January 27, 2005

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 05000237/2004013;
05000249/2004013

Dear Mr. Crane:

On December 31, 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 January 7, 2005.

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, three NRC identified findings and three self-revealed findings of very low safety significance (Green) were identified. Each of these issues involved a violation of NRC requirements. However, because of their very low safety significance and because they have been entered into your corrective action program, the NRC is treating these issues as Non-Cited Violations, in accordance with Section VI.A.1 of the NRC's Enforcement Policy.

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

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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 05000237/2004013; 05000249/2004013
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:	05000237/2004013; 05000249/2004013
Licensee:	Exelon Generation Company
Facility:	Dresden Nuclear Power Station, Units 2 and 3
Location:	6500 North Dresden Road Morris, IL 60450
Dates:	October 1 through December 31, 2004
Inspectors:	C. Phillips, Senior Resident Inspector M. Sheikh, Resident Inspector P. Pelke, Reactor Engineer K. Stoedter, Senior Resident Inspector, Quad Cities D. Kimble, Senior Resident Inspector, LaSalle D. Eskins, Resident Inspector, LaSalle W. Slawinski, Senior Radiation Specialist D. Jones, Reactor Engineer B. Palagi, Senior Operations Engineer G. Roach, Reactor Engineer R. Smith, Reactor Engineer R. Schulz, Illinois Emergency Management Agency
Observer:	D. Melendez-Colon, Reactor Engineer
Approved by:	Mark Ring, Chief Branch 1 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000237/2004013; IR 05000249/2004013, 10/01/2004 - 12/31/2004, Exelon Generation Company, Dresden Nuclear Power Station, Units 2 and 3; Equipment Alignment, Operability Evaluations, Outage Activities, Surveillance Testing, Problem Identification and Resolution, routine integrated report.

This report covers a 3-month period of baseline resident inspection, announced baseline radiation protection inspections, and announced baseline inservice (71111.08) inspection for the Dresden Nuclear Power Station, Unit 3. The inspection was conducted by Region III inspectors and the resident inspectors. Six findings or violations were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, “Significance Determination Process” (SDP). Findings for which the SDP does not apply may be ‘green’ or be assigned 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. On November 12, 2004, a performance deficiency was self-revealed when operators were performing surveillance procedure DOS 500-07, “Reactor Mode Switch in Shutdown Functional and Scram Auxiliary Functions Valve Operability Test,” Revision 21. The operators manually scrammed the plant with the expectation of no rod movement with the reactor in Mode 5. Ten control rods moved after the mode switch was taken to shutdown. The surveillance test procedure had a prerequisite, Step E.2.a, that stated, “If fuel is in the reactor vessel, then verify all control rods are fully inserted or control rods are removed per DOP 300-18.” Not all the control rods were fully inserted because some had been replaced and were in the process of being vented. The venting procedure opened the 3-0305-101 and 3-0305-102 valves. These were the control rod piston inlet and outlet valves. The open position of these valves allowed a flow path that caused the rods to insert when the scram signal was inserted. The crew knew that the valves were open but did not understand that the equipment lineup would cause the control rods to insert. The operating crew did not understand and therefore did not meet the procedure prerequisite. The primary cause of this violation was related to the cross-cutting area of Human Performance.

The finding was greater than minor because if left uncorrected the failure to adhere to surveillance test prerequisites could become a more significant safety concern. The inspectors completed a Phase 1 significance determination of this issue using IMC 0609, “Significance Determination Process,” Appendix G, Check List 7, dated May 25, 2004. The three areas listed in the checklist that would require a Phase 2 or 3 analysis, and therefore indicate a more significant issue, were not applicable to this finding. Therefore, the inspectors concluded that the finding was of very low safety significance. Operations personnel were temporarily removed from duties, all control rod drive blades involved and adjacent fuel were inspected using cameras, hydraulic control units for the control rod drives were walked down to verify valve positions, and all operations personnel were briefed on this event. This issue was a Non-Cited Violation of Technical Specification 5.4.1, which required the implementation of written surveillance procedures for the control rod drive system. (Section 1R22)

Cornerstone: Mitigating Systems

Green. A self revealed finding of very low safety significance was identified involving a Non-Cited Violation of Technical Specification 5.4.1. On November 13, 2004, a licensee contracted worker failed to follow station procedures and standards and ignored protected pathway equipment signs. This error resulted in the temporary loss of power to station safety related systems. The worker was performing electrical work, when he inadvertently operated the bus 39 to bus 38 crosstie breaker, causing it to trip. Work was stopped, power was restored back to station loads in less than 1 hour, and the worker was counseled. By the end of the report period the licensee had not completed their Apparent Cause Evaluation to further discuss corrective actions. The primary cause of this violation was related to the cross-cutting area of Human Performance.

This finding was greater than minor because if maintenance personnel continued to perform unrestrained work within protected pathway boundaries it would become a more significant safety concern. The finding was of very low safety significance because the operators rapidly restored power to station loads, other mitigating systems were available, and the total exposure time was short. (Section 1R20.1.(2))

Green. On December 15, 2004, the inspectors identified a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI. The licensee failed to take prompt and effective actions regarding the validation of completed surveillance tests after it was identified that maintenance and test equipment (M&TE) used to perform the tests was identified as lost on October 4, 2004. The accuracy of the instrumentation used during the performance of the tests could not be demonstrated. The licensee had knowledge of the problem and the opportunity to re-perform the surveillance tests during a maintenance outage between November 2, and December 10, 2004, and chose not to re-perform the surveillance tests. As corrective action, the licensee prepared an engineering evaluation that gave reasonable assurance that the functions used by the M&TE were within calibration when these tests were performed. The primary cause of this finding was related to the cross-cutting area of Problem Identification and Resolution.

The finding was greater than minor because if left uncorrected the failure to re-perform surveillance testing after M&TE is lost could become a more significant safety concern if it can not be adequately demonstrated that the equipment tested with the M&TE will perform within expected parameters. This finding was of very low safety significance because the inspector identified that portions of other surveillance tests using different, calibrated M&TE, could be combined to show that the installed equipment was satisfactory. (Section 4OA2)

Cornerstone: Barrier Integrity

Green. A self-revealing event, that operators mispositioned a valve in the flow path for draining the Unit 2 torus to the Unit 2 hotwell, was identified on October 8, 2004. Operators failed to return valve 2-1501-35, “U2 Torus to Hotwell Isolation Valve,” to its correct position after completion of Clearance Order 30831 on September 17, 2004. This event was a Non-Cited Violation of TS 5.4.1 having very low safety significance. The primary cause of this violation was related to the cross-cutting area of Human Performance.

The finding was greater than minor, in that, the failure to follow procedures when returning valves to the correct position after being taken out-of-service, if left uncorrected, could become a more significant safety concern. This finding had very low safety significance because the mispositioned valve was identified, returned to the correct position, and the torus level was returned to Technical Specification requirements within the Technical Specification allowed outage time. The involved non-licensed operators were temporarily removed from shift duties. The licensee re-verified a sample of 10 safety related clearance orders; performed a valve lineup on the accessible portions of the high pressure coolant injection, low pressure coolant injection, and core spray systems; and re-verified a sample of the last five clearance orders performed by the individuals involved in this event. No additional issues were identified. (Section 1R04)

Green. A finding was identified by the inspectors involving the failure to adequately perform an operability evaluation. This failure was a Non Cited Violation of 10 CFR 50, Appendix B, Criterion III, "Design Control." On November 13, 2003, the licensee identified in Engineering Evaluation EC34593 that 8 inch diameter 150 lb flanges were installed on the main steam relief valve discharge lines on each unit since construction. The engineering evaluation stated that 300 lb flanges were required. Operability Evaluation 03-013, "Electromatic Relief Valve (ERV) Discharge Piping Flanges," stated that the discharge flanges were operable and no further actions were required. The inspectors reviewed the operability evaluation on August 18, 2004. The inspectors identified that the licensee's evaluation demonstrated that the stresses on the flanges exceeded the Code allowable values, but the licensee's evaluation did not state this fact. The operability evaluation was closed with no specific action required to return the flanges to their design specifications. The primary cause of this violation was related to the cross-cutting area of Human Performance.

The finding was greater than minor because if left uncorrected the failure to perform adequate operability evaluations could become a more significant safety concern. If the inspectors had not intervened, the licensee would not have taken action to bring the relief valve discharge flanges up to Code requirements. As corrective action, the licensee re-performed the evaluation and determined that the flanges were operable, but degraded. The licensee planned further evaluation to make a successful case for Code Committee approval or replace the flanges during the next refueling outage for both Units 2 and 3. To correct the problems with operability evaluations, the licensee

had previously implemented a Technical Rigor program. This finding had very low safety significance because the flanges were determined to be operable. (Section 1R15)

Green. On November 5, 2004, a performance deficiency was self-revealed when electrical maintenance personnel removed the 3 B drywell cooler breaker, that was tagged out-of-service in the racked to test position, to perform a preventive maintenance task. During the performance of Unit 3 Division 1 undervoltage testing, alarm E-4, "DW [drywell] Cooler Blower Trip," on panel 923-5, was received in the control room. A non-licensed operator was dispatched to the breaker cubicle and found the cubicle empty. Electrical maintenance personnel had removed the breaker to perform a preventive maintenance task that was scheduled to be performed after the completion of undervoltage testing. The primary cause of this violation was related to the cross-cutting area of Human Performance.

The finding was greater than minor because if left uncorrected the failure to adhere to clearance order tag requirements and the failure to be aware of plant equipment status prior to re-alignment or removal could become a more significant safety concern. The inspectors completed a Phase 1 significance determination of this issue using IMC 0609, "Significance Determination Process," Appendix G, Check List 7, dated May 25, 2004. The three areas listed in the checklist that would require a Phase 2 or 3 analysis were not applicable to this finding, therefore, the inspectors concluded that the finding was of very low safety significance. The electricians were temporarily removed from duties and counseled, and all electrical maintenance department personnel were briefed on this event. This issue was a Non-Cited Violation of Technical Specification 5.4.1 which required the implementation of written procedures for the control of locking and tagging of plant equipment. (Section 1R20.1.(1))

B. Licensee Identified Findings

No findings of significance were identified.

Report Details

Summary of Plant Status

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

- On October 2, 2004, the unit was taken offline to balance the main generator rotor to reduce high bearing vibration, and returned to full power on October 3.
- On October 28, 2004, load was reduced to 83 percent due to high turbine generator bearing vibration, and remained there for the remainder of the month.
- On November 1, 2004, the unit was taken offline to inspect the main generator rotor due to high bearing vibration and inspection results on Unit 3. A crack on the rotor was detected and repairs were made. In addition, the reactor head was removed and the steam dryer was inspected. The licensee identified a 6 inch crack on an external weld. The dryer was repaired and modifications were made to strengthen the dryer. The unit returned to full power on December 13, 2004.

