UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
2443 WARRENVILLE ROAD, SUITE 210
LISLE, IL 60532-4352

February 2, 2009

Mr. Charles G. Pardee
Senior Vice President, Exelon Generation Company, LLC
President and Chief Nuclear Officer (CNO), Exelon Nuclear
4300 Winfield Road
Warrenville IL 60555

SUBJECT: DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3 INTEGRATED INSPECTION REPORT 05000237/2008-005; 05000249/2008-005

Dear Mr. Pardee:

On December 31, 2008, the U.S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your Dresden Nuclear Power Station, Units 2 and 3. The enclosed