Bldg DP during fan swap; March 14, 2004
CR 206777; Reactor building dP lost while stopping/starting RB Vent; March 7, 2004

CR 197643; Reactor Building Differential Pressure >.25"; January 26, 2004
CR 173440; Rx bldg dP <.25" H2O due to starting unit 3 RB ventilation; August 28, 2003

1R15 Operability Evaluations

NED-I-EIC-0303; "Reactor Water Level ATWS RPT/ARI Logic and ECCS Initiation Setpoint Analysis and Reactor Pressure ATWS RPT/ARI Logic and Setpoint Analysis," Rev. 5
DIS 0263-07, "Unit 2 ATWS RPT/ARI and ECCS Level Transmitters Channel Calibration Test and EQ Maintenance Inspection," Rev. 12
SIL No. 470S2; Non-Condensable Gas Buildup in RPV Water Level Instrumentation; August 28, 1992
IR 245395; NRC Concern with Reactor Level Density Error; August 18, 2004
IR 245856; ERV discharge piping break flange does not meet USAS B31.1; August 19, 2004
CR 229550; Unit 3 reactor Bldg Vent intake plenum collapsing; June 18, 2004
EC Eval 350806; U3 Reactor Building Ventilation Duct Collapsing
EC No. 334998; U3 Reactor Building Ventilation Duct Collapsing; June 27, 2004
CR 2411737; U3 RX Building HVAC duct work found collapsed in three areas; August 4, 2004
CR 223169; U2 RX HVAC duct has split creating in-leakage path for SBGT; May 24, 2004
OE 03-013, "Electromatic Relief Valve Discharge Flanges," Revision 1
CR 204690: HPCI Steamline Water Carryover during Design Basis Events; February 27, 2004
CR 194722; Previously Identified Problem Lead to a Smoked Component; January 12, 2004
CR 208093; CR 105X Auxiliary Contactor Failure Analysis Results; March 12, 2004
CR 215956; Unit 3 Containment Cooling Service Water Keep Fill Line Does not Meet Design Span Requirements; April 20, 2004
Control Room Logs August 23, and September 14, 2004
DOA 6500-12, "Low Switchyard Voltage," Revisions 5 and 6
Technical Specifications 3.8.1, "AC Sources -Operating"

1R17 Permanent Plant Modification

EC 340303;Replacement of SBGT Solenoid Valves 2/3-7541-43A and B; Rev. 000
WO# 600451-01; Replace solenoid valve 2/3-7541-43B on AO-7510-B OPER; September 14, 2004
MA-AA-726-620, "Installation instructions for 0-600 Volt EQ Related Splices," Rev. 0
50.59 Screening Form

1R19 Post Maintenance Testing

WO# 99054268-01; D2/3 72M Eq Limitorque Vlv Oper Surv 2/3-7507-B
MA-AA-723-301, "Periodic Inspection of Limitorque Model SMB/SB/SBD-000 Through 5 Motor Operated Valves," Rev. 1
WO# 565269-07; D3 2Y EQ GE 3B CS Pump Motor Surv
DOS 0040-32, "LPCI and Core Spray Motor EQ Surveillance," Rev. 06

DOS 1400-05; “Core Spray System Pump Operability And Quarterly IST [Inservice Test] Test With Torus Available
WO 344213-01; Disassemble & Inspect CS Min Flow Stop Check Valve
WO 724335-01; “U3 250V Station Battery Cell #113,” Revision 0 and Revision 1
MA-DR-EM-6-83001; “Battery Cell Jumpering,” Revision 0
WO# 00736967, “Inspect/Repair Unit 2 HPCI Discharge Testable Check Valve 2-2301-7"
DOS 7100-06, “Seat Leakage Testing of Valve 2-2301-7," Revision 02

1R20 Refueling and Outage Activities

IR 240083; Outage vulnerability to crane issues; July 29, 2004
IR 240341; Supply identifies pre-outage milestone not met; July 29, 2004
IR 245395; NRC concern with reactor level density error; August 18, 2004
Operator Aid # 8, Rev. 0
DGP 01-01, “Unit Startup,” Revision 116
DGP 01-S1, “Startup Checklist,” Revision 63
DGP 01-S3, “Unit 2 Mater Outage Checklist,” Revision 18
DGP 02-01, “Attachment B, Reactor Cooldown Monitoring,” Revision 88

1R22 Surveillance Testing

WO# 9905344-01; D2/3 72M EG Limitorque Vlv Oper Surv 2/3 7504-B
MA-AA-723-301, “Periodic Inspection of Limitorque Model SMB/SB/SBD-000 Through 5 Motor Operated Valves,” Revision 1
Unit 2/3 DEP 40-09, “Limitorque Valve Operator Maintenance,” Revision 12

1R23 Temporary Modifications

Exelon Procedure CC-MW-112-1001, “Temporary Configuration Change Packages,” Revision 3
DAN 902(3)-4 G-13, “Steam Dryer Parameter Alarmed,” Revision 02
CR 254954; TCCP 347984, Steam Dryer Parameter Temporary Recorder;” dated September 20, 2004

2PS3 Radiological Environmental Monitoring and Radioactive Material Control Programs