Unit 3 began the inspection period at 822 MWe (100 percent thermal power).

- On October 3, 2004, load was reduced to 34 percent to perform control rod pattern adjustment and main generator rotor thermal sensitivity testing, and returned to full power on October 6.
- On October 26, 2004, the unit was taken offline to perform a scheduled refueling outage. During the outage, cracking was identified on the steam dryer. The dryer was repaired and modifications were made to strengthen the dryer. The outage was extended to repair a crack that was discovered in the main generator rotor shaft. The unit returned to full power on December 11, 2004.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather (71111.01)

.1 Preparation for Winter Readiness

a. Inspection Scope

The inspectors conducted the following reviews of the licensee's preparations for winter conditions to verify that the plant's design features and implementation of procedures were sufficient to protect mitigating systems from the effects of adverse weather:

- Unit 2 preliminary winter preparations completed as of October 26, 2004.

Documentation for selected risk-significant systems was reviewed to ensure that these systems would remain functional when challenged by inclement weather. Cold weather protection, such as heat tracing, was verified to be in operation where applicable.

This represented one inspection sample.

b. Findings

No findings of significance were identified.

.2 Specific Reviews of Licensee Preparedness for Inclement Weather

a. Inspection Scope

The inspectors performed an assessment of the licensee's implementation of the station's winter readiness process including freeze protection of the 200,000 gallon floor drain surge tank 2/3-2012-359. The inspector also verified that the scuppers installed in 1983 in the roof parapets of the turbine building, reactor building, and crib house were in place in accordance with a commitment described in Updated Final Safety Analysis Report Section 2.4.2. Without the scuppers these roofs would not sustain loadings from local intense precipitation.

This represented two inspection samples.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04Q&S)

.1 Partial System Walkdowns - Unit 2 Torus to Hotwell Isolation Valve Mispositioned

a. Inspection Scope

The inspectors reviewed the circumstances surrounding the mispositioning of 2-1501-35, "U2 Torus to Hotwell Isolation Valve." The valve was found mispositioned on October 8, 2004. The inspectors reviewed Issue Report (IR) 261526; DOP 1600-02, "Torus Water Level Control," Revision 14; M29, Diagram of L.P. [low pressure] Coolant Injection Piping, Sheet 1, Revision CE; and Prompt Investigation 261526. The inspectors interviewed the shift operations supervisor and the shift manager involved with this event.

This represented one inspection sample.

b. Findings

Introduction: A self-revealing event was identified on October 8, 2004, in that operators mispositioned a valve in the flow path for draining the Unit 2 torus to the Unit 2 hotwell. Operators failed to return valve 2-1501-35, "U2 Torus to Hotwell Isolation Valve," to its correct position after completion of Clearance Order 30831 on September 17, 2004. This event was a Non-Cited Violation (NCV) of Technical Specification (TS) 5.4.1 having very low safety significance (Green).

Description: On October 8, 2004, during a high pressure coolant injection (HPCI) quarterly surveillance test, Unit 2 torus water level exceeded the TS high level of -1.5 inches due to water from the condensate storage tank being added to the torus while the HPCI pump was operating on minimum flow. When the operators attempted to pump down the torus level, the level would not decrease. It was later identified that the 2-1501-35 valve was in the wrong position which prevented the torus pump down. The licensee performed a prompt investigation. The licensee concluded that the valve was incorrectly returned to service on September 17, 2004, when completing Clearance Checklist 30831.

Analysis: The inspectors determined that failing to return the 2-1501-35 valve to the correct position after completion of a clearance order was a performance deficiency warranting a significance evaluation. The inspectors concluded the finding was greater than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," issued on June 20, 2003. If left uncorrected, the failure to follow procedures when returning valves to the correct position after being taken out-of-service would become a more significant safety concern. The licensee entered TS 3.6.2.2 Condition A and Emergency Operating Procedure DEOP 200-1, "Primary Containment Control," due to U2 torus level exceeding -1.5 inches. The torus could not be drained to within the TS required level with 2-1501-35 in the wrong position. The mispositioned valve was identified, the valve was returned to the correct position, and the torus level was returned to an acceptable level within the TS allowed outage time.

The inspectors completed a significance determination of this issue using IMC 0609, "Significance Determination Process," Appendix A, dated September 10, 2004. The finding was associated with a degraded containment barrier. For the Phase 1 Screening, the inspectors answered "No" to all three questions under the Barrier Integrity Cornerstone and the issue screened as Green, having very low safety significance.

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 30831 Step 10, required valve 2-1501-35 to be locked in the open position. On September 17, 2004, the operators did not lock open the valve in the required position. The valve was in the locked closed position for about 3 weeks until it was found on October 8, 2004. The valve was restored to its correct position. The involved non-licensed operators were temporarily removed from shift duties. The licensee reverified a sample of 10 safety related clearance orders; performed a valve lineup on the accessible portions of the high pressure coolant injection, low pressure coolant injection, and core spray systems; and re-verified a sample of the last five clearance orders performed by the individuals involved in this event. No additional issues were identified. The licensee entered this into the corrective action program as IR 261526. Because this violation was of very low safety significance and it was entered into the corrective action program, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy. **(NCV 05000237/2004013-01)**

.2 Complete System Walkdown

a. Inspection Scope

The inspectors performed one complete semi-annual walkdown of the Unit 3 250 VDC station battery system. The inspectors reviewed the electrical and mechanical system checklists and drawings to ensure all vital components in this system were properly aligned. 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 IRs to review past issues that had been identified and their corrective actions.

This represented one inspection sample.

b. Findings

No findings of significance were identified.

1R06 Flooding (71111.06)

a. Inspection Scope

The inspectors reviewed the Updated Final Safety Analysis Report Section 3.4.1.2 for internal flood analysis and reviewed the licensee’s procedures for internal flooding. The inspectors walked down the Unit 2 and Unit 3 component cooling service water (CCSW) pump areas and vaults to verify compliance with the licensee’s Updated Final Safety Analysis Report and reviewed the licensee’s previously implemented corrective actions for deficiencies associated with internal flood protection. In addition, the inspectors reviewed the final results and observed portions of DOS 1500-20, “CCSW Pump Vault Penetration Surveillance Testing,” Revision 0, on Unit 2.

This represents two inspection samples.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07A)

a. Inspection Scope

On December 30, 2004, the inspectors reviewed the results of the heat removal surveillance test performed on the Unit 2 isolation condenser heat exchanger in September 2003, to determine if the heat removal rate of the heat exchanger was acceptable.

This represented one inspection sample.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

a. Inspection Scope

On November 18, 2004, the inspectors conducted a review of the implementation of the licensee's inservice inspection program for monitoring degradation of the reactor coolant system boundary and the risk significant piping system boundaries.

Specifically, the inspectors conducted a record review of the following three nondestructive examination activities to evaluate compliance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requirements and to verify that indications and defects were dispositioned in accordance with the ASME Code:

1. Ultrasonic examination of a Unit 3 core spray tee to pipe weld # 3/1/1404-10/10-44C;
2. Ultrasonic examination of a Unit 3 shutdown cooling tee to pipe weld # 3/1/1001B-16/SDA-04F; and
3. Ultrasonic examination of a Unit 3 low pressure coolant injection elbow to elbow weld # 3/2/1507-24/24-3.

The inspectors also reviewed the following examination from the previous outage, with recordable indications that have been accepted by the licensee for continued service, to verify that the licensee's acceptance for continued service was in accordance with the ASME Code:

Radiograph of reactor head spray line 2-0304-2-1/2"-A, flange to pipe weld 1, (porosity and tungsten inclusions).

The inspectors reviewed the following two pressure boundary welds for Class 1 or 2 systems which were completed since the beginning of the previous refueling outage, to verify that the welding acceptance (e.g., radiography) and preservice examinations were performed in accordance with ASME Code requirements:

1. Code Class 1, Reactor head spray line 2-0304-2-1/2"-A, flange to pipe weld 1; and
2. Code Class 1, Reactor head spray line 2-0304-2-1/2"-A, pipe to pipe weld 2.

The inspectors reviewed a sample of inservice inspection related problems documented in the licensee's corrective action program to assess conformance with 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements. In addition, the inspectors verified that the licensee correctly assessed operating experience for applicability to the Inservice Inspection group. This represented one inspection sample.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11Q)

a. Inspection Scope

The inspectors observed an evaluation of an operating crew on December 13, 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 observed that the licensee’s evaluators identified that the operators were not able to complete the tasks in accordance with applicable plant procedures and took the appropriate actions to remediate and re-evaluate the individuals that failed the scenario prior to the operators returning to duties on shift. 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 issue reports written regarding licensed operator requalification training were entered into the licensee’s corrective action program with the appropriate significance characterization.

This represented one inspection sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

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

- Unit 2 & Unit 3 high pressure coolant injection systems.

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

This represented one inspection sample.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors evaluated the 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 DOS 6600-07; "Testing LPCI Swing Bus Protective Relays And Auto Transfer Function (Time Delay Relay Found Out Of Acceptance Criteria)"; Revision 18;
- Electrical bus outage on Unit 2;
- Electrical bus outage on Unit 3; and
- Unit 3 core spray modification work.

These reviews constitute four inspection samples.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance Related to Non-routine Evolutions and Events (71111.14)

a. Inspection Scope

The inspectors reviewed personnel performance during planned and unplanned plant evolutions. The review was performed to ascertain that operators' responses were in accordance with the required procedures.

- Shutdown cooling automatically isolated on high temperature during cooldown of Unit 3 reactor just before being placed in service.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The following operability evaluation was reviewed during the last inspection period, Units 2 & 3 Electromatic Relief Valve Discharge Flanges (OE 03-013, Rev. 1), and the sample was taken credit for in Inspection Report 05000237;249/2004-010. At that time the inspectors brought deficiencies within the closed operability evaluation to the attention of the licensee. Specifically, the discharge flanges were operable, but degraded and required repair. The licensee attempted to perform more precise calculations to demonstrate that the discharge flanges were not degraded. This attempt extended over the end of the last inspection period. The licensee was unsuccessful in this attempt. The following finding was based on previous review.

b. Findings

Introduction: A Green finding was identified by the inspectors involving the failure to adequately perform an operability evaluation. This failure was a violation of 10 CFR 50, Appendix B, Criterion III, "Design Control."

Description: On November 13, 2003, the licensee identified in Engineering Evaluation EC34593 that 8 inch diameter 150 lb flanges were installed on the main steam relief valve discharge lines on each unit. The engineering evaluation stated that 300 lb flanges were required. Operability Evaluation 03-013, “Electromatic Relief Valve (ERV) Discharge Piping Flanges,” stated that the discharge flanges were operable and no further actions were required. The inspectors reviewed the operability evaluation on August 18, 2004, which resulted in the following observation.

Operability Evaluation 03-013 paragraph 2.2(a) stated, in-part: “EC EVAL 345943 demonstrated that the flanges will maintain their structural integrity and preserve the piping pressure boundary. Therefore, there is no need to postulate the failure of this piping due to failure of the flanges. Although the design margins are reduced, the flanges continue to meet the code allowable stresses.”

However, per the calculations in Engineering Evaluation 345943, the 8 inch flanges were shown to not meet Section III of the ASME B&PV Code 1977 Edition through Summer 1977 Addenda allowable stresses. In addition, the licensee’s calculations did not include all the mechanical loads that would be placed on the flanges and the bolts securing the flanges.

Based on the inspectors’ review the licensee re-opened the operability evaluation and re-performed the analysis, including the additional mechanical loads. The calculations confirmed that loading on the flanges and flange bolts exceeded those allowed by Code, but that the yield stresses were not exceeded for the material used. Therefore the relief valve discharge flanges were operable but degraded.

The licensee stated that the proposed repair would involve adding a spacer between the flanges to increase strength and using larger bolts of different material for added strength. The licensee stated that this repair method was proposed to reduce dose that would be accumulated in cutting and re-welding of the flanges. This proposed repair needed to be brought before an ASME Code committee for approval.

Analysis: The inspectors determined that the failure to perform an adequate operabilty evaluation was a performance deficiency warranting a significance evaluation. The inspectors concluded that the finding was greater than minor because if left uncorrected the failure to perform adequate operability evaluations could become a more significant safety concern. If the inspectors had not intervened, the licensee would not have taken action to bring the relief valve discharge flanges up to Code requirements.

The inspectors completed a Phase 1 significance determination of this issue using IMC 0609, “Significance Determination Process (SDP),” Appendix A, dated September 10, 2004. The inspectors determined that this finding impacted the Barrier Integrity Cornerstone. The relief valve flanges were in the torus atmosphere space. Any failure in the flange material would result in steam entering the atmosphere above the water instead of being condensed under the water. The steam would more rapidly increase the pressure within the torus. The inspectors entered the Barrier Integrity Cornerstone column of the Phase I SDP sheet and answered “No” to all three questions. Therefore the inspectors concluded that the finding was of very low safety significance. The primary cause of this finding was related to the cross-cutting area of Human Performance.

Enforcement: 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that measures be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. This design basis requirement was to maintain Electromatic Relief Valve Discharge flange stresses below Code acceptance limits.

Contrary to the above, from November 13, 2003, to August 18, 2004, the operability evaluation for the stresses on the Electromatic Relief Valve discharge flanges on both units, did not include all required piping reaction forces nor properly identify that flanges had exceeded Code stress limits. Consequently, the licensee did not initiate corrective action to modify operable but degraded flanges until the inspectors brought it to management's attention. As corrective action, the licensee re-performed the evaluation and determined that the flanges were operable, but degraded. The licensee planned further evaluation to make a successful case for Code Committee approval or replace the flanges during the next refueling outage for both Units 2 and 3. To correct the problems with operability evaluations, the licensee had previously implemented a Technical Rigor program. Because this violation was of very low safety significance and because the issue was entered into the licensee's corrective action program (Issue Report 245856), the issue is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (**NCV 05000237/2004013-02; 05000249/2004013-02**).

1R16 Operator Workarounds (71111.16)

.1 Semi-annual Review of the Cumulative Effects of Operator Workarounds

a. Inspection Scope

The inspectors also performed a semi-annual review of the cumulative effects of operator workarounds. The inspectors reviewed the cumulative effects of workarounds on the reliability, availability, and potential for improper operation of the system. Additionally, reviews were conducted to determine if the workarounds could increase the possibility of an initiating event, affect multiple mitigating systems, or impact the operators' ability to respond to accidents or transients.

This represented one inspection sample.

b. Findings

No findings of significance were identified.

.2 Quarterly Review

a. Inspection Scope

The inspectors assessed the following three operator workaround issues to determine the potential effects on the functionality of the corresponding mitigating system:

- Keep fill alignment on Unit 2 emergency core cooling system;
- Unit 2 generator rotor vibration; and
- Realignment of Unit 3 high pressure coolant injection system to the condensate storage tank.

During this inspection, the inspectors reviewed the technical adequacy of the workaround documentation against the Updated Final Safety Analysis Report and other design information to assess whether the workaround conflicted with any design basis information. The inspectors compared the information in abnormal or emergency operating procedures to the workaround information to ensure that the operators maintained the ability to implement important procedures when needed. Multiple entries into the corrective action program were also reviewed to ensure that the operator workarounds had been entered into this process.

This represented three inspection samples.

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 system was 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:

- Unit 3 emergency diesel generator.

This represented one inspection sample.

b. Findings

No findings of significance were identified.

1R20 Outage Activities (71111.20)

.1 Refuel Outage

a. Inspection Scope

The inspectors evaluated outage activities for a scheduled Unit 3 refueling outage (D3R18) that began on October 26, 2004, and ended on December 7, 2004. The inspectors reviewed activities to ensure that the licensee considered risk in developing, planning, and implementing the outage schedule.

The inspectors observed or reviewed the reactor shutdown and cooldown, outage equipment configuration and risk management, electrical lineups, selected clearances, control and monitoring of decay heat removal, control of containment activities, startup and heatup activities, and identification and resolution of problems associated with the outage. Additionally, the inspectors conducted tours of the Unit 3 drywell, both at the beginning of the outage and prior to drywell closure, to ensure that the material conditions of the space and components were acceptable.

During this outage the licensee conducted inspections of the generator rotor shaft and the steam dryer. Both the rotor and the dryer were found to have significant cracks in them, and were repaired with modifications made to prevent cracks in the future.

The inspectors reviewed the circumstances surrounding the physical removal of a breaker from a cubicle while it was out-of-service in the test position on November 5, 2004. In addition, the inspectors reviewed the circumstances surrounding a failure to observe protected pathway requirements that resulted in a trip of bus 38 on November 13, 2004.

b. Findings

There were two findings associated with maintenance work conducted during the refueling outage.

(1) Removal of the 3D Drywell Cooler Breaker While Out-of-Service in Rack-to-Test

Introduction: A Green finding was self revealed when electricians removed the 3D drywell cooler breaker while it was tagged out-of-service in the racked-to-test position on November 5, 2004. The finding was an NCV of TS 5.4.1. for failing to follow procedures.

Description: During the performance of Unit 3 Division 1 undervoltage testing, alarm E-4, “DW [drywell] Cooler Blower Trip,” on panel 923-5, was received in the control room. A non-licensed operator was dispatched to the breaker cubicle and found the cubicle empty. Electrical maintenance personnel had removed the breaker to perform a preventive maintenance task that was scheduled to be performed after the completion of undervoltage testing.

The electrical maintenance foreman had observed that the out-of-service tag, clearance order (CO) 29577, had been hung and approved by operations management personnel, and assumed the preventive maintenance task was ready to work. The electrical maintenance foreman did not review the CO, nor did the electricians in the field verify the out-of-service position on the CO card or the position of the breaker prior to removing it from the cubicle.

Analysis: The inspectors determined that the failure to ensure that the breaker was in the correct position prior to physically removing it from the cubicle was a performance deficiency warranting a significance evaluation. The inspectors concluded the finding was greater than minor in accordance with IMC 0612, “Power Reactor Inspection Reports,” Appendix B, “Issue Screening,” issued on June 20, 2003. The inspectors concluded that the finding was greater than minor because, if left uncorrected, the failure to adhere to clearance order tag requirements and the failure to be aware of plant equipment status prior to re-alignment or removal could become a more significant safety concern. The primary cause of this violation was related to the cross-cutting area of Human Performance.

Unit 3 was in Mode 5 and the drywell coolers were not needed when this event occurred. There were no personnel injuries or equipment damage associated with this event. The results of the Unit 3 undervoltage surveillance test in progress at the time the breaker was removed were not affected by the equipment re-alignment. The inspectors completed a Phase 1 significance determination of this issue using IMC 0609, “Significance Determination Process,” Appendix G, Check List 7, dated May 25, 2004. The three areas listed in the checklist that would require a Phase 2 or 3 analysis were not applicable to this finding, therefore, the inspectors concluded that the finding was of very low safety significance. This finding affected the Barrier Integrity Cornerstone.

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 requirement was OP-MW-109-101, "Clearance and Tagging," Revision 2. Step 5.1.1, stated, "No device or equipment shall be operated while CO [clearance order] tags are attached...," and Step 5.2.2, stated, "a component with a danger tag attached to it shall not be removed from a system." Clearance Checklist 29577, Step 8, required the 3B drywell cooler breaker to be in the racked-to-test position. On November 5, 2004, electricians removed the 3B drywell cooler breaker from its cubicle while the breaker was danger tagged in the rack-to-test position. As immediate corrective action, the breaker was restored and the electricians were temporarily removed from duties and counseled on their actions. In addition, the electrical maintenance department was briefed on this event. The licensee entered this into the corrective action program as IR 270871. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy.
(NCV 05000249/2004013-03)

(2) Failure to Observe Protected Pathway Requirements Results in Loss of Bus 38

Introduction: A Green finding was self revealed as an NCV of TS 5.4.1. The finding involved the licensee's failure to adequately ensure that a contract worker followed station standards while working in an area flagged with a protected pathway sign. A human performance error resulted in a momentary trip of a 480 V breaker and the loss of safety related loads.

Description: On November 13, 2004, during the Unit 3 refueling outage, modifications to replace transformer 35 and transformer 38 were performed. The licensee had a number of protected pathway barriers on several electrical buses due to planned maintenance work on Division 1 electrical distribution equipment. The performance of this work placed Unit 2 electrical distribution in an elevated risk condition (yellow). The work performed on transformer 35 and 38 consisted of energizing the buses via the crosstie breakers, taking the transformers out-of-service, removing interferences, disconnecting the transformers, removing the radiators, removing the old transformers, installing the new transformers and radiators, installing interferences that were removed, and connecting and testing of the new transformers. Bus 38 was energized from bus 39 through a crosstie breaker. The licensee had placed a protected pathway barrier in front of bus 39 to ensure work was not initiated on the wrong equipment. However, an electrician was placing foreign material barrier plates and other mounting hardware into a metal bucket, which was almost full, and which was positioned near the bus 39 to bus 38 crosstie breaker's trip pushbutton. As the electrician tossed the last part into the bucket, it came into contact with the crosstie breaker and subsequently tripped the power supply to bus 38; Unit 3 received a full Group II and III isolation, Reactor Protection System B channel ½ scram, and an instrument bus transfer to bus 35-2. As a result, operations personnel entered a number of Dresden Operating Abnormal Procedures. The licensee restored power to the station loads that were lost.

Upon the licensee's investigation into this matter, it was determined that the contract worker had turned away and lost visual contact with the bucket while still in motion and prior to release of the part. The electrician did not see the plate bump into the breaker, however since the breaker was scarfed, his action was considered to be the most probable source of the breaker trip. The licensee's prompt investigation stated that the electrician failed to identify risk to the protected path equipment

Analysis: The inspectors used Inspection Manual Chapter (IMC) 0612, Appendix B, “Issue Screening,” dated June 20, 2003, and determined that the mechanic’s failure to implement station standards by inadvertently operating protected equipment and disregarding the protected pathway sign was a performance deficiency and it was more than minor. The inspectors concluded that this issue was more than minor because it was associated with the Reactor Safety cross-cutting area of Human Performance and affected the Mitigating Systems objective to ensure the availability of safety related systems. The inspectors completed a significance determination of this issue using IMC 0609, Appendix G, Attachment 1, “Significance Determination of Reactor Inspection Findings for at Shutdown Operations,” dated May 25, 2004. The inspectors completed phase 1 operational checklist 7, “BWR Refueling Operation with RCS Level > 23 feet,” and answered “No” to questions requiring phase 2 or phase 3 analyses. The inspectors determined that this finding was of very low safety significance. The primary cause of this violation was related to the cross-cutting area of Human Performance.

Enforcement: Technical Specification 5.4.1 required, in part, that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, Revision 2, Appendix A, February 1978, Section 9 recommends procedures for performing maintenance. One of the documented instructions that implemented the TS was Work Order 0400804-07, Step 14, which stated, “Supervisor and craft personnel are to maintain compliance to the latest procedures and Nuclear Generation Group Maintenance Standards.” The craft personnel did not comply with station’s maintenance department fundamentals which were bounded by the station’s standards. The electrician did not follow protected pathway requirements by tossing a metal plate that bounced and made contact with the bus 38 to bus 39 crosstie breaker’s trip pushbutton while work was being performed on systems that would lead to a loss of redundancy in critical plant functions. As an immediate corrective action, work was stopped, power was restored back to station loads, and the worker was counseled. At the end of the report period, the licensee had not completed their Apparent Cause Evaluation to further discuss licensee’s corrective actions.

Because this issue is of very low safety significance and has been entered into the licensee’s corrective action program as Issue Report 273150, this violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy. (**NCV 05000249/2004013-04**)

.2 Unit 2 Maintenance Outage

a. Inspection Scope

The inspectors evaluated outage activities for an unscheduled Unit 2 maintenance outage that began on November 1, 2004, and ended on December 12, 2004, in order for the licensee to attempt to make repairs to correct a recurring high vibration problem on the number 9 inboard generator bearing. The licensee had identified a crack on the Unit 3 generator rotor shaft during inspections conducted during the refueling outage discussed above. The licensee suspected there might be an identical crack on Unit 2. The reactor was shut down and the licensee did identify an identical crack on the Unit 2 generator rotor. The licensee had identified a crack in a weld on the outside upper portion of the Unit 3 steam dryer during scheduled inspections. While Unit 2 was shutdown, the licensee decided to inspect the Unit 2 steam dryer and identified a crack similar to the one on Unit 3. Both the rotor and the dryer were repaired. Both were modified to prevent cracking in the future. The inspectors reviewed activities to ensure that the licensee considered risk in developing, planning, and implementing the outage schedule.

The inspectors observed or reviewed the reactor shutdown and cooldown, outage equipment configuration and risk management, electrical lineups, selected clearances, control and monitoring of decay heat removal, control of containment activities, startup and heat-up activities, and identification and resolution of problems associated with the outage.

b. Findings

There were no findings of significance during this outage.

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 the 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:

- Unit 2 DOS 2300-04, “High Pressure Coolant Injection (HPCI) Testable Check Valve Manual Full Stroke Operablility Test,” Revision 9;
- Unit 3 DOS 7000-04, “Local Leak Rate Testing of Double Gasket Seals,” Revision 01;
- Unit 3 DOP 2000-24, “Drywell Sump Operation,” Revision 13; and
- Unit 3 DOS 0500-07, “Reactor Mode Switch in Shutdown Functional and Scram Auxiliary Functions Valve Operability Test,” Revision 21.

This represented four inspection samples.

b. Findings

Introduction: A Green finding was self revealed when operators were performing surveillance procedure DOS 500-07, “Reactor Mode Switch in Shutdown Functional and Scram Auxiliary Functions Valve Operability Test,” Revision 21. The operators manually scrammed the plant with the expectation of no rod movement with the reactor in Mode 5. Ten control rods moved after the scram switches were operated. The finding was a violation of TS 5.4.1 for failure to follow procedures. The finding increased the probability of damage to the reactor fuel due to control rod motion among unsupported fuel bundles.

Description: Operating surveillance test DOS 500-07 was performed on Unit 3 on November 12, 2004, to perform post-maintenance testing on the 3-0302-157B scram discharge volume drain valve, and the 3-0302- 20A and 20B scram dump valves. Unit 3 was in Mode 5 with fuel partially off loaded and no core alterations were in progress, when the mode switch was taken to the shutdown position. Ten control rods fully or partially inserted. The control rod movement was unexpected. The surveillance test procedure had a prerequisite, Step E.2.a, that stated, “If fuel is in the reactor vessel, then verify all control rods are fully inserted or control rods are removed per DOP 300-18.” Not all the control rods were fully inserted because some had been replaced and were in the process of being vented. The operating crew did not understand and therefore did not meet the procedure prerequisite.

Analysis: The inspectors determined that the failure to ensure that the plant conditions met the prerequisites for the surveillance test was a performance deficiency warranting a significance evaluation. The inspectors concluded that the finding was greater than minor because, if left uncorrected, the failure to adhere to surveillance test prerequisites and the failure to be aware of plant equipment status could become a more significant safety concern.

The inspectors completed a Phase 1 significance determination of this issue using IMC 0609, “Significance Determination Process,” Appendix G, Check List 7, dated May 25, 2004. The three areas listed in the checklist that would require a Phase 2 or 3 analysis were not applicable to this finding, therefore the inspectors concluded that the finding was of very low safety significance. Procedure DOP 0300-04 included a caution that when inserting control rods for the first time after coupling, then the blade guide should be visually checked to verify it is not rising out of the core. Unexpected movement of the control rods could have resulted in blade guide movement which may have resulted in fuel damage. Therefore, the inspectors concluded that this finding was within the Initiating Events Cornerstone.

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 8.b.2.k listed surveillance tests for the control rod drive system. Surveillance test procedure DOS 500-07, “Reactor Mode Switch in Shutdown Functional and Scram Auxiliary Functions Valve Operability Test,” Revision 21, Step E.2.a, stated, “If fuel is in the reactor vessel, then verify all control rods are fully inserted or control rods are removed per DOP 300-18.” Contrary to the above, on November 12, 2004, prior to the performance of DOS 500-7, the operators in the control room did not verify that all control rods were fully inserted or removed. As immediate corrective action, involved Operations personnel were temporarily removed from duties, all control rod drive blades involved and adjacent fuel were inspected using cameras, hydraulic control units for the control rod drives were walked down to verify valve positions, and all operations personnel were briefed on this event. Because this violation was of very low safety significance and it was entered into the licensee’s corrective action program (IR 273067), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy. **(NCV 05000249/2004013-05)**

1R23 Temporary Modification (71111.23)

a. Inspection Scope

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

- Temporary Configuration Change Package No. 349380, “Install Bleeder Valve on the Unit 2 ‘A’

Condenser Hood Low Vacuum Sensing Line to Prevent Moisture Buildup," Revision 0.

This represented one inspection sample.

b. Findings

No findings of significance were identified.

1EP6 Drill and Training Evaluations (71114.06)

December 13, 2004, Emergency Preparedness Performance Indicator Drill

a. Inspection Scope

The inspectors observed station personnel during a licensee-only-participation emergency preparedness drill exercise on December 13, 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.

This represented one inspection sample.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

.1 Plant Walkdowns and Radiation Work Permit Reviews

a. Inspection Scope

The inspectors selectively reviewed the licensee's access controls and survey data for the following work areas located within radiation, high radiation and locked high radiation areas in the plant to determine if radiological controls, postings, and barricades were adequate:

- Unit 3 Drywell;
- Unit 3 Reactor Water Cleanup System Pump Aisle;
- Unit 3 Low and High Pressure Heater Bays; and
- Unit 3 Steam Dryer/Separator Pit.

The inspectors reviewed the radiation work permits (RWPs) that governed access to these areas and that provided radiological information to ensure the work control instructions and control barriers that had been specified were adequate. The inspectors also walked down and surveyed (using an NRC survey meter) selected areas in the Unit 2 and Unit 3 Reactor, Turbine, and Radwaste Buildings to verify that radiological conditions were consistent with area postings and controls.

These reviews represented one inspection sample.

b. Findings

No findings of significance were identified.

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

.1 Inspection Planning

a. Inspection Scope

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

The inspectors reviewed the Unit 3 refueling outage (D3R18) work and the associated work activity exposure and time/labor estimates for the following ten work activities which were likely to result in the highest personnel collective exposures or were otherwise radiologically significant activities:

- Reactor Water Cleanup System Maintenance Activities;
- Reactor Disassembly/Reassembly and Related Activities;
- Dryer Modification - Diver Support Crew Activities;
- Nuclear Instrumentation System Maintenance Activities;
- Drywell Main Steam Isolation Valve Maintenance;
- Drywell Main Steam Safety, Electromatic and Target Rock Valve Maintenance;
- Drywell In-Service Inspections;
- "B" Recirculation Pump and Motor Maintenance Activities;
- Drywell Insulation Maintenance Activities; and
- Steam Dryer Modification Diving Activities.

The inspectors determined site specific trends in collective exposures based on plant historical exposure and source term data. The inspectors reviewed procedures associated with maintaining occupational exposures ALARA and assessed those processes used for D3R18 to project dose and track work activity exposures.

These reviews represented four inspection samples.

b. Findings

No findings of significance were identified.

.2 Radiological Work Planning

a. Inspection Scope

The inspectors obtained the licensee's list of Unit 3 outage work activities ranked by estimated exposure and reviewed the following radiologically significant work activities:

- Reactor Disassembly/Reassembly and Related Activities (RWP 10004164);
- Dryer Modification - Diver Support Crew Activities (RWP 10004168);
- Nuclear Instrumentation System Maintenance Activities (RWP 10004191);
- Drywell Main Steam Isolation Valve Maintenance (RWP 10004194);
- Drywell Main Steam Safety, Electromatic and Target Rock Valve Maintenance (RWP 10004196);
- Drywell In-Service Inspections (RWP 10004206);
- "B" Recirculation Pump and Motor Maintenance Activities (RWP 10004208);
- Drywell Insulation Maintenance Activities (RWP 1000 4209); and
- Steam Dryer Modification Diving Activities (RWP 10004165).

For each of the activities listed above, the inspectors reviewed the ALARA Plan and associated total effective dose equivalent (TEDE) ALARA evaluation, time/labor estimates and dose projections, and exposure mitigation criteria in order to verify that the licensee had established radiological engineering controls that were based on sound radiation protection principles in order to achieve occupational exposures that were ALARA. This also involved determining that the licensee had reasonably grouped the radiological work into activities that were based on historical precedence, industry norms, and/or special circumstances.

The inspectors compared the exposure results achieved through the first half of the scheduled 30-day outage including the dose rate reductions and person-rem expended with the doses projected in the licensee's ALARA planning for the above listed work and for other selected outage activities. Reasons for inconsistencies between intended (projected) and actual work activity doses were evaluated to determine if the activities were planned reasonably well and to ensure the licensee identified any work planning deficiencies.

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

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

The inspectors evaluated if work activity planning included consideration of the benefits of dose rate reduction activities such as shielding provided by water filled components/piping, system flushing and hydrolazing and sequencing of scaffold and shielding installation/removal along with logic-ties in the work scheduling process in order to maximize dose reduction.

The licensee's work in progress reports were reviewed by the inspectors for those outage jobs that approached their respective dose estimates or that were otherwise evaluated by the ALARA staff to assess work progress. The reports were reviewed to verify that the licensee could identify problems and address them as work continued. Outage jobs of any dose significance that exceeded 125 percent of their dose projections or were anticipated to exceed dose projections were also reviewed to ensure work was suspended, if warranted, and identified problems were entered into the licensee's corrective action program consistent with the licensee's procedure.

These reviews represented seven inspection samples.

b. Findings

No findings of significance were identified.

.3 Verification of Dose Estimates and Exposure Tracking Systems

a. Inspection Scope

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

The inspectors reviewed the licensee's process for adjusting outage exposure estimates when unexpected changes in scope, emergent work or other unanticipated problems were encountered which could significantly impact worker exposures. This included determining that adjustments to estimated exposure (intended dose) were based on sound radiation protection and ALARA principles and not adjusted to account for failures to effectively plan or control the work. The frequency and scope of these adjustments was reviewed to evaluate the adequacy of the original ALARA planning process.

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

These reviews represented three inspection samples.

b. Findings

No findings of significance were identified.

.4 Job Site Inspections and ALARA Control

a. Inspection Scope

The inspectors observed the following three jobs that were being performed in high or locked high radiation areas that potentially represented significant radiological risk to workers:

- Steam Dryer Modification Diving Activities;
- Drywell Permanent Shielding; and
- "B" Recirculation Pump Maintenance Activities.

The licensee's use of ALARA controls for these work activities was evaluated using the following:

The licensee's use of engineering controls to achieve dose reductions was evaluated to verify that procedures and controls were consistent with the licensee's ALARA reviews.

Job sites were observed to determine if workers were cognizant of work area radiological conditions and utilized low dose waiting areas and were effective in maintaining their doses ALARA by moving to the low dose waiting area when subjected to temporary work delays.

The inspectors reviewed the radiation exposures of individual divers that were involved in the steam dryer modification project to determine whether significant exposure variations existed that may be attributed to poor ALARA practices or to radiation protection staff work oversight or dose monitoring problems. The inspectors also reviewed selected whole body count results and internal dose assessments for several workers that had small intakes during the first half of the outage to evaluate the adequacy of these ongoing assessments.

These reviews represented three inspection samples.

b. Findings

No findings of significance were identified.

.5 Source Term Reduction and Control

a. Inspection Scope

The inspectors reviewed licensee records to understand historical trends and current status of plant source terms. The inspectors discussed the plant's source term with radiation protection and chemistry staffs to determine if the licensee has developed a good understanding of the input mechanisms and the methodologies and practices necessary to achieve reductions in source term.

The inspectors reviewed selected exposure reduction initiatives taken for D3R18 such as flushing, use of shielding, and hydrolazing. The inspectors discussed the water chemistry control initiatives implemented by the licensee and its impact on source term reduction compared to industry practices.

The inspectors reviewed the licensee's 2004 - 2007 Source Term Reduction Plan and discussed its ongoing implementation with members of the radiation protection staff. The inspectors determined if specific sources had been identified by the licensee for exposure reduction initiatives and that priorities were established or being considered for the implementation these actions.

These reviews represented three inspection samples.

b. Findings

No findings of significance were identified.

.6 Radiation Worker Performance

a. Inspection Scope

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

This review represented one inspection sample.

b. Findings

No findings of significance were identified.

.7 Identification and Resolution of Problems

a. Inspection Scope

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

The inspectors verified that identified problems were entered into the corrective action program for resolution, and that they had been properly characterized, prioritized, and were being addressed. This included outage ALARA critiques/lessons learned for exposure performance from the licensee’s previous Unit 3 refueling outage in 2002.

Corrective action reports (issue reports (IRs)) generated during the first half of the Unit 3 outage related to the ALARA program were selectively reviewed and staff members were interviewed to verify that follow-up activities were being conducted in a timely manner commensurate with their importance to safety and risk using the following criteria:

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

The licensee's corrective action program was also reviewed to determine if repetitive deficiencies in problem identification and resolution had been addressed.

These reviews represented three inspection samples.

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety

2PS3 Radiological Environmental Monitoring Program (71122.03)

.1 Followup on Condensate Storage Tank Underground Line Leak

a. Inspection Scope

The inspectors reviewed the licensee's water sampling results and sampling locations following the identification of an underground pipe leak from the condensate storage tanks in late August 2004. Tritium and other radio nuclide sampling results were reviewed for various periods in 2004, focusing on tritium analyses for September and October 2004. Sampling results reviewed included samples collected from protected area (inside fence-line) shallow and deep wells, protected area storm drains, the licensee's onsite waste water treatment facility effluent, those areas excavated to repair the leaking line, the Unit 1 intake canal, and selected licensee monitoring wells located outside the protected area south of the plant. The sampling data was reviewed to determine if samples were collected from representative locations so as to demonstrate Offsite Dose Calculation Manual (ODCM), TS and 10 CFR Part 20 concentration limits were met. Additionally, the assumptions and the results of licensee calculations of offsite dose associated with the leak were reviewed to verify compliance with ODCM limits and 10 CFR 50, Appendix I objectives.

These reviews represented one inspection sample.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA2 Identification and Resolution of Problems (71152)

.1 Routine Quarterly Review

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 issue reports and attending daily issue report review meetings.

b. Findings

No findings of significance were identified.

.2 Unit 2 Reactor Pressure Outside of Band High

Introduction

The inspectors reviewed Issue Report (IR) 185002. During leak testing of the Unit 2 reactor vessel at the close of the refueling outage in 2003, licensee personnel failed to monitor the correct indications for reactor pressure. Reactor pressure rose to the point that a relief valve lifted causing reactor pressure to drop rapidly.

This represents one annual system sample.

a. Effectiveness of Problem Identification

(1) Inspection Scope

The inspectors reviewed IR 185002 and the associated root cause report to verify the licensee's identification of the problems was complete, accurate, and timely, and that the consideration of extent of condition review, generic implications, common cause, and previous occurrences was adequate.

(2) Issues

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

b. Prioritization and Evaluation of Issues

(1) Inspection Scope

The inspectors reviewed IR 185002 and the associated root cause report. The inspectors considered the licensee's evaluation and disposition of performance issues, and application of risk insights for prioritization of issues.

(2) Issues

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

c. Effectiveness of Corrective Actions

(1) Inspection Scope

The inspectors reviewed the corrective actions which resulted from the root cause report associated with IR 185002 to determine if the issue report addressed generic implications and that corrective actions were appropriately focused to correct the problem. In addition, the inspectors observed several significant evolutions during the Unit 3 refueling outage and the Unit 2 maintenance outage that occurred this period.

(2) Issues

There were no issues in the area of Effectiveness of Corrective Actions.

.3 Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a review of the licensee's corrective action program (CAP) and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review was focused on unplanned limiting condition of operation (LCO) entries, but also considered the results of daily inspector CAP item screening discussed in Section 4OA2.2 above, licensee trending efforts, and licensee human performance results. The inspectors' review nominally considered the 6 month period of July 2004 through December 2004, although some examples expanded beyond those dates when the scope of the trend warranted.

Corrective actions associated with a sample of the issues identified in the licensee's trend report were reviewed for adequacy. The inspectors also evaluated the report against the requirements of the licensee's CAP as specified in LS-AA-120, "Issue Identification and Screening Process," and 10 CFR 50, Appendix B. The inspectors evaluated the licensee trending methodology and observed that the licensee did not perform a detailed review on a regular basis. The identification of trends was left up to the individual departments. The licensee did not routinely reviewed cause codes, involved organizations, key words, and system links to identify potential trends in the CAP data.

b. Findings

The inspectors identified one trend in regard to the control of maintenance and test equipment (M&TE). The licensee lost a total of 23 pieces of M&TE during the period of June 30, to December 1, 2004. The licensee recognized an Exelon wide trend in the loss of M&TE in 2003 and performed a common cause analysis (188112). The corrective actions the licensee developed from this common cause analysis consisted of generating and implementing an expectations statement on the control of M&TE, performing a management observation of the control of M&TE once per month, reviewing information on M&TE control at the plan of the day meeting once per month, and installing a card reader and camera on the M&TE storage facility. The installation of the card reader and camera have not yet been performed. The other corrective actions have not been effective as evidenced by the continuing loss of M&TE equipment. During the review of corrective actions for lost M&TE the inspectors identified one finding.

.4 Lost Maintenance and Test Equipment

b. Findings

Introduction: The inspectors identified an NCV of 10 CFR 50, Appendix B, Criterion XVI, having very low safety significance (Green), in that the licensee failed to take prompt and effective actions regarding the validation of completed surveillance tests after it was identified that a piece of the maintenance and test equipment (M&TE) used to perform the tests was lost.

Description: Issue Report 259982, “Recalibrations Needed Due To Lost Measuring and Test Equipment,” stated that fluke 189 had been lost. A fluke is a digital multi-meter that measures alternating current (AC) and direct current (DC) voltage, AC and DC current, and resistance. When an M&TE item is lost, such as a fluke, the applicable surveillance tests it was used to perform become questionable, requiring re-performance of the test or tests, or a documented engineering evaluation to provide justification for the plant instrumentation’s accuracy and the instrument’s ability to meet its safety function.

The fluke was used in the following channel calibration surveillances, calibrating the following instruments:

Surveillance Test	Instruments
DIS 0700-01, “Recirculation Flow Instrument Calibration,” Rev. 26	2-260-26A -total flow unit 2-260-26B-total flow unit
DIS 0263-01, “Rx [Reactor] Vessel Low Water Level Scram and Low Low Water Level Isolation Transmitter Calibration and EQ [equipment qualification] Maintenance Inspection,” Rev. 28.	Level Transmitter-2-263-58A Level Transmitter-2-263-58B
DIS 700-07, “Preventive Maintenance & Calibration of APRM [average power range monitor] and RBM [rod block monitor] Strip Chart Recorders and Isolators,” Rev. 13	IRM [intermediate range monitor] Recorder Isolator 2-0740-2A IRM Recorder Isolator 2-0740-4b

An isolator provides isolation between safety related and non-safety related circuits.

DIS-0700-01 and DIS 0263-01 must be performed while the unit is off line and in shutdown mode. The other surveillance test DIS 700-07 can be performed while at power. DIS-0700-01 and DIS 0263-01 are 24-month TS Surveillance Requirements.

One of the functions for DIS 0700-01 was to calibrate and verify functionality of flow- biased, high neutron flux, reactor protection system instrumentation. Procedure DIS 0700-01 fulfills Surveillance Requirement 3.3.1.1.17.

Some of the functions calibrated in DIS 0263-01 were reactor vessel water level low (Surveillance Requirement 3.3.1.1.17, required in Modes 1 and 2), Reactor Protection System Instrumentation; reactor vessel water level low (Surveillance Requirement 3.3.6.1.6, Modes 1,2, and 3 required), Primary Containment Isolation; reactor vessel water level low low (Surveillance Requirement 3.3.6.1.6, Modes 1,2, and 3 required), and Primary Containment Isolation-Main Steam Isolation.

The inspectors noted that the licensee's IR 259982 stated that the three surveillance tests needed to be performed and verified through work control, and that the re-calibrations were scheduled for March 14, 2005. The inspectors asked the licensee for justification for selecting a March 14, 2005, surveillance test completion date. The licensee re-evaluated the date based on the inspectors' question and on November 11, 2004, initiated IR 270381. The IR stated that, "Given pre-outage, outages, and a tight surveillance schedule in spring for emergency core cooling system fragments, the current schedule date of March 14, 2005, was chosen." The justification did not appear appropriate to the inspectors.

After further discussion with licensee management, the inspectors were informed several weeks prior to the end of the Unit 2 outage that the issue would be corrected prior to start-up by performing DIS 0700-01 to verify the lost fluke's accuracy. One day before scheduled start-up of Unit 2, the licensee began performing DIS 0700-01. However, the licensee stopped the task before completion. Unit 2 was restarted without performing the surveillance test. The surveillance tests were not scheduled to be performed until the next refueling outage or the next forced outage. The next refueling outage would be the next time the surveillance tests are due again in their normal test frequency.

Analysis: The inspectors determined that the failure to re-perform or satisfactory evaluate the surveillance tests following the loss of the M&TE was a performance deficiency warranting a significance evaluation. The inspectors concluded the finding was greater than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," issued on June 20, 2003. The inspectors concluded that the finding was greater than minor because, if left uncorrected, the failure to re-perform or satisfactorily evaluate surveillance testing after M&TE is lost could become a more significant safety concern, if it can not be adequately demonstrated that the equipment will perform within expected parameters. The primary cause of this violation was related to the cross-cutting issue of Problem Identification and Resolution.

The inspectors completed a Phase 1 significance determination of this issue using IMC 0609, "Significance Determination Process," Appendix A, dated September 10, 2004. Based on the surveillance tests that were not re-performed the inspectors concluded that this finding was within the Mitigating Systems Cornerstone. The inspectors entered the table of questions for Mitigating Systems and answered the first question "No." Therefore, the finding screened as having very low safety significance (Green).

Enforcement: 10 CFR 50, Appendix B, Criterion XVI, states, in part, measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and non-conformances are promptly identified and corrected. The licensee's procedure, MA-AA-716-040, stated in Section 4.6.1 that deficient M&TE is M&TE out of tolerance, lost, or damaged. Section 4.6.7 stated, in part, that the questionable time period starts with the last successful calibration, which was September 4, 2003. The Exelon QA Topical Report, EGC-1A, Rev. 70 commits to ANSI/ANS-3.2-1988, Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants. Section 5.2.16 of ANSI/ANS-3.2 states in part, "Special calibration shall be performed when the accuracy of either installed or calibrating equipment is questionable." Contrary to the above, fluke 2679019 was identified lost on October 4, 2004, and special calibration for the surveillance test procedures affected by the questionable instrumentation was not performed. These surveillance test procedures were not scheduled to be performed until the next refueling outage, which would be the next regularly scheduled performance of the surveillance test. The licensee had the opportunity to perform these surveillance tests during an unscheduled outage between November 2 and December 10, 2004, and did not re-perform the tests. As a corrective action after completion of the outage and subsequent to the end of the inspection period the licensee prepared an engineering evaluation that addressed the performance of the three surveillance tests and gave reasonable assurance that the functions used by the M&TE were within calibration when these tests were performed. Further, the inspector identified that portions of other surveillance tests using different, calibrated M&TE, could be combined to show that the installed equipment was satisfactory. The licensee also enhanced the controls for assigning work that needed to be re-performed due to lost M&TE into the work control process. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program (IR 283577), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy. **(NCV 05000237/2004013-06)**

4OA3 Event Follow-up (71153)

(Closed) Licensee Event Report (LER) 50-237;249/2004-003-01: Unit 3 Scram Due to Loss of Offsite Power and Subsequent Inoperability of the Standby Gas Treatment System for Units 2 and 3.

As discussed in Inspection Report No. 50-237;249/2004010, the cause of the Unit 2/3 emergency diesel generator output breaker trip was still under investigation at the time of the original LER submittal. The licensee submitted a supplemental report dated October 29, 2004, which concluded that the apparent cause of the output breaker trip was indeterminate. The electrical relay (HFA 127-XT32X, Drawing 12E-3343) that should have prevented the transformer 32 feed breaker from closing onto bus 33 was sent to an offsite lab for failure analysis. No faults could be found with the relay when it was operated electrically. The lab did identify that one of the relay contacts would hang up when it was actuated manually; however, this is not the method used to actuate the relay when installed in the plant. The licensee concluded that the most probable cause for the Unit 2/3 emergency diesel generator output breaker trip was the effect of the electrical transient experienced during the loss of offsite power on contact fingers within the electrical relay which resulted in an automatic out-of-phase parallelling offsite power with the emergency diesel generator. As corrective action, the licensee revised procedure DGA-12, "Partial or Complete Loss of AC Power," to require that affected transformer feed breaker control switches be placed in the pull-to-lock position during the restoration of normal offsite power sources to preclude automatic operation of the breakers. No findings were identified.

4OA4 Cross-Cutting Findings

.1 A finding described in 1R04 of this report had, as its primary cause, a Human Performance deficiency, in that operators mispositioned a valve in the flow path necessary to pump down the Unit 2 torus to the Unit 2 Hotwell.

.2 A finding described in 1R15 of this report had, as its primary cause, a Human Performance deficiency. The licensee identified in Engineering Evaluation EC34593 that 8 inch diameter 150 lb flanges were installed on the main steam relief valve discharge lines on each unit instead of 300 lb flanges as required. Operability Evaluation 03-013, "Electromatic Relief Valve (ERV) Discharge Piping Flanges," stated that the discharge flanges were operable and no further actions were required.
Per the calculations in Engineering Evaluation 345943, the 8 inch flanges were shown to not meet Section III of the ASME B&PV Code 1977 Edition through Summer 1977 Addenda allowable stresses. Therefore, the 8 inch relief valve discharge line flanges were operable but degraded. In addition, the licensee's calculations did not include all the mechanical loads that would be placed on the flanges and the bolts securing the flanges.

.3 A finding described in 1R20.1.(1) of this report had, as its primary cause, a Human Performance deficiency, in that maintenance personnel removed a breaker that was out-of-service in the racked to test position.

.4 A finding described in 1R20.1.(2) of this report had, as its primary cause, a Human Performance deficiency, in that a contract worker failed to implement station procedures and standards as instructed. The electrical worker failed to implement protected pathway barrier requirement by tossing a piece of metal causing a trip of bus 38.

.5 A finding described in 1R22 of this report had, as its primary cause, a Human Performance deficiency, in that the operating crew performed a reactor scram surveillance test that resulted in unexpected rod motion because they did not understand and therefore did not meet the procedure prerequisite to have all rods inserted or removed prior to the start of the test.

.6 A finding described in 4OA2 of this report had, as its primary cause, a Problem Identification and Resolution deficiency, in that the licensee failed to take prompt and effective actions regarding the validation of surveillance tests performed after it was identified that the maintenance and test equipment used to perform the surveillance tests was identified as lost.

4OA6 <u>Meetings</u>

.1 <u>Interim Exit Meetings</u>

Interim exit meetings were conducted for:

- Occupational Radiation Safety ALARA program inspection during the licensee's Unit 3 refueling outage with Mr. D. Wozniak on November 10, 2004.

- Inservice Inspection (IP 71111.08), with Mr. D. Wozniak on November 19, 2004.

ATTACHMENT: SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

D. Bost, Site Vice President
D. Wozniak, Plant Manager
R. Bauman, ISI Coordinator
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
K. Hall, NDE Level III
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
P. Salas, Regulatory Assurance Manager
N. Spooner, Site Maintenance Rule Coordinator
B. Surges, Operations Requalification Training Supervisor
B. Svaleson, Maintenance Director
S. Taylor, Radiation Protection Manager

NRC

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

IEMA

R. Schulz, Illinois Emergency Management Agency

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000237/2004013-01	NCV	Unit 2 Torus to Hotwell Isolation Valve Mispositioned
05000237/2004013-02 05000249/2004013-02	NCV	Failure to Adequately Perform an Operability Evaluation
05000249/2004013-03	NCV	Electricians Removed the 3d Drywell Cooler Breaker While it Was Tagged Out-of-service in the Racked-to-test Position
05000249/2004013-04	NCV	Failure to Adequately Ensure That a Contract Worker Followed Station Standards While Working in an Area Flagged with a Protected Pathway Sign
05000249/2004013-05	NCV	Unexpected Control Rod Motion During Surveillance Testing
05000237/2004013-06	NCV	Failure to Take Prompt and Effective Actions Regarding the Validation of Surveillance Tests Performed after it Was Identified That the Some of the Maintenance and Test Equipment (M&te) Used to Perform the Tests Was Lost

Closed

50-237;249/2004-003-01	LER	Unit 3 Scram Due to Loss of Offsite Power and Subsequent Inoperability of the Standby Gas Treatment System for Units 2 and 3

Discussed

None

LIST OF ACRONYMS USED

ALARA	As Low As Is Reasonably Achievable
BWR	boiling water reactor
CFR	Code of Federal Regulations
CO	Clearance Order
CR	Condition Report
D3R18	Dresden Station's 18th Unit 3 Refueling Outage
DIS	Dresden Instrument Surveillance
DOS	Dresden Operating Surveillance
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
HPCI	high pressure coolant injection
IEMA	Illinois Emergency Management Agency
IMC	Inspection Manual Chapter
IR	Inspection Report (in Summary of Findings)
IR	Issue Report (in body of report)
LER	Licensee Event Report
MWe	megawatts electrical
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
OA	Other Activities
ODCM	Offsite Dose Calculation Manual
PI	Performance Indicator
RCS	reactor coolant system
RWP	Radiation Work Permit
SDP	Significance Determination Process
TEDE	Total Effective Dose Equivalent
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.

1R01 Adverse Weather

-DOS-10-28, "Preparation for Cold Weather for Radwaste," Revision 13
-DOS-10-19, "Preparation for Cold Weather for Unit 1 and Out Buildings," Revision 17
-DOS-10-31, "Preparation for Cold Weather at the Lift Station, Goose Lake Pump Station, Security Diesel Building, and Cooling Towers," Revision 9
-DOS-10-25, "Preparation for Cold Weather for Unit 3," Revision 11
-DOS-10-22, "Preparation for Cold Weather for Unit 2," Revision 13
-UFSAR, Section 2.3, "Meteorology," Revision 4
-UFSAR, Section 2.4.2, "Effects of Local Intense Precipitation on Roof Structures"
-UFSAR, Section 1.9.2.1, "Category 1, Equipment Modifications or Additions Required by the NRC"
-UFSAR, Section 11.2.2.2.16, "Floor Drain Surge Tank"
-IR 00274284; NRC Identified 2/3 Floor Drain Surge Tank Heaters not On; dated 11/17/2004
-Drawing B-1751, Revision B; Modifications to Existing Parapets
-Drawing 12E-5789, Revision C; Tank Agitator and Surge Tank Heaters, Maximum Recycle Radwaste System, Units 2 and 3
-Out-of-Service 990004464, "Floor Drain Surge Tank," dated 05/06/1999
-Monthly Issues from Dresden Nuclear Power Station Nuclear Oversight, Period Covering - October 1, 2004, through October 31, 2004.
-Dresden Winter Readiness Certification dated November 15, 2004
-WC-AA-107; "Seasonal Readiness;" Revision 0
-DOS 0010-19; "Preparation for Cold Weather for Unit 1 and the Outbuildings," Revision 17
-DOS 0010-22; "Preparation for Cold Weather for Unit 2," Revision 13
-DOS 0010-25; "Preparation for Cold Weather for Unit 3," Revision 11
-DOS 0010-28; "Preparation for Cold Weather for Radwaste." Revision 13
-IR 00225412; Winter Readiness Work Orders not Scheduled Properly; dated 06/02/2004

1R04 Equipment Alignment

-DRE96-181; "Sizing of the Unit 3, 250 VDC Battery Charger 3 and Swing Charger 2/3;" Revision 000A
-WO#99053807-01; D3 20Y Replace 250V Station Main Battery
-WO#99053807-02; Pre-Outage Work: Stage the new Unit 3 250 VDC Station Battery in TB3.
-WO#99053807-03; This work order task performs a Battery Service Test (per DES 8300-15, which includes re-charging) on the new Unit 3, 250V Battery Bank, prior to installation. This pre-installation testing is required to minimize out of service time once installed.
-WO#99053807-04; Measure and record battery connections resistances on the installed 250 VDC battery (which will be replaced) per procedure DES 8300-15
-WO#99053807-05; Make needed repairs to 250VDC Station Battery Rack during the course of cell replacement under task 01
-WO#99053807-06; Clean Battery Racks during replacement of battery cells under task 01
-DES 8300-15; "Unit 3 250 Volt Station Battery Service Test," Revision 12

-DOA 6900-04; “Failure Of Unit 2(3) 250 VDC Power Supply,” Revision 14
-DGA-03; “Loss of 250 VDC Battery Chargers Concurrent with a Design Basis Accident;” Revision 08
-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 3 250 VDC Safety Related Battery
-EC 343968, Revise 250VDC Calculation in Response to Engineering 2003 SSDI FASA Findings
-IR 00266125; Weak FME Practices for D3R18 Staged Equipment; dated 10/22/2004
-IR 00265808; Foam Spacer Needs to be Restored Between Battery & Rack; dated 10/21/2004
-IR 00265809; NRC Identified Concern with DOA 6900-4, Failure of the 250 V; dated 10/21/2004
-IR 00264165; 250 VDC Cubicle Terminal Block End Plate Broken; dated 10/20/2004

1R06 Flooding

-Dresden - UFSAR, Section 3.4.1.2; “Internal Flood Protection Measures;” Revision 5; January 2003
-Dresden Station, Units 2 and 3, Flooding of Critical Equipment - Dresden Station Special Report No. 33; dated August 20, 1973
-DOA 0400-02, ”Localized Flooding in Plant,” Revision 15
-DTP 70, Revision 00; Data Sheet 1, Unit 2 CCSW Pump Vault Flood Protection Barrier Leakage; Dated March 28, 2004
-DTP 70, Revision 00; Data Sheet 2, Unit 2 CCSW Pump Vault Flood Protection Barrier Leakage; Dated September 1, 2004
-DOS 1500-20, “CCSW Pump Vault Penetration Surveillance Testing,” Revision 0
-DIS 4400-01, “Condenser Pit High and High-High Water Level Switch Functional Check,” Revision 13
-DOS 1500-21, “CCSW Pump Vault Watertight Door Leak Test,” Revision 0
-DOS 4400-01, “Containment Cooling Service Water Vault Floor Drain,” Revision 08
-DTP 70, Revision 00; Evaluation of CCSW Pump Vault Flood Protection Leakage Test Results
-TRM Condensate Pump Room Flood Protections 3.7.o; Revision 0
-AR 175749; CCSW Pump Vault Historical Operability; dated 09/15/03
-EC 343339, Revision 3; CCSW Pump Vault Penetration Leakage Criteria

1R07 Heat Sink Performance

-WO#99023075; D2 5Y TS Isolation Condenser Heat Removal Test; September 3, 2003
-IR 286339; Transcription error in DOS 1300-01 5-Year test calculation; dated 12/29/04

1R08 Inservice Inspection

GE-PDI-UT-1; PDI Generic Procedure for the Ultrasonic Examination of Ferritic Pipe Welds; Revision 3; dated September 26, 2003
GE-PDI-UT-2; PDI Generic Procedure for the Ultrasonic Examination of Austenitic Pipe Welds; Revision 3; dated September 26, 2003
ER-AA-335-005; Radiographic Examination; Revision 1; dated October 10, 2002
IR 00243177; Perform NDE and Inspect FW Heater SS OP Vent Connections D3R18; dated 08/10/04

1R11 Operator Requalification

LT017, “Simulator Exercise Guide,” Revision 01, dated September 2004
IR 284315; Crew Failed Out-of-The-Box Evaluation, dated 12/20/04

1R12 Maintenance Effectiveness

Procedures:
-ER-AA-310-1004; Maintenance Rule - Performance Monitoring; Revision 1
-ER-AA-310; Implementation of the Maintenance Rule; Revision 3
-R-AA-310-1006; Maintenance Rule - Expert Panel Roles and Responsibilities; Revision 1

Engineering Evaluations:
-EC 351751; Operating Considerations for Unavailability of CST Suction to HPCI

Issue Reports:
-198748; HPCI Inlet Drain Pot Level High Alarm Received After Scram; dated 2/01/04
-179654; U2 HPCI GSLO Pmp Auto Starting; dated 10/07/2003
-248494; High Tritium Activity in On-Site Wells and Storm Drains; dated 8/3/04
-250360; Documentation of HPCI/LPCI Operability Review; dated 9/6/04
-257124; U2 HPCI Sump Pump Stopped Working; dated 9/27/04
-173114; Potential for Hot Spots in HPCI Room; dated 8/26/03
-264352; Scaffolding That Should Have Been Removed; dated 10/17/04
-270992; NRC Identifies Scaffold Touching Safety Related Equipment; dated 11/6/04

Maint Rule HPCI Data:
-Z23-1; Provide Emergency Pressure Control; Nov 02 to Nov 04
-Z23-3; HPCI Room Cooler; Oct 02 to Oct 04

1R13 Maintenance Risk Assessments and Emergent Work Control

-IR 00264548; 39-7/38-7 Undervoltage Time Delay Relay Found out of Spec.; dated 10/18/04
-OOS# 00025434; TR32 Control power/protective relays
-OOS# 00025438; Fire Proct Reserve Aux XFMR #32 Deluge SV
-OOS# 00025439; TR 32 Cooling Equipment
-OOS# 00025441; U3 Reserve Aux Transformer 32 (RAT)
-DOS 6600-07, “Testing LPCI Swing Bus Protective Relays and Auto Transfer Function,” Revision 18
-DOS 6600-03, “Bus Undervotage and ECCS Integrated Functional Test for Unit 2/3 Diesel Generator to Unit 3,” Revision 20
-HU-AA-1211; Revision 1; Attachment 2; HLA / IPA Briefing Worksheet

1R14 Personnel Performance Related to Non-routine Evolutions and Events

IR 267241; D3R18 Cooldown delay; dated 10/26/04

1R16 Operator Workarounds

Procedures:
-OP-AA-102-103; Operator Work-Around Program; Revision 1
Contingency Plans:
-OP-AA-108-111; HPCI Swap to CST for ATWS or SSD; 10/12/04

-Operations list of Current Operator Workarounds and Operator Challenges as of 11/2/2004
-IR 00252871; U3 Torus Level Decrease; dated 09/14/04

1R19 Post Maintenance Testing

-IR 002755731; Unit 3 EDG Fuel Primimg Pump Failed to Operate; dated 11/22/04
-WO 757836; Unit 3 EDG Fuel Primimg Pump Failed to Operate; November 23, 2004
-IR 00277840; Discovered a Crack Into the Turbocharger Discharge Airduct DG; dated 11/30/04
-IR 00277396; NRC Questions Guidance in DOS 6600-01; dated 11/29/04
-WO#747298-01; OP D3 1M TS Unite Diesel Generator Operability; November 17, 2004
-WO#695912; Replace the Inlet Air Silencer Box Due to Cracking; October 14, 2004
-WO#672873; MM- Replace Pump W/ Flange Ends and Flex Hoses in Lines; November 11, 2004

1R20 Outage Activities

-DGP-02-01, Attachment B; Reactor Cooldown Monitoring Log; Revision 90
-MA-AA-716-008; Foreign Material Exclusion Program; Revision 1
-D3R18 BWR Reactor Vessel Disassembly, Dryer Modification Work, In-Vessel Maintenance and Inspection Activities, and Reactor Re-Assembly FME Plan; Revision B
-TODI 04-012, Required Plant Conditions for Alternate Decay Heat Removal (ADHR) during R3R18
-IR 00269238; NRC Concerns; dated 11/1/04
-IR 00277146; U3 Drywell Closeout Identified Discrepancies; dated 11/29/04
-MA-AA-796-024; Scaffold Installation, Inspection, and Removal; Revision 3
-MA-AA-716-025;Scaffold Installation, Modification, and Removal Request Process; Revision 0
-DOS 0040-10, “Unit 2 Shutdown Power Sources and Distribution,” Revision 9
-DOS 0040-11, “Unit 3 Shutdown Power Sources and Distribution,” Revision 11
-DOS 1000-02, “Alternate Decay Heat Removal Using Shutdown Cooling and Fuel Pool Cooling,” Revision 8
-DOS 1600-10, “Drywell Closeout Inspection Plan,” Revision 31
-ER-AA-600-1043; Shutdown Risk Management; Revision 1
-OU-DR-104; Revision 2; Shutdown Safety Management Program
-OU-DR-104; Revision 1; Shutdown Safety Management Program
-OP-MW-109-101; Revision 2; Clearance Preparation and Approval Checklist
-DOP 1900-03, “Reactor Cavity, Dryer/Separator Storage Pit and Fuel Pool Level Control,” Revision 37
-DOS 10000-02, “Alternate Decay Heat Removal Using Shutdown Cooling and Fuel floor Cooling”
-OU-AA-103; Revision 4; Shutdown Safety Management Program

1R22 Surveillance Testing

-ER-AA-380, Revision 3, Primary Containment Leakage Rate Testing Program
-DTP 47, Revision 13, Leak Rate Testing Program
-IR 00258727; Subject: OE19209 - Appendix J Local Leak Rate Boxes are not; dated 10/4/04

-IR 00261867; Cannot Perform LLRT of Containment Penetration X-317A; dated 10/9/04
-IR 00261989; Penetration 3-1600-X317A seal number 2 leakage >5 SCFH; dated 10/9/04
-Unit 3 Appendix A, Revision 97, Unit NSO Daily Surveillance Log
-IR 00273067; CRD Inserting During DOS 500-07; 11/12/04
-DOP 0300-04, "Control Rod Drive Venting," Revision 14
-DOP 0300-18, "Control Rod Drive Mechanism Removal/Replacement," Revision 16
-DOP 0300-08, "Control Rod Drive System Hydraulic Control Unit Isolation/Pump Isolation," Revision 18
-DOS 0500-07, "Reactor Mode Switch in Shutdown Functional and Scram Auxiliary Functions Valve Operability Test," Revision 21
-WO#497698; D3 30M/RFL TS VVRT DBL Gasket Seal #1 X-317A, HPCI; dated 10/6/04
-WO#497699; D3 30M/RFL TS VVRT DBL Gasket Seal #2 X-317A, HPCI; dated 10/6/04
-DOS 7000-04, "Local Leak Rate Testing of Double Gasket Seals," Revision 01
-DOP 2000-24, "Drywell Sump Operation," Revision 13
-Unit 2(3), Appendix A; Revision 97; Unit NSO Daily Surveillance Log
-WO#693356; D2 QTR/CSD IST HPCI TEST INJ CHK VLV OPER TEST SURV; dated 9/23/04
-IR 00254769; Valve won't cycle during DOS 2300-04; dated 9/20/04
-Performance Trend Data for 2-2301-7; Last test date: 05/02/04

1R23 Temporary Plant Modifications

-TCCP 349380; Revision 0; Install Bleeder Valve on the U2 'A' Condenser Hood Low Vacuum Sensing Line to Prevent Moisture Buildup
-Engineering Change 349380-000; Install Bleeder Valve on the U2 'A' Condenser Hood Low Vacuum Sensing Line to Prevent Moisture Buildup; May 25, 2004
-LS-AA-104-1001; "Install Bleeder Valve on the U2 'A' Condenser Hood Low Vacuum Sensing Line to Prevent Moisture Buildup," Revision 1

2OS1 Access Control to Radiologically Significant Areas

-RWP 10004203; D3R18 Drywell Inspections; Revision 0
-RWP 10004181; D3R18 Observations/Tours and Inspections; Revision 0
-RWP 10004165; D3R18 Reactor Steam Dryer Modification Diving Activities; Revision 2

2OS2 ALARA Planning and Controls

-RP-AA-401; Operational ALARA Planning and Controls; Revision 4
-RP-AA-400; ALARA Program; Revision 3
-Historical Outage Dose Information for Units 2 and 3
-D3R18 RWP Preparation Matrix, Dose Estimates and Associated Time/Labor Estimates and Project View Work Planning information
-D3R18 Daily Dose Reports and Graphs for November 4 - 10, 2004
-RWP 10004147 (Revision 1); Associated TEDE ALARA Evaluations and Recognized Risk Personnel Contamination Dose Assessment; D3R18 RWCU System Maintenance
-RWP 10004164 (Revision 1); Associated ALARA Plan, TEDE ALARA Evaluations and Recognized Risk Personnel Contamination Dose Assessment; D3R18 Reactor Disassembly/Reassembly and Related Activities

-RWP 10004168 (Revision 0); Associated ALARA Plan, TEDE ALARA Evaluations and Recognized Risk Personnel Contamination Dose Assessment; D3R18 Dryer Modification - Support Crew Activities
-RWP 10004191 (Revision 1); Associated ALARA Plan, TEDE ALARA Evaluations and Recognized Risk Personnel Contamination Dose Assessment; D3R18 Nuclear Instrumentation System Maintenance Activities
-RWP 10004194 (Revision 1); Associated ALARA Plan and TEDE ALARA Evaluations; D3R18 Drywell Main Steam Isolation Valve Maintenance
-RWP 10004196 (Revision 1); Associated ALARA Plan and TEDE ALARA Evaluations; D3R18 Drywell Main Steam Safety, Electromatic and Target Rock Valve Maintenance
-RWP 10004206 (Revision 1); Associated ALARA Plan, TEDE ALARA Evaluations and Recognized Risk Personnel Contamination Dose Assessment; D3R18 Drywell In-Service Inspection Activities
-RWP 10004208 (Revision 2); Associated ALARA Plan, TEDE ALARA Evaluations and Recognized Risk Personnel Contamination Dose Assessment; Drywell “B” Recirc Pump and Motor Maintenance Activities
-RWP 10004209 (Revision 1); Associated ALARA Plan and TEDE ALARA Evaluations; D3R18 Drywell Insulation Maintenance Activities
-RWP 10004165 (Revision 2); Associated ALARA Plan; D3R18 Reactor Steam Dryer Modification Diving Activities
-RP-AA-401, Attachment 7; Work-In-Progress Review for RWP 10004165, dated November 8, 2004
-RP-AA-401, Attachment 7; Work-In-Progress Review for RWP 10004186, dated November 6, 2004
-RP-AA-401, Attachment 7; Work-In-Progress Review for RWP 10004201, dated November 2, 2004
-RP-AA-401, Attachment 7; Work-In-Progress Review for RWP 10004209, dated November 8, 2004
-RP-AA-401, Attachment 7; Work-In-Progress Review for RWP 10004188, dated November 1, 2004
-RP-AA-401, Attachment 7; Work-In-Progress Review for RWP 10004165, dated November 2, 2004
-RP-AA-401, Attachment 7; Work-In-Progress Review for RWP 10004164, dated November 4, 2004
-RP-AA-401, Attachment 7; Work-In-Progress Review for RWP 10004659, dated November 5, 2004
-Dresden Station Units 2/3 Boiling Water Reactor Assessment and Control Historical Survey Data
-Dresden Station 2004 - 2007 Source Term Reduction Plan, Listing of D3R18 Temporary Shielding Packages and Hydrolazing Schedule
-Focus Area Self-Assessment Report; Outage Readiness and Preparation, dated October 13, 2004
-Nuclear Oversight “Rad Practices” Field Observation Summary Matrix for November 1-5, 2004
-D3R17 Post Outage Radiation Protection Department Report 2002
-IR 00270562; Bubble Suits Failed to Maintain Integrity; dated 11/05/04
-IR 00265362; Venture Exceeds Dose Goal Estimate on Scaffold; dated 10/20/04
-IR 00269685; SAIC Dosimeter Inadvertently Zeroed for Steam Dryer Diver; dated 11/2/04

2PS3 Radiological Environmental Monitoring Program

-Water Sampling Results from Various Monitoring Wells, the Waste Water Treatment Facility, Unit 1 Intake Canal, Storm Sewers and Excavated Areas for Various Periods between June and October 2004
-Estimates of Offsite Doses from Difficult to Measure Nuclides and ODCM Calculation Worksheets; dated October 19, 2004

4OA3 Event Follow-up

-LER 2004-003-001, "Unit 3 Scram due to Loss of Offsite Power and Subsequent Inoperability of the Standby Gas Treatment System for Units 2 and 3," dated October 29, 2004
-AR 00219063, Assignment 17, Perform EACE on TR32 UV Relay Autopsy
-Procedure DGA-12, Revision 49, "Partial or Complete Loss of AC Power"
-Drawing 12E-3343, Revision Y, "Schematic Diagram 4160V Bus 34 Main & Reserve Feed GCB's"
-Exelon PowerLabs, "Failure Analysis of an HFA Relay from Dresden Station Unit 3, Manufacturer: General Electric Model No. 12HFA51A42F," dated June 10, 2004