July 17, 2003

Mr. John L. Skolds, President
Exelon Nuclear
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: DRESDEN NUCLEAR POWER STATION
NRC INSPECTION REPORT 50-237/03-06; 50-249/03-06

Dear Mr. Skolds:

On June 30, 2003, 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. R. Hovey and other members of your staff on June 24, 2003.

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, the inspectors identified four issues of very low safety significance (Green). Three of these issues were determined to involve violations 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. Additionally, three licensee-identified violations which were determined to be of very low safety significance are listed in Section 4OA7 of this report. If you contest these Non-Cited Violations, you should provide a response within 30 days of the date of this inspection report with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator, Region III; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspectors at the Dresden Nuclear Power Station.

Since the terrorist attacks on September 11, 2001, NRC has issued five Orders and several threat advisories to licensees of commercial power reactors to strengthen licensee capabilities, improve security force readiness, and enhance controls over access authorization. In addition to applicable baseline inspections, the NRC issued Temporary Instruction 2515/148, "Inspection of Nuclear Reactor Safeguards Interim Compensatory Measures," and its subsequent revision, to audit and inspect licensee implementation of the interim compensatory measures required by order. Phase 1 of TI 2515/148 was completed at all commercial nuclear power plants during calender year 2002 and the remaining inspection activities for Dresden are scheduled for completion in 2003. The NRC will continue to monitor overall safeguards and security controls at Dresden.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Mark Ring, Chief
Branch 1
Division of Reactor Projects

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

Enclosure: Inspection Report 50-237/03-06; 50-249/03-06

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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:	50-237/03-06(DRP); 50-249/03-06(DRP)
Licensee:	Exelon Generation Company
Facility:	Dresden Nuclear Power Station, Units 2 and 3
Location:	6500 North Dresden Road Morris, IL 60450
Dates:	April 1 through June 30, 2003
Inspectors:	D. Smith, Senior Resident Inspector B. Dickson, Resident Inspector R. Lerch, Project Engineer P. Pelke, Reactor Engineer M. Sheikh, Reactor Engineer J. Belanger, Senior Physical Security Inspector T. Madeda, Physical Security Inspector J. Maynen, Physical Security Inspector D. Nelson, Radiation Specialist W. Slawinski, Senior Radiation Specialist C. Phillips, Senior Operations Engineer R. Schulz, Illinois Department of Nuclear Safety
Approved by:	Mark Ring, Chief Branch 1 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000237-03-06, IR 05000249-03-06, 04/01/2003 - 06/30/2003, Exelon Generation Company, Dresden Nuclear Power Station, Units 2 and 3; Personnel Performance Related to Non-routine Evolutions and Events, Post Maintenance Testing, Identification and Resolution of Problems, and Event Follow-up.

This report covers a 3-month period of baseline resident inspection, and announced baseline inspections on radiation safety and safeguards. The inspection was conducted by Region III specialist inspectors and resident inspectors. Four Green findings, three of which involved Non-Cited Violations (NCVs), 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 a 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. The inspectors identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion V, for the operators failure to follow a procedure, in monitoring reactor pressure, during power ascension activities. This failure resulted in the licensee inadvertently operating with reactor steam dome pressure above the Technical Specification limit of 1005 psig for more than 2 hours.

The finding was considered more than minor because the issue affected the initiating events objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during power operations. However, the finding was determined to be of low safety significance because the highest reactor steam dome pressure recorded during this event remained within the design basis accident limits. (Section 1R14)

Green. A self-revealed finding was identified for a performance issue associated with the licensee's failure to implement effective corrective actions to resolve cracks on the main steam isolation valve (MSIV) accumulator nozzle welds.

The finding was considered more than minor because the issue affected the initiating events objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during power operations. Also, the finding affected the mitigating systems objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. However, the issue was determined to be of low safety significance because the 1A MSIV was capable of performing its intended safety function of fast closing in 3 to 5 seconds, and the MSIV remained open at all times. (Section 4OA2)

Cornerstone: Mitigating Systems

Green. A self-revealed Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion V, was identified for the failure of two electricians to install a jumper and perform concurrent verification in accordance with a work order on June 13, 2003. This rendered the Unit 2 “D” electromatic relief valve (ERV) inoperable.

This finding was more than minor because it affected the mitigating systems cornerstone objective. However, the finding was of very low significance because there was no loss of safety function in that four of the five ERVs remained operable. (Section 1R19)

Green. The inspectors identified a Non-Cited Violation of the Unit 2 and 3 operating licenses for 34 mechanical penetration seals not containing the required minimum of 8" of ceramic fire blanket to establish a 3-hour rated fire barrier.

The finding was more than minor because it affected the mitigating systems cornerstone objective. However, the finding was of low safety significance because for 33 of the 34 seals, no credible fire scenarios could be developed due to physical configuration of post-fire safe shutdown equipment on either side of the penetration seals or the deficient penetration seals were not used to protect safe shutdown capability. For the remaining penetration, the inspectors determined that the recovery actions in the isolation condenser room could be successfully implemented and ensure safe shutdown of the plant. (Section 4OA3)

B. Licensee Identified Findings

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

Report Details

Summary of Plant Status

Unit 2 began the inspection period at 912 MWe (95 percent thermal and 100 percent of rated electrical capacity). On April 27, 2003, load was reduced to 850 MWe to perform control rod drive exercises, to reverse circulating water flow through the condenser, and to perform turbine control valve testing. The unit was returned to full power the same day. On May 10, 2003, load was reduced to 225 MWe to perform main condenser venting and a deep/shallow rod swap. The unit returned to full power on May 13, 2003. On May 18, 2003, load was reduced to 780 MWe to swap from the A train steam jet air ejector to the B train. The unit was returned to full power the same day. On May 23, 2003, load was reduced to 160 MWe and the turbine was taken offline to perform inspections and repairs to the condenser waterboxes. The unit was returned to full power on May 27, 2003. Load was reduced to 667 MWe on June 1, 2003, after a seal cooler leak developed on the 2C reactor feed pump. The unit was returned to full power on June 3, 2003. Later that day, load was reduced to 680 MWe to address the degradation of the 2A condensate/condensate booster pump thrust bearing. The unit was returned to full power on June 5, 2003. On June 29, 2003, load was reduced to 850 MWe to exercise the turbine control valves. The unit was returned to full power the same day.

Unit 3 began the inspection period at 63 percent thermal power. The unit was ascending in power from a downpower to address drywell inleakage from the 1A main steam isolation valve accumulator and reached full power on April 3, 2003. On May 3, 2003, load was reduced to 550 MWe to perform a deep shallow rod swap and control rod scram time testing. The unit was returned to full power on May 5, 2003. On June 1, 2003, load was reduced to 880 MWe to reverse circulating water flow through the condenser. The unit was returned to full power the same day. On June 11, 2003, the unit was shut down for a forced outage to address hydrogen leakage into the generator stator water cooling system. Other outage activities included cold shutdown testing and steam leak repairs. The unit went critical on June 16, 2003, and the generator was synchronized to the grid on June 18, 2003. The unit was returned to full power on June 20, 2003. On June 28, 2003, load was reduced to 550 MWe to perform control rod adjustments. The unit was returned to full power on June 30, 2003.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather (71111.01)

a. Inspection Scope

The inspectors assessed the licensee’s implementation of the station’s summer readiness process including the high pressure coolant injection system, and a review of tornado/severe wind and securing from cold weather operations procedures.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignments (71111.04)

.1 Partial System Walkdowns

a. Inspection Scope

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

The inspectors performed equipment alignment walkdowns of the following systems:

- Unit 2/3 emergency diesel generator; and
- 2B core spray.

b. Findings

No findings of significance were identified.