Offsite Dose Calculation Manual; Chapters 5 - 6 and Appendices A - C (Revision 2), Chapter 10 (Revision 4), Chapter 11 (Revision 2), Chapter 12 (Revision 5) and Appendix F (Revision 2)
Environmental Inc. Midwest Laboratory, Sampling Procedures Manual; Revision 7
Dresden Nuclear Power Station Annual Radiological Environmental Operating Report for 2002 (dated May 2003) and for 2003 (dated May 2004)
CY-AA-170-100; Radiological Environmental Monitoring Program; Revision 1
RP-AA-503; Unconditional Release Survey Method; Revision 0
TID-2004-003; Unconditional Release Detection Thresholds and Dose Consequences; Revision 0

Focus Area Self-Assessment Report; Radioactive Material Control and Radiological Environmental Monitoring Program; dated June 14, 2004
Nuclear Oversight Health Physics/Radiation Protection Audit Report; Audit No. NOSA-DRE-03-06; dated May 16, 2003
REMP, ODCM, Non-Radiological Effluent Monitoring Audit Report
Audit No. NOSA-DRE-03-08; dated November 19, 2003
Field Rotameter (serial numbers 95W012433 & 91W506166) Quarterly Flow Verifications for January 2003 - July 2004
Field Rotameter (serial numbers 95W012433 & 91W506166) Calibration Certificates; dated May 6, 2003 and July 17, 2003, respectively
Master Rotameter (serial number 91W513308) Calibration Certificate; dated August 12, 2002 and August 8, 2003
Annual Maintenance and Monthly Flow Checks for Environmental Air Sample Pumps; Pump #s 445,455,486,457,454,458,446,443,452,431,453,451,429 and 462; January 2003 - June 2004
Murray and Trettel, Inc. Monthly Reports on the Meteorological Monitoring Program at the Dresden Station; January 2003 - May 2004
CR 00147626; Magenta Tools Found Outside RCA; dated March 5, 2003
CR 00116137; High Tritium Concentration in Non-REMP Well; dated July 18, 2002 and Associated Apparent Cause Evaluation; dated January 21, 2003
CR 00157499; Excessive Off-Site Sampler Shutdowns for 2002; dated May 6, 2003
CR 00181131; Yellow Velcro Strap Found in Venture Break Area; dated October 14, 2003
CR 00181367; Infra-Red Camera Released While Isotopic Showed Cobalt and Manganese; dated October 16, 2003
CR 00181692; Individual Alarms Portal Monitor at Braidwood Station; dated October 16, 2003
CR 00183938; PM-7 Gatehouse Alarm - Found Particle on Boot Lace; dated October 30, 2003
CR 00184544; Low-Level Radioactive Clothing Found Outside the RCA; dated November 2, 2003

4OA1 Performance Indicator Verification

LS-AA-2150; Attachment 1; Monthly PI Data Elements for RETS/ODCM Radiological Effluent Occurrences; April 2003 - June 2004
Quarterly Summary Data of Dresden Station Units 2/3 Maximum Doses Resulting from Airborne Releases and from Aquatic Effluents; 2nd Quarter 2003 - 2nd Quarter 2004

4OA3 Event Follow-up

Operability Evaluation No. 04-008, “Main Condenser Hood/Bay,” Revision 0
Operability Evaluation No. 04-008, “Unit 2 & 3 “2A and 3C Condenser Bay Vacuum Indication/Switch Sometimes Indicates a Non-conservative Value after a Flow Reversal to East-to-West Flow,” Revision 1
CR 218325; Condenser Vacuum Indication Error Caused by Condensation; May 1, 2004
CR 221099; Bad vacuum indication taken from the 3C main condenser hood was previously reported in CR 218325;

CR 213244; Apparent Error in Condenser Turbine Hood Vacuum Indication; April 5, 2004
CR 234361; NRC Questions Potentially Exceeding Tech Spec Comp Time; July 1, 2004
DOA 200-01, “Recirculation (Recirc) Pump Trip - One or Both Pumps,” Revision 25
DOS 0500-18, “Verification of Flow Control Line and Average Core Thermal Power,” Revision 27
NES-EIC-40.01, “Large Motor (>2kV) Repair Requirements,” Revision 1
CR 217533; Unit 2 TADS trigger manually disabled; April 28, 2004
CR 217570; Unit 2 scram due to 2A Recirc pump trip; April 28, 2004
LER 50-237/249/2004-003-00, “Unit 3 Scram Due to Loss of Offsite Power and Subsequent Inoperability of the Standby Gas Treatment System for Units 2 and 3,” dated July 6, 2004
CR 219063; Switching Fault Causes LOOP and Reactor Scram; May 5, 2004
WO 0069088-01, “Rebuild Operating Mechanism All Phases BT 6-7 CB”
Exelon Energy Delivery Maintenance Template, Circuit Breaker 2 Pressure SF6, 2004 Revision 2
CR 219062, “Unit 3 Startup Aborted due to Insufficient SRMs,” dated 05/08/2004
Technical Specification 3.3.1.2, “SRM Instrumentation”
Control Room Logs May 8, 2004