.2 Complete System Walkdown

a. Inspection Scope

The inspectors performed a complete walkdown of Unit 2 250 VDC battery system. The inspectors reviewed the electrical system checklist and electrical drawings for this system to ensure all vital components in this system were energized. The inspectors reviewed outstanding work orders associated with the system to determine whether there were any deficiencies that could affect the ability of the system to perform its safety-related function. The inspectors also reviewed all temporary modifications and operator workarounds to verify the operational impact on the system. The inspectors reviewed licensee condition reports (CRs), to verify past issues that had been identified and their corrective actions.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

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

- Unit 2 low pressure coolant injection southwest corner room, Fire Zone 11.2.1;
- Unit 3 low pressure coolant injection southeast corner room, Fire Zone 11.1.2;
- Unit 2 reactor water clean up system area - Fire Zone 1.1.2.3;
- Unit 2 feedwater regulator valve mezzanine, Fire Zone 3.A.1.1;
- Unit 3 emergency diesel generator room - Fire Zone 9.0.B;
- Unit 2 545' reactor building, Fire Zone 1.1.2.3; and
- Unit 2 534' turbine building 8.2.6.A.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11)

a. Inspection Scope

The inspectors observed crew #5 on June 2, 2003, in simulator training. The scenario consisted of an isolation condenser tube leak, 2B control rod drive pump failure, 2A control rod drive pump trip, scram discharge volume partial hydraulic lock (anticipated transient without scram), and 2B standby liquid control pump failure.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

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

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

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

The inspectors reviewed the following systems:

- Unit 3 reactor feedwater system; and
- Unit 2 and 3 isolation condenser system.

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 completed evaluations of maintenance activities on the:

- Unit 2 high pressure coolant injection exhaust drain pot piping due to plugging;
- 2B electrohydraulic control maintenance and inspection;
- 2B standby liquid control system pump; and
- Unit 3 reactor recirculation control system/motor generator set.

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.

- Technical Specification 3.4.10.1 High Pressure Condition on Unit 2 Following Power Manipulation.

b. Findings

Unit 2 Reactor Steam Dome Pressure Exceeded Technical Specification Limit

Introduction. A finding for failure to follow a procedure was identified for the operators inadvertently exceeding the Technical Specification required reactor steam dome pressure of 1005 on May 12, 2003.

Description. On May 12, 2003, during Unit 2 power ascension activities, operators inadvertently exceeded the Technical Specifications reactor steam dome pressure limit of 1005 psig. A review of process computer data indicated that reactor pressure reached and stabilized at 1008 psig. This historical data also indicated that this condition had been in existence for more than 2 hours. Technical Specification 3.4.10.A.1 required that if reactor pressure of 1005 psig is exceeded reactor pressure shall be restored to less than 1005 psig within 15 minutes. Upon recognition of this abnormal plant condition, the operators immediately reduced reactor pressure to below 1005 psig. The licensee's investigation determined that poor control panel monitoring by the operators during power ascension activities was the cause of this event.

Prior to this event, on May 10, 2003, operators had reduced power to approximately 225 MWe to perform the Unit 2 main condenser venting and deep/shallow rod swaps. During load recovery on May 12, 2003, the operators ramped up reactor power using recirculation pumps. This power ramp occurred through the reactor recirculation pump speed "exclusion zone" region where the licensee has determined that extended plant operation could potentially cause failure of the jet pump support riser leaves due to the effect of excessive harmonic vibrations. The licensee established administrative requirements for how long the plant should be operated in the "exclusion zone." Dresden General Operating Procedure (DGP) 03-01, "Routine Power Changes," Revision 59, limited the amount of time the Unit 2 recirculation pumps are between 67 percent and 82 percent pump speed (~ 935 rpm).

During power ascension through the "exclusion zone," poor communication between the operators in the main control room and the non-licensed operators stationed at the recirculation pump motor generator set resulted in overshooting the preestablished reactor recirculation pump speed of 944 rpm. Pump speed reached 980 rpm and operators halted power ascension. The station qualified nuclear engineer determined that no adverse effects occurred regarding thermal limits. However, the operators did not check or record reactor pressure at that time.

Section G.5 of DGP 03-01 required that the operators document the initial and final values for specific reactor parameters when performing power changes greater than 25 MWe. During this evolution reactor power was ramped from 714 MWe to 844 MWe. Reactor pressure was one of the parameters required to be documented. A review of historical data on the narrow range pressure recorder and the process computer indicated that 1005 psig was exceeded at 1307 hours, which was also about the time that operators recognized that the preestablished recirculation pump speed had been exceeded. In addition, this time period overlapped two operating crew shifts. At 1509, the shift manager on the oncoming crew identified Unit 2 reactor pressure was at 1008 psig. After lowering pressure below 1005 psig, the licensee started a prompt investigation, documented this issue in condition report 158419, and performed a root cause investigation which concluded this event was caused by the failure of the operators to accurately predict and monitor reactor pressure during and following power ascension. Also, the licensee determined that multiple administratively required main control board walkdowns were ineffective in identifying this issue. The walkdowns included 15 minute walkdowns by the reactor operators, more detailed hourly walkdowns by the reactor operators, and walkdowns of the main control room board by the oncoming and off-going operators during shift turnover. These oncoming and off-going walkdowns were performed by the primary reactor operators, the auxiliary operators, the unit supervisors and the shift managers.

Analysis. The inspectors used Inspection Manual Chapter (IMC) 0612, “Power Reactor Inspection Reports,” Appendix B, “Issue Disposition Screening,” to disposition this issue and determined that it was more than minor because the issue was associated with the reactor safety crosscutting attribute of human performance and affected the initiating event objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during power operations. Additionally, the inspectors evaluated this issue using Inspection Manual Chapter 0609, “Significance Determination Process” (SDP). The inspectors conducted a Phase 1 screening and determined that this issue was of low safety significance, because the steam dome pressure during the time the reactor was above technical specification limits was within the design basis accident limits.

Enforcement. 10 CFR 50, Appendix B, Criterion V, requires that activities affecting quality be accomplished in accordance with procedures. On May 12, 2003, operators failed to note and record reactor pressure as required in Section G.5 of DGP 03-01. Failure of the operators to follow this procedure was considered a violation. However, because of its low safety significance and because it was entered into the corrective action program as CR 001548419, the NRC is treating this issue as a Non-Cited Violation (**NCV 50-237/03-006-01**), consistent with Section VI.A.1 of the NRC’s Enforcement Policy.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

Throughout the inspection period, the inspectors reviewed operability evaluations (OE) to ensure that operability was properly justified and that the affected component or system remained available, such that no unrecognized increase in risk occurred. The inspectors used the Dresden Updated Final Safety Analysis Report (UFSAR) in assessing the following issues involving system operability:

- Units 2 and 3 electromatic relief valve/target rock valve pressure switches 2(3)-0203-3A, B, C, D, and E setpoint drift (OE #03-006);
- Units 2 and 3 existing hydraulic flow calculation for the isolation condenser make-up pumps does not support the Extended Power Uprate, Appendix R analysis (OE #03-007);
- Engineering Change (EC) #342944, “Containment Cooling Service Water System Water Hammer Loss of Keep-fill Analysis;” and
- CR157770; Failure analysis report D2 high pressure coolant injection motor gear unit removed motor.

b. Findings

No findings of significance were identified.

1R16 Operator Work-Arounds (71111.16)

a. Inspection Scope

During this period the inspectors reviewed cumulative operator work-arounds and Engineering Change (EC) #347105 “Electrohydraulic Control Scram Pressure Switch Suspect to Half Scram,” to assess any potential effect on the functionality of mitigating systems. During this review, the inspectors determined if the operators’ ability to implement abnormal or emergency operating procedures was impacted.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed post-maintenance test results to confirm that the tests were adequate for the scope of the maintenance completed and that the test data met the acceptance criteria. The inspectors also reviewed the tests to determine if the systems were restored to the operational readiness status consistent with the design and licensing basis documents. The inspectors reviewed post-maintenance testing activities involving risk significant equipment in the mitigating systems cornerstone:

- Maintenance of Unit 2 shutdown cooling heat exchanger 2C outlet motor operated valve;
- Replacement of 15 V regulator in average power range monitor 4;
- High pressure coolant injection system post maintenance test performed per DOS 2300-3; and
- Replacement of 3D containment cooling service water discharge check valve.

b. Findings

Introduction. A self-revealed finding for failure to follow a work order was identified when electricians’ actions resulted in rendering a valve inoperable.

Description. On June 13, 2003, the Unit 2 control room received alarm ‘ADS Main DC Pwr Failure’ (902-3, C-15) and the “D” electromatic relief valve (ERV) light indication went out. Work Order 99022222-02 was being performed by two electricians to jumper/lift leads to defeat the valve logic on Valve 2-1001-4C, Unit 2 shutdown cooling heat exchanger 2C outlet motor operated valve, to support post maintenance testing. Step F.4 of the work order required the electricians to install a jumper between terminal strips NN96 and NN99 at the 902-4 panel in the control room. Step D.1 of the work order required the use of concurrent verification in accordance with administrative procedure HU-AA-101, “Human Performance Tools and Verification Practices,” Revision 1. However, the electricians installed the jumper between terminal strips NN96 and NN99 in the wrong panel, Panel 902-3, instead of Panel 902-4. This caused a short in the “D” ERV controller circuit resulting in two blown power fuses and damage to the power transfer relay. The “D” ERV was declared inoperable.

Analysis. The event was more than minor because it affected the mitigating systems cornerstone objectives. However, the event was Green because there was no loss of safety function in that four of the five ERVs remained operable.

Enforcement. 10 CFR 50, Appendix B, Criterion V, requires that activities affecting quality be accomplished in accordance with procedures. Step F.4 of Work Order 99022222-02 required the electricians to install a jumper between terminal strips NN96 and NN99 at the 902-4 panel in the control room. Step D.1 of the work order required the use of concurrent verification in accordance with HU-AA-101. Contrary to the above, the electricians did not adequately implement concurrent verification and installed the jumper between terminal strips NN96 and NN99 in panel in Panel 902-3.

Because rendering the “D” ERV inoperable was of very low safety significance and the issue has been entered into the licensee’s corrective action program as Condition Report #163084, this violation is being treated as a Non-Cited Violation (**NCV 50-237/03-006-02**), consistent with Section VI.A of the NRC Enforcement Policy.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors reviewed and evaluated several outage activities during a Unit 3 forced outage. The evaluation was performed to ensure that the licensee appropriately considered risk factors during the development and execution of planned activities. The inspectors conducted walkdowns of systems vital to maintaining the unit in a safe/shutdown condition. The inspectors also ensured that Technical Specification requirements were verified to have been met for changing modes.

b. Findings

No findings of significance were identified.

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 Technical Specifications. Following the completion of the 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 review included surveillance testing activities associated with the following:

- DIS 0500-01, Reactor Vessel Hi Pressure Scram Pressure Switch Calibration;

- DIS 0500-06, Revision 18, Condenser Low Vacuum Pressure Switches Channel Calibration and Channel Functional Test;
- DOS 7501-01, Standby Gas Treatment Monthly Operability Surveillance;
- DIS 1300-01, Revision 18, Sustained High Reactor Pressure Calibration;
- Unit 2 and 3 Automatic Depressurization System;
- Unit 2 Standby Liquid Control Tank Heater Surveillance Test and Standby Liquid Control System Pump Test for the Inservice Testing Program; and
- Unit 2 Emergency Diesel Generator Monthly Operability and Semiannual Fast Start Operability Surveillance.

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 Radiological Boundary Verification

a. Inspection Scope

The inspectors conducted walkdowns of the radiologically protected area to verify the adequacy of radiological area boundaries and postings. Specifically, the inspectors walked down numerous radiologically significant area boundaries (high and locked high radiation areas) in the Unit 2 and Unit 3 Reactor and Turbine Buildings and the Radwaste Building to determine if these areas and selected radiation areas were properly posted and controlled in accordance with 10 CFR Part 20, the licensee's procedures and Technical Specifications. The inspectors challenged access control boundaries to verify that locked high radiation area (LHRA) access was properly controlled and independent radiation surveys were performed to verify boundary radiological conditions. The inspectors also observed the radiological conditions of work areas within those radiation and high radiation areas walked down to assess the radiological housekeeping and contamination controls.

b. Findings

No findings of significance were identified.

.2 High Risk Significant, High Radiation Area, and Very High Radiation Area Access Controls

a. Inspection Scope

The inspectors reviewed the licensee's procedures, radiation protection (RP) job standards and evaluated RP practices for the control of access to radiologically significant areas (high, locked high, and very high radiation areas) and assessed compliance with the licensee's Technical Specifications, procedures and the requirements of 10 CFR Part 20. In particular, the inspectors evaluated the licensee's control of keys to LHRAs and very high radiation areas (VHRAs), the use of access control guards to control entry into such areas, and methods and practices for independently verifying proper closure and locking of access doors upon area egress. The inspectors reviewed LHRA/VHRA key issuance/return and door lock verification logs for 2003 through June, and key inventory records for May and June 2003, to verify the adequacy of accountability practices and documentation. The inspectors also reviewed records and evaluated the licensee's practices for RP and station management approval of access to Level 2 LHRAs and VHRAs to verify compliance with procedure requirements and those of 10 CFR 20.1602. Additionally, the inspectors evaluated the licensee's methods for the movement of highly radioactive material within the plant to ensure temporary high radiation areas were properly controlled and posted.

b. Findings

No findings of significance were identified.

.3 Review of Radiologically Significant Work Practices

a. Inspection Scope

The inspectors reviewed the licensee's procedures, RP job standards and RP practices for at power and initial entries into the drywell, and for traversing in-core probe (TIP) room access to determine the adequacy of the radiological controls and hazards assessment associated with such entries. Work instructions provided in radiation work permits and in high level activity briefings/worksheets used for drywell entries were discussed with RP staff to determine their adequacy relative to industry practices and NRC Information Notices. The inspectors also reviewed the licensee's procedure and practices for dosimetry placement, use of multiple dosimetry and for extremity monitoring for work in high radiation areas having significant dose gradients for compliance with the requirements of 10 CFR 20.1201(c) and applicable Regulatory Guides.

b. Findings

No findings of significance were identified.

.4 Control of Non-Fuel Materials Stored in the Spent Fuel Pools

a. Inspection Scope

The inspectors reviewed the licensee's programmatic controls and practices for the underwater storage of highly activated or contaminated materials (non-fuel) in the spent fuel or other storage pools. Radiation protection and fuel handling procedures were reviewed, involved staff were interviewed, the most recent inventory record for the spent fuel pools was reviewed and a walkdown of the refuel floor was conducted. In particular, the radiological controls for non-fuel materials stored in the Unit 2 and Unit 3 spent fuel pools were examined to ensure adequate radiological boundaries were in-place to reduce the potential for the inadvertent movement of material stored in the pools. Overall, the inspectors assessed the adequacy of the administrative and physical controls for underwater storage of non-fuel materials for consistency with the licensee's procedures and with Regulatory Guide 8.38, Information Notice 90-33 and applicable Health Physics Positions in NUREG/CR-5569. Additionally, the inspectors discussed with RP management its plans to place a locking device on all those materials stored in the pools with radiation levels in excess of 1000 mrem/hour and to consolidate its fuel handling procedures governing short and long term storage of such materials.

b. Findings

No findings of significance were identified.

.5 Internal Dose Assessments

a. Inspection Scope

The inspectors reviewed the licensee's procedure and evaluated its methods for the assessment of internal dose as required by 10 CFR 20.1204. Specifically, the inspectors reviewed dose assessments performed for several actual intakes that occurred during the licensee's fall 2002 refueling outage, to ensure the doses were calculated correctly and included the impact of hard to detect radionuclides such as pure beta and alpha emitters, as applicable. The inspectors also discussed with RP staff its monitoring of reactor coolant chemistry parameters, operational conditions and 10 CFR Part 61 scaling factors, to determine if an adequate program was developed to identify changes that could alter the plant's radionuclide mix and affect dose assessments.

b. Findings

No findings of significance were identified.

.6 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed an RP self-assessment, Nuclear Oversight quarterly audit reports, the condition report (CR) database and a variety of individual CRs related to radiation worker performance in radiologically significant areas and radiological access controls generated between November 2002 and June 2003, including a common cause analysis of radiological boundary control issues. The inspectors evaluated the effectiveness of the self-assessment process to identify, characterize, and prioritize individual problems and repetitive issues and trends, and to implement corrective actions to achieve lasting results.

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation and Protective Equipment (71121.03)

.1 Rescue Capabilities During Use of One-Piece Atmosphere Supplying Respiratory Protection Devices

a. Inspection Scope

The inspectors reviewed the licensee's respiratory protection and confined space entry procedures and discussed their implementation relative to the requirements of 10 CFR 20.1703(f) for standby rescue persons whenever one-piece atmosphere supplying suits, or any combination of respiratory protection and personnel protective equipment were used which the wearer may have difficulty extricating himself. Specifically, the inspectors reviewed the licensee's work planning process and implementing practices, and interviewed RP staff and a member of the licensee's confined space rescue team regarding the following aspects of 10 CFR 20.1703:

- designation of an adequate number of standby rescue workers and their training/instruction
- presence of equipment staged at the work site for the safety of the rescuer and for extrication of the respiratory equipment user
- practices for continuous communication between standby rescuer(s) and the respiratory protection user(s)
- provisions for immediate availability of the standby rescuer

The inspectors discussed with RP management its proposal for enhancing the radiation work permit and as-low-as-is-reasonably-achievable (ALARA) planning process and for developing safety plans for those jobs not performed in confined space atmospheres to formally address work provisions for standby rescuers.

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety

2PS2 Radioactive Material Processing and Transportation (71122.02)

.1 Radioactive Waste System

a. Inspection Scope

The inspectors reviewed the liquid and solid radioactive waste system description in the Updated Final Safety Analysis Report (UFSAR) for information on the types and amounts of radioactive waste (radwaste) generated and disposed. The inspectors reviewed the scope of the licensee's audit program with regard to radioactive material processing and transportation programs to verify that it met the requirements of 10 CFR 20.1101(c).

b. Findings

No findings of significance were identified.

.2 Walkdowns of Radioactive Waste Systems

a. Inspection Scope

The inspectors performed walkdowns of the liquid and solid radwaste processing systems located in the Radwaste Building to verify that the systems agreed with the descriptions in the FSAR and the Process Control Program, and to assess the material condition and operability of the systems. The inspectors reviewed the current processes for transferring waste resin into shipping containers to determine if appropriate waste stream mixing and/or sampling procedures were utilized. The inspector also reviewed the methodologies for waste concentration averaging to determine if representative samples of the waste product were provided for the purposes of waste classification in 10 CFR 61.55. During this inspection, the licensee was not conducting waste processing.

b. Findings

No findings of significance were identified.

.3 Waste Characterization and Classification

a. Inspection Scope

The inspectors reviewed the licensee's radiochemical sample analysis results for each of the licensee's waste streams, including dry active waste (DAW), evaporator bottoms, spent resins, and filters. The inspectors also reviewed the licensee's use of scaling factors to quantify difficult-to-measure radionuclides (e.g., pure alpha or beta emitting radionuclides). The reviews were conducted to verify that the licensee's program assured compliance with 10 CFR 61.55 and 10 CFR 61.56, as required by Appendix G of 10 CFR Part 20. The inspectors also reviewed the licensee's waste characterization and classification program to ensure that the waste stream composition data accounted for changing operational parameters and thus remained valid between the annual sample analysis updates.

b. Findings

No findings of significance were identified.

.4 Shipment Preparation

a. Inspection Scope

The inspectors observed aspects of the preparation of a shipment of radioactive dry active waste. The inspectors observed the surveying of the packaging loaded on the conveyance and the placarding and visual checks of the conveyance. The inspectors also observed the radiation worker practices of the workers performing the tasks to verify that the workers had adequate skills to accomplish each task. The inspectors reviewed the records of training provided to personnel responsible for the conduct of radioactive waste processing and radioactive shipment preparation activities. The review was conducted to verify that the licensee's training program provided training consistent with NRC and Department of Transportation (DOT) requirements.

b. Findings

No findings of significance were identified.

.5 Shipping Records

a. Inspection Scope

The inspectors reviewed five non-excepted package shipment manifests/documents completed in 2003 to verify compliance with NRC and DOT requirements (i.e., 10 CFR Parts 20 and 71 and 49 CFR Parts 172 and 173).

b. Findings

No findings of significance were identified.

.6 Identification and Resolution of Problems

a. Inspection Scope

The inspector reviewed the reports of focused self-assessments performed during 2002 on radwaste material condition/equipment reliability and during 2003 on the radioactive material processing and transportation program to evaluate the effectiveness of the self-assessment process to identify, characterize, and prioritize problems. The inspectors selectively reviewed 2002 and 2003 condition reports that addressed radioactive waste and radioactive materials shipping program deficiencies, to verify that the licensee had effectively implemented the corrective action program.

b. Findings

No findings of significance were identified.

3. SAFEGUARDS

Cornerstone: Physical Protection

3PP2 Access Control (Identification, Authorization and Search of Personnel, Packages, and Vehicles) (IP71130.02)

a. Inspection Scope

The inspectors reviewed the licensee's protected area access control testing and maintenance procedures. The inspectors observed licensee testing of all protected area access control equipment to determine if testing and maintenance practices were performance based. On two occasions, the inspectors observed in-processing search of personnel, packages, and vehicles to determine if search practices were conducted in accordance with regulatory requirements.

The inspectors reviewed security related event reports and safeguard log entries associated with the access control program for the period April 2002 through March 2003. The inspectors also reviewed the licensee's corrective action program to determine if security related issues associated with the access control program were appropriately identified and resolved.

b. Findings

No findings of significance were identified.

3PP3 Response to Contingency Events (71130.03)

a. Inspection Scope

The inspectors walked down the licensee's protected area intrusion alarm system to identify potential vulnerabilities. The inspectors, accompanied by licensee security representatives, observed testing of selected protected area intrusion alarm zones. Alarm zone detection was evaluated by conducting various testing methods.

The inspectors also reviewed the effectiveness of alarm station personnel to recognize and identify activities in the protected area alarm detection zones on the assessment monitors. The inspectors also reviewed the field of view provided by the assessment aids to ensure compliance with the licensee's security plan.

The inspectors also reviewed a sample of licensee force-on-force drill records, and interviewed security management personnel to determine if the licensee had appropriately identified and resolved issues associated with the contingency response program.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA1 Performance Indicator (PI) Verification (71151)

.1 Reactor Coolant System Specific Activity PI

a. Inspection Scope

The inspectors evaluated the chemistry department's PI data analysis methods and records to verify that the licensee had accurately assessed and reported the PI for the reactor coolant system (RCS) specific activity indicator under the barrier integrity cornerstone in accordance with the criteria specified in Nuclear Energy Institute 99-02, Revisions 1 and 2, "Regulatory Assessment Performance Indicator Guideline." Specifically, the inspectors reviewed the dose equivalent iodine calculation procedure, the RCS specific activity performance indicator procedure and interviewed members of the licensee's chemistry staff involved in the determination and verification of RCS specific activity. The inspectors also selectively reviewed the licensee's Unit 2 and Unit 3 chemistry sample analysis results for maximum dose equivalent iodine for the period between July 2002 and May 2003. These reviews were performed to verify that the licensee adequately determined dose equivalent iodine values, and to verify adherence to station procedures and to the guidance contained in Nuclear Energy Institute 99-02 relative to assessing and reporting the RCS specific activity performance indicator. Additionally, the inspectors observed a chemistry technician collect reactor water samples to verify the adequacy of the sampling technique and for consistency with the sampling procedure, and discussed with chemistry staff the calculation for dose equivalent iodine to determine its adequacy.

b. Findings

No findings of significance were identified.

.2 Physical Protection Performance Indicators

a. Inspection Scope

The inspectors verified the data for the Physical Protection Performance Indicators pertaining to Fitness-For-Duty Personnel Reliability, Personnel Screening Program and Protection Area Security Equipment. Specifically, a sample of plant reports related to security events, security shift activity logs, fitness-for-duty reports, and other applicable security records were reviewed for the period between April 2002 through March 2003.

b. <u>Findings</u>

No findings of significance were identified.

.3 <u>Initiating Events and Mitigating Systems</u>

a. <u>Inspection Scope</u>

The inspectors reviewed a sample of plant records and data against the reported performance indicators in order to determine the accuracy of the indicators.

- Unit 2 and Unit 3 Unplanned Power Changes (1^{st} Quarter 2002 through 1^{st} Quarter 2003)
- Emergency AC Power System Unavailability (1^{st} Quarter 2002 through 1st Quarter 2003)

b. <u>Findings</u>

No findings of significance were identified.

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

a. <u>Inspection Scope</u>

The inspectors conducted an inspection of the licensee's corrective action program. The inspectors selected corrective actions for two issues for periodic review of the problem identification and resolution program per NRC inspection procedure (IP) 71152. Additionally, the inspectors verified that: 1) the licensee identified issues at an appropriate threshold; 2) that these issues were correctly entered in the corrective action program; and 3) that these issues were properly addressed for resolution:

- 1A main steam isolation valve accumulator crack; and
- Standby liquid control system loose relief valve bolts.

b. <u>Findings</u>

.1 <u>1A Main Steam Isolation Valve Accumulator Crack</u>

<u>Introduction</u>. A self-revealed finding was identified on March 29, 2003, when the licensee entered the drywell to investigate leakage from the drywell pneumatic system. The licensee identified a crack in the 1A main steam isolation valve (MSIV) accumulator nozzle weld.

<u>Description</u>. On March 18, 2003, the Unit 3 onshift crew received control room drywell pneumatic low pressure alarms indicating there was leakage from the drywell pneumatic system. The operators also noted frequent cycling of the pumpback air compressors.

The licensee initiated troubleshooting activities which identified several minor leaks outside the drywell and that the load setpoint of both compressors had drifted low. The licensee concluded that there was actual leakage into the drywell, commenced a downpower on March 28, 2003, and made a drywell entry on March 29, 2003, which revealed a crack on the nozzle weld of the 1A MSIV accumulator manifold piping. The accumulator was repaired and the licensee performed a root cause investigation for this problem.

The licensee determined that manifold replacement work was performed on the 1A MSIV during the October 2002 refueling outage. During this work activity, the 24 feet of air line piping was placed under undue stress as it was cantilevered from the accumulator after disconnection from the valve manifold. This stress resulted because the workers had not added any rigid support to the piping. Also, a significant amount of other plant equipment work was performed in the same area of this valve which was believed to have contributed to the initiation of the crack of the cantilevered piping. Subsequent nondestructive examinations and local leaking rate testing were unable to detect the crack at that time because it was not large enough. The unit started up from the outage in October 2002 and subsequently the frequency of venting of the drywell began to increase. The increased venting frequency was not noticed by the station prior to the failure.

The licensee’s root cause investigation determined that the weld failure was due to mechanically induced stresses from cantilevered piping coupled with force from maneuvering the pipe in a congested work environment. Previous accumulator failures have occurred since 1987 with the most recent failure occurring on May 28, 2001, on the Unit 3 1B MSIV. This failure was similar to the 1A MSIV failure. However, the root causes of these previous failures had not been properly determined and thus the implemented corrective actions were not effective in preventing the 1A MSIV failure. The licensee determined the extent of condition was limited to both the Unit 2 and Unit 3 inboard MSIVs because of their long piping runs between the accumulator and the valve manifold.

<u>Analysis</u>

The finding was considered more than minor because the issue affected the mitigating system objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Also, the issue affected the initiating event objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during power operations. However, the issue was determined to be of low safety significance (Green) because General Electric had performed a spring plus air test using the highest containment pressure the MSIVs were required to close against and had demonstrated MSIV closure with a minimum accumulator pressure of 60 psig. The lowest drywell pneumatic pressure was 80 psi. Therefore, the 1A MSIV was capable of performing its intended safety function of fast closing in 3 to 5 seconds, and the MSIV remained open at all times. This finding was associated with the reactor safety cross-cutting attribute of human performance **FIN 50-249/03-006-03**. This issue was entered into the licensee’s corrective action program as CR 151287. This finding was not a violation of regulatory requirements because the previous and recent accumulator failures were not considered to be significant conditions adverse to quality and the individual instances were identified and corrected.

.2 Loose Bolts on the Unit 3 “A” Standby Liquid Control System Relief Valve

As a result of licensee-identified measuring and test equipment concerns, the inspector reviewed three condition reports for unrecovered (lost) torque wrenches. Of the three, the out of tolerance evaluation for CR 137925 was questionable with regard to verifying that the torque wrench was not used on safety related equipment. A new condition report, CR 159434, was initiated because the torque wrench was used on the replacement of safety-related Unit 3 standby liquid control relief valves.

The inspector reviewed the work that used torque wrench #0013622580. Work Order 00420336-07 was for replacement of the Unit 3 “A” standby liquid control relief valve and Work Order 00420336-09 was for replacement of the Unit 3 “B” standby liquid control relief valve. The “A” work package stated that on October 12, 2002, prior to beginning the task to replace the valve, the “A” relief valve bolts were found to be out of torque (finger tight). The statement did not reference whether the finger tight bolts were on the discharge side or the inlet side. The work package stated that this was an ASME Section XI component, designated the work as safety related, and required 318 foot pounds torque on the inlet flange and 70 to 78 foot pounds torque on the discharge flange bolts. The inspector identified that a condition report had not been written for the nonconforming condition nor was an operability evaluation performed for the past affect on component and system functional performance. Maintenance marked the finger tight bolts (not meeting the acceptance criteria for torque values) as Condition 7, which is “Satisfactory”.

Condition Report 162781 was initiated by Maintenance on June 11, 2003, to document the fact that no engineering operability evaluation was performed on the Unit 3 “A” standby liquid control relief valve to determine if there was a potential affect on a safety function. The disposition is pending. Upon reviewing this condition report the inspector identified that the licensee assumed that the finger tight bolts were on the discharge flange and not the inlet. However, the work package did not identify which flange had finger tight bolts. Based on the inspectors’ question, the condition report was rewritten to state that it could not be determined as to which flange was found finger tight. Engineering simultaneously initiated a separate condition report (CR 162835) on June 11, 2003, to evaluate the historical significance of finger tight bolts on system operability for both the inlet and discharge flanges. Presently, there are no concerns over the torque values on the “A” standby liquid control relief valve because it was replaced a day after the bolts were found finger tight. This is an unresolved item pending the licensee’s completion and the inspector’s review of the historical operability evaluation (**URI 50-237/03-006-04**).

4OA3 Event Follow-up (71153)

(Closed) Unresolved Item 50-237/02-17-01: Inadequate Fire Retardant Material in Fire Barriers.

Introduction. The inspectors identified a Non-Cited Violation (NCV) for 34 mechanical penetration seals not containing the minimum ceramic fire blanket to establish a 3-hour fire barrier.

Description. On October 29, 2002, while performing a fire protection walkdown of the U2 West low pressure coolant injection room, the inspector noted degraded insulation around penetration No. F-42-04 (Transco type M2) for an abandoned heating steam line. The inspector reported this condition to the licensee and questioned whether the fire seal was degraded. The inspector was informed that the degraded insulation did not make up the fire seal and that in accordance with a design drawing there should be 8 inches of ceramic fire blanket along the length of the pipe starting from the opposite side of the penetration. The opposite side of the penetration was disassembled and no fire blanket was found in the penetration. The licensee generated CR 129421 and the penetration was repaired per Work Order 504220-01. The licensee decided to inspect a sample of M2 penetrations and found more discrepancies (blanket material ranging between 2 and 6 inches). The licensee subsequently expanded the scope to inspect 100 percent of the 89 Transco type M2 penetrations, and 10 percent of the 11 Transco type M13 penetrations. As a result, 31 Transco type M-2 and 3 Transco type M-13 mechanical penetration seals were found degraded with less than the 8 inches of ceramic fire blanket. Eight of these had no ceramic fire blanket.

Analysis. This finding was determined to be more than minor because it affected the mitigating systems cornerstone objective. The requirement to have fire barriers was to ensure that a postulated fire would not propagate to more than one fire area, thus jeopardizing the availability of safe shutdown equipment. The inspectors evaluated the finding using IMC 0609, Appendix F, “Fire Protection Significance Determination Process.” The finding affected the fire barriers, one of the defense-in-depth elements. Consequently, the finding met the criteria of Phase 1, Step 1 (IMC 0609, Appendix F, Figure 4-1). A Phase 2 evaluation was performed to determined the risk significance of this finding.

The licensee determined that there was a total of 34 degraded penetration seals. Thirty-three of the 34 seals screened out of the SDP Phase 2 evaluation because: 1) credible fire scenarios could not be developed due to physical configuration of post-fire safe shutdown equipment on either side of the penetration seals or 2) the deficient penetration seals were not used to protect safe shutdown capability. Only one penetration seal, F-87-03 (west wall of the Unit 2 isolation condenser “2" valve room), was evaluated using the Phase 2 evaluation since the penetration seal forming the fire area boundary interface with recovery areas was affected by the finding. A transient fire could occur near the deficient seal on the Unit 2 reactor building side since transient combustible material was routinely stored on the floor under the penetration seal. Combustion products could potentially damage the seal and propagate to the isolation condenser area where manual operation of valves was needed.

The licensee used ignition frequencies of 7.26E-7 per reactor year for a transient fire from an extension cord based on Electric Power Research Institute (EPRI) FIVE Methodology ($\log_{10}$(IF)= -6.14). The penetration seal was assumed to be highly degraded because no ceramic fire blanket was found (FB=0). There was no automatic fire suppression and manual fire fighting capabilities were assumed to be in normal operating states because no finding was identified within this capability (AS=0; MS= -1). Since the exposure time for the degraded condition existed more than 30 days, the estimated likelihood rating for the postulated fire event was determined to be 1E-7 per reactor year.

The inspectors determined that the recovery actions in the isolation condenser room could be successfully implemented since this room had high ceiling and the location for manual operation of the safe shutdown path valve was 17 feet away from the penetration seal. Therefore, credit was given for implementation of the alternative shutdown strategy. Based upon the inspectors' evaluation of the Fire Protection SDP using these inputs, the finding was determined to be Green.

Enforcement. Unit 2 and 3 License Nos. DPR-19 and DPR-25 require that the licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report (UFSAR). Section 9.5.1, "Fire Protection System," of the UFSAR states, in part, that the design bases and system descriptions are described in the Dresden Fire Protection Reports. Dresden Fire Protection Report Volume 1, "Updated Fire Hazards Analysis," Section 2.3.1.3, states, in part, that fire penetration seals provided in fire barriers are documented on the F-drawings (Drawings F-41 through F-196, F-353 and F-457) and in penetration details for mechanical penetration seals. The F-drawings and penetration details require a minimum of 8 inches of ceramic fire blanket to establish a 3-hour rated fire barrier. Contrary to the above 31Transco type M-2 and 3 Transco type M-13 mechanical penetration seals had less than the 8 inches of ceramic fire blanket. Because the degraded penetration seals were of very low safety significance and have been entered into the licensee's corrective action program in numerous condition reports (including CR 143871), this violation is being treated as a Non-Cited Violation (**NCV 50-237/50-249/03-006-05**), consistent with Section VI.A of the NRC Enforcement Policy.

4OA4 Cross-Cutting Issues

Section 4OA2 of this report describes ineffective corrective actions implemented by the licensee in resolving cracks associated with both Unit 2 and 3 main steam isolation valve manifold blocks and accumulators. As noted in 4OA2, the licensee discovered the tenth crack with the 1A MSIV on March 29, 2003, while troubleshooting leakage from the drywell pneumatic system.

Section 71111.14 describes the licensee's failure to follow a procedure during power ascension which resulted in the operators inadvertently exceeding the technical specification for reactor steam dome pressure of 1005 on May 12, 2003.

4OA6 Meetings

.1 Exit Meeting

The inspectors presented the inspection results to Mr. R. Hovey and other members of the licensee's staff on June 24, 2003. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

.2 Interim Exit Meetings

- Safeguards Inspection with Ms. V. Gengler on April 11, 2003.
- Radiation Protection inspection with Mr. J. Henry on May 16, 2003.
- Occupational radiation safety inspection of access controls for radiologically significant areas with Mr. D. Bost on June 27, 2003.
- Operator Licensing Requalification with Mr. R. Rybak on June 27, 2003.

4OA7 Licensee-Identified Violations

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

Cornerstone: Mitigating Systems

Title 10 of the Code of Federal Regulations (CFR) Part 55.53, states, in part, each license contains and is subject to the following conditions whether stated in the license or not:

(h) The licensee shall complete a requalification program as described by 10 CFR 55.59.

Title 10 CFR 55.59 (a), "Requalification Requirements,"states, in part, that each licensee shall...

(1) Successfully complete a requalification program developed by the facility licensee that has been approved by the Commission. This program shall be conducted for a continuous period not to exceed 24 months in duration.

(2) Pass a comprehensive requalification written examination and an annual operating test.

1. As described in CR 00113996, 54 licensed operators did not meet the requalification examination requirements of 10 CFR 55.59. A comprehensive written examination for the 24 month requalification period defined by the licensee as January 10, 2000 through January 4, 2002, was not administered to the operators by the station training department personnel within the time frame required by

10 CFR 55.59. This resulted in 54 licensed operators not being in compliance with 10 CFR 55.53 (h) from January 5, 2002, until all licensed operators successfully completed a comprehensive written examination on July 17, 2002. Unresolved Items (URI) 50-237/249/02-15-01(DRS) and URI 50-237/249/02-15-02(DRS) are closed.

2. As described in CR 00117708, 28 licensed operators did not meet the requalification examination requirements of 10 CFR 55.59 (a)(1) and (a)(2). A comprehensive written examination for the 24 month requalification period defined by the licensee as January 30, 1998, through January 30, 2000, was not administered to the operators by the station training department personnel within the time frame required by 10 CFR 55.59 (a)(1) and (a)(2). This resulted in 28 licensed operators not being in compliance with 10 CFR 55.53 (h) from January 31, 2000, until all licensed operators successfully completed a comprehensive written examination on February 21, 2000. URI 50-237/249/02-15-03(DRS) is closed.

3. As described in CR 00120517, 10 licensed operators did not meet the requalification examination requirements of 10 CFR 55.59 (a)(1) and (a)(2). An annual operating test was not administered to 10 licensed operators during calendar year 2001. The licensee was not in compliance from January 1, 2002, until the 10 licensed operators successfully completed an operating test on January 4, 2002. URI 50-237/249/02-15-04(DRS) is closed.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
D. Bost, Station Director
H. Bush, Lead Radiation Protection Supervisor
R. Conklin, Radiation Protection Supervisor
T. Fisk, Chemistry Manager
R. Gadbois, Shift Operations Superintendent
V. Gengler, Dresden Site Security Director
J. Hansen, Regulatory Assurance Manager
J. Henry, Operations Director
R. Hovey, Site Vice President
C. Kolotka, Chemistry Supervisor
T. Loch, Supervisor, Turbine Systems Group
D. Nestle, Radiation Protection Technical Manager
M. Overstreet, Radiation Protection Supervisor
R. Quick, Security Manager
R. Ruffin, Regulatory Assurance - NRC Coordinator
R. Rybak, Acting Regulatory Assurance Manager
F. Sadnick, Project Manager, Wackenhut Corporate
A. Shahkarami, Engineering Director
J. Sipek, Nuclear Oversight Director
N. Spooner, Site Maintenance Rule Coordinator
B. Svaleson, Maintenance Director
S. Taylor, Radiation Protection Director

Nuclear Regulatory Commission
M. Ring, Chief, Division of Reactor Projects, Branch 1

IDNS
R. Schulz, Illinois Department of Nuclear Safety

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-237/03-006-01	NCV	Reactor Steam Dome Pressure Exceeds Technical Specification Limit
50-237/03-006-02	NCV	Failure to follow work order rendered the Unit 2 "D" Electromatic Relief Valve Inoperable
50-249/03-006-03	FIN	Ineffective Corrective Actions to Resolve Main Steam Isolation Valve Accumulator
50-237/03-006-04	URI	Loose Bolts on the Standby Liquid Control System Relief Valve
50-237/03-006-05 50-249/03-006-05	NCV	Degraded Mechanical Penetration Fire Barriers

Closed

50-237/03-006-01	NCV	Reactor Steam Dome Pressure Exceeds Technical Specification Limit
50-237/03-006-02	NCV	Failure to follow work order rendered the Unit 2 "D" Electromatic Relief Valve Inoperable
50-249/03-006-03	FIN	Ineffective Corrective Actions to Resolve Main Steam Isolation Valve Accumulator
50-237/03-006-05 50-249/03-006-05	NCV	Degraded Mechanical Penetration Fire Barriers
50-237/02-17-01	URI	Inadequate Fire Retardant Material in Fire Barriers
50-237, 249/02-15-01(DRS) 50-237, 249/02-15-02(DRS)	URI	54 licensed operators did not meet the requalification examination requirements of 10 CFR 55.59.
50-237, 249/02-15-03(DRS)	URI	28 licensed operators did not meet the requalification examination requirements of 10 CFR 55.59 (a)(1) and (a)(2).
50-237, 249/02-15-04(DRS)	URI	10 licensed operators did not meet the requalification examination requirements of 10 CFR 55.59 (a)(1) and (a)(2).

Discussed

None.

LIST OF ACRONYMS USED

ALARA	As-Low-As-Is-Reasonably-Achievable
CFR	Code of Federal Regulations
CR	Condition Report
DIS	Dresden Instrument Surveillance
DOS	Dresden Operating Surveillance
DOT	Department of Transportation
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
EC	Engineering Change
ERV	Electromatic Relief Valve
FSAR	Final Safety Analysis Report
IDNS	Illinois Department of Nuclear Safety
IMC	Inspection Manual Chapter
LHRA	Locked High Radiation Area
MWe	megawatts electrical
MSIV	Main Steam Isolation Valve
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
OA	Other Activities
OE	Operability Evaluation
PI	Performance Indicator
RCS	Reactor Coolant System
RP	Radiation Protection
SDP	Significance Determination Process
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
VHRA	Very High Radiation Area
WO	Work Order

LIST OF DOCUMENTS REVIEWED

1R01 Adverse Weather Protection

OP-AA-108-109, ‘‘Seasonal Readiness,” Revision 1

DOA 0010-02, “Tornado Warning/Severe Winds,” Revision 4

NOS Summer Readiness Assessment (3/24 - 4/4/2003)

CR 00132155; Summer Readiness; November 18, 2002

“Summer Readiness System Engineer Material Condition Challenge Boards,” report dated May 14, 2003

DOP 6900-18, “125 VDC and 250 VDC Portable Battery Charger Use,” Revision 0

DOS 0010-24, “Securing from Cold Weather Operations for Unit 2,” Revision 7

DOS 0010-27, “Securing from Cold Weather Operations for Unit 3,” Revision 5

1R04 Equipment Alignment

DOP 1400-M1; “Unit 2 Core Spray System,” Revision 20

DOP 1400-E1; “Unit 2 Core Spray Electrical,” Revision 3

CR 157770; Failure analysis for D2 high pressure coolant injection system motor gear unit removed motor; May 8, 2003

CR 160447; Configuration control event - instrument air valve found closed; May 25, 2003

CR 162417; Safe shutdown emergency light #230 failed need to replace battery; June 9, 2003

DOP 6900-E4, “Unit 2 DC Electrical Systems Checklist,” Revision 11

DOP 6900-01, “250 VDC Electrical System,” Revision 25

System Description Manual 263001, “250 VDC Distribution System”

UFSAR 8.3.2.1.1, “250-V System”

Technical Specification 3.8.4, “DC Sources - Operating”

Drawing 12E-2321, “Key Diagram 250V DC Motor Control Centers”

Assignment Report 00146934-10, “Complete UFSAR Change 97120 and Enter NLI Calc into Passport”

NLI calculation C-017-081-2, “Evaluate the GNB Class 1E Batteries installed in the Dresden and Quad Cities Stations to Determine the Design Life based on Recent Qualification Testing,” dated 12/10/97

1R05 Fire Protection

CR 151123; Safe shutdown light 229 inoperable; April 1, 2003

CR 151147; Safe shutdown 261 failed acceptance criteria; April 1, 2003

CR 154067; 2/3 diesel fire pump; April 18, 2003

CR 154209; Penetration F-143-01 found with 4" of ceramic fire blanket; April 18, 2003

CR 154066; 2/3 diesel fire pump auto start; April 18, 2003

CR 155618; Penetration F-138-4 Fire Barrier is not installed; April 29, 2003

CR 156332; Oil leak on diesel engine for unit 2/3 diesel fire pump; April 29, 2003

CR 156982; Incorrect acceptance curves in fire pump capacity survey; May 1, 2003

CR 159219; 2/3 annual fire door inspections; May 16, 2003

CR 161902; Fire doors inoperable due to excessive clearances; June 4, 2003

CR 163871; Unexpected safe shutdown light failure and Technical Requirement Manual entry; June 18, 2003

CR 164097; Nuclear Oversight identified transient combustible log deficiencies; June 19, 2003

CR 164083; Nuclear Oversight identified inadequate resources for fire drill control; June 19, 2003

1R11 Operator Requalification

CR 152592; Failed training exams; April 10, 2003

CR 157200; Licensed operator requalification training OBE failure; April 28, 2003

1R12 Maintenance Effectiveness

CR 159254; Logic errors found in Dresden shutdown risk models; May 12, 2003

1R13 Maintenance Risk Assessments and Emergent Work Control

CR 151969; Foreign material found in high pressure coolant injection line; April 7, 2003

CR 152170; 2A reactor feed pump vent fan damper connected wrong after removing gag; April 8, 2003

1R15 Operability Evaluations

Operability Evaluation 03-006; Units 2 and 3 Electromatic Relief Valve/Target Rock Valve Pressure Switches 2(3)-0203-3A, B, C, D, and E Setpoint Drift; dated June 2, 2003

Engineering Change Nos. 343023 and 343024; "Interim Revision to the Electromatic Relief Valve/Target Rock Valve Pressure Switch Setpoints;" dated May 30, 2003

CR 00159552; DIS 0250-03 Emergency Relief Valve (ERV)/Target Rock Pressure Switches out of tolerance Tech Spec Violation; May 19, 2003

CR 00159576; ERV pressure switch 3-203-3E out of tolerance; May 19, 2003

CR 00159815; ERV pressure controller 2-203-3B was found out of tolerance; May 21, 2003

CR 00159816; ERV pressure controller 2-203-3C was found out of tolerance; May 21, 2003

1R16 Operator Workaround

CR#160078; Evaluate need to adjust reactor pressure as an Operator Workaround; May 12, 2003

1R19 Post Maintenance Testing

DOS 1500-02; "Containment Cooling Service Water Pump Test and Inservice Test (IST)," Revision 47

WO 020057995503; VT2 per NSP ER-AA-335-015, "VT-2 Visual Examination, " Revision 2

CR 160419; 3D containment cooling service water pump discharge check valve 3-1501-1D leaks excessively; May 24, 2003

CR 163084; Installed jumper in wrong panel; 06/13/03

Prompt Investigation, "Installed Jumper in Wrong Panel" (2-1001-4C)

WO 99022222, D2 6Y PM SURV limitorque valve operator 2-1001-4C

1R22 Surveillance Test

CR 152065; High pressure coolant injection surveillance data sheet information; April 8, 2003

CR 151413; CV-1 fast closure failure; April 8, 2003

CR 152829; Negative trend - containment cooling service water vault penetration seal test failures; April 10, 2002

CR 154100; Reactor medium range level isolator out of tolerance; April 22, 2003

CR 154101; Standby liquid control level switch and local indicator out of tolerance; April 22, 2003

CR 155928; Recorder failure results in extra emergency diesel generator start; April 30, 2003

CR 156174; UFSAR diesel generator cooling water pump in-service testing concern; April 30, 2003

CR 156567; Votes testing of 3-1301-2 valve; April 30, 2003

CR 157082; Narrow range level transmitter LT 2-646A found out of tolerance; May 2, 2003

CR 157141; 24-volt master trip solenoid valve test (weekly); May 4, 2003

CR 159552; DIS 0250-03 “Electromatic Relief Valve/Target Rock Valve Pressure Switch Calibration,” Revision 37/ Target Rock plant specifications out of tolerance technical specifications violation; May 19, 2003

DIS 0250-03; “Electromatic Relief Valve/Target Rock Valve Pressure Switch Calibration,” Revision 37

CR 159576; Emergency relief valve pressure switch 3-203-3E out of tolerance; May 19, 2003

CR 159811; DOS 0040-12 not revised to reflect engineering change; May 21, 2003

CR 160111; Annual core spray pump discharge pressure calibration; May 22, 2003

CR 160528; Turbine control valve #1 did not fast close at 5% position; May 27, 2003

CR 160684; DIS 1300-03 stopped due to unexpected isolation condenser response; May 28, 2003

CR 161798; DIS 3900-01 Revision 19 doesn’t work for Unit 2 - 24 month calendar; June 2, 2003

DOS 1100-02; “Standby Liquid Control Tank Heater Surveillance Test,” Revision 11

DOS 1100-04; “Quarterly Standby Liquid Control System Pump Test for the Inservice Testing (IST) Program,” Revision 24

DOS 6600-01; “Diesel Generator Surveillance Tests,” Revision 80

CR 163196; Pump failed to reach required flow; June 13, 2002

CR 163643; Full scram vulnerability during quarterly control valve testing; June 17, 2003

CR 163882; Unit 2 Emergency diesel generator intermittent field ground alarm; June 18, 2003

CR 164095; Containment cooling service water piping movement observed when starting 3A CCSW pump; June 19, 2003

CR 164855; Poor Housekeeping - valve wrench stored in standby liquid control support; June 25, 2003

WO 00567519; Unit 2 1M TS sustained high reactor pressure; April 22, 2003

71120 Refueling and Outage Activities

CR 161933; Elevated hydrogen gas detected in stator cooling water roof vent; June 4, 2003

71152 Problem and Identification Resolution

CR 156394; Maintenance rule functional failure found during work order review; April 28, 2003

CR 161568; Apparent NRC concern with condition report resolution; June 2, 2003

CR 161861; Station response to NRC identified concern inadequate; June 2, 2003

CR 162361; Inadequate information pertaining to an erect scaffold; June 8, 2003

2OS1 Access Control to Radiologically Significant Areas

RP-AA-460; Controls for High and Very High Radiation Areas; Revision 2

RP-AA-460; Attachments 1 - 5 for Selected Periods in 2003

RP-AB-460; TIP Area Access Controls; Revision 0

RP-DR-JOB-025; Posting LHRA & VHRA Areas; Revision 0

RP-DR-ALR-001; Steam Sensitive Area Entries; Revision 2

RP-DR-JOB-001; Movement or Transfer of Highly Radioactive Material; Revision 4

DOP 1600-22; Drywell Entry (Initial or at Power); Revision 14

DRS 5600-01; Quarterly High, Locked High, and Very High Radiation Posting and Door Checks; Revision 07 and Completed Checklist A for 1st and 2nd Quarter of 2003

RP-AA-210; Dosimetry Issue, Usage, and Control; Revision 3

INPO 91-014; Draft Guidelines for Radiological Protection at Nuclear Power Stations; Revision 2

DFP 0800-39; Control of Material/Equipment Hanging in Units 2 and 3 Spent Fuel Pools; Revision 13

MA-AA-716-008; Foreign Material Exclusion Program; Revision 1

Dresden Unit 2 and Unit 3 Fuel Pool Inventory; May 2003
RP-AA-220; Bioassay Program; Revision 1

Internal Dose Calculation Records and Associated Whole Body Count Results for Selected Workers in October and November 2002

Focus Area Self-Assessment Report; Access Control to Radiological Significant Areas and ALARA Planning and Controls; June 9 - 11, 2003

Nuclear Oversight (Quarterly) Continuous Assessment Reports; NOA-DR-02-2Q, 02-3Q, 02-4Q, and 03-1Q

Radiation Protection CR Database Listing for November 2002 - June 2003

CR 00165219; Procedure Violation Re: Items in Spent Fuel Pools; dated June 25, 2003 (NRC-identified)

CR 00165237; NRC Identifies RP Program Enhancement Opportunities; dated June 25, 2003 (NRC-identified)

CR 00165223; Incomplete/Inaccurate Entries in LHRA/VHRA Key Log; dated June 25, 2003 (NRC-identified)

CR 00150394; Additional Exposure Received for Reactor Water Cleanup Work; dated March 21, 2003

CR 00149162; High Radiation Area @ U2 HPCI With No Brief; dated March 14, 2003

CR 00131842; High Radiation Area Discovered in D3 Reactor Building; dated November 15, 2002

CR 00158527; Locked High Radiation Area Postings Not Removed; dated May 11 2003

CR 00136561; High Rad Bucket Found Tied-Off of Hand Rail; dated December 27, 2002

CR 00 151403; Excessive Dose Picked-up When Dose Profiling Radwaste Liners; dated March 24, 2003

Common Cause Analysis Report; Potential Negative Trend Identified in RP Boundaries; dated June 19, 2003

CR 134958; Drywell entry on wrong radiation work permit; January 28, 2003

2PS2 <u>Radioactive Material Processing and Transportation</u>

CR 114259; Findings from radwaste material condition/equipment assessment; December 3, 2002

CR 127754; Need to get controllers replaced as soon as possible; December 18, 2002

CR 128793; Received CR Annun2/3-923-5 E-3 during steady state operation; October 28, 2002

CR 132244; Material condition of maximum recycle general area; November 25, 2002

CR 134637; Maximum concentrator recycle pump trip; December 16, 2002

CR 135575; 3B condensate pump seals damage; February 12, 2003

CR 137443; Significant amounts of resin have been found in the waste storage tanks; January 7, 2003

CR 138475; Leak on resin transfer line from Unit 2 regenerator room to radiation waste; January 9, 2003

CR 143321; Spend resin pump would not start; February 13, 2002

CR 143636; B max recycle concentrator recirculation pump tripped; February 12, 2003

CR 144865; Radioactive shipment arrives at Barnwell missing Department of Transportation marking; February 21, 2002

CR 146985; 2/3 B Distillate pump spuriously trips; March 5, 2003

CR 152120; Resins found in river discharge receiver; April 16, 2003

CR 152964; Resin found in Unit 3 control rod drive pump suction filter; April 16, 2003

CR 154039; Radwaste solids material condition issues; April 16, 2003

NOA-DR-02-1Q; Nuclear Oversight Continuous Assessment Report;
January-March 2002

NOA-DR-02-2Q; Nuclear Oversight Continuous Assessment Report; April-June 2002

NOA-DR-02-3Q; Nuclear Oversight Continuous Assessment Report;
July-September 2002

NOA-DR-02-4Q; Nuclear Oversight Continuous Assessment Report;
October-December 2002

NOS Field Observation; Proper placarding/radiation postings; September 30, 2002

Radwaste Inleakage List; May 13, 2002

Radwaste Material Condition; Focus Area Self-Assessment Plan #2002-001

Training Administration System Total Course Completion Report; May 14, 2003

RP-DR-605, Revision 0; "10CFR61 Waste Stream Sampling Analysis"

RP-DR-605, Revision 0; Data Sheets; May 2, 2003

Focus Area Self-Assessment Report 141537; February 1, 2003 - March 15, 2003

0403-11795; Type B, Unit 1 Filters.; April 11. 2003

0203-11723; Type B, Unit 2/3 Filters.; February 21. 2003

0103-11681; Type B, Activated Hardware; January 13. 2003

0103-11686; Type B, Unit 1 Filters.; January 17, 2003

DW-03-031; LSA II DAW; May 15, 2003

2OS3 <u>Radiation Monitoring Instrumentation and Protective Equipment</u>

RP-AA-440; Respiratory Protection Program; Revision 3

Confined Space Rescue Callout List; undated

SA-AA-114; Confined Space Entry; Revision 3
RWP 10001576 and Associated ALARA Plan; D3R17 Drywell Control Rod Drive Replacement; Revision 2

RWP 013007 and Associated ALARA Plan; Reactor Water Cleanup Demin Vessel Inspection and Repair; Revision 1

3PP2 Access Control (Identification, Authorization and Search of Personnel, Packages, and Vehicles)

SY-AA-101-122; Testing Security Equipment; Revision 6

Security Logged Events; April 2002 through March 2003

3PP3 Response to Contingency Events

Force-on-Force Exercises; August 2002 to February 2003

SY-AA-101-122; Testing Security Equipment; Revision 6

Dresden Condition Reports (Security Related); November 2002 to March 2003

4OA1 Performance Indicator Verification

DCP 3207-01; Gamma Isotopic Analysis; Revision 17

Attachment 13 to DCP 3207-01; Dose Equivalent Iodine Calculation; Selected Records from July 2002 - May 2003

DCP 1019-01; Plant System Sampling; Revision 31

Reactor Coolant System Isotopic Sampling Results and Associated Dose Equivalent Iodine Calculations for Selected Periods Between November 2002 and May 2003

RS-AA-122-117; Performance Indicator - Protection Area Security Equipment

RS-AA-122-118; Performance Indicator - Personnel Screening Program

RS-AA-122-119; Performance Indicator - Fitness-for-Duty

Security Event Reports; April 2002 through March 2003

CR 152093; unit 3 unplanned power changes indicator in variance; April 9, 2003

4OA7 Licensee-Identified Violations

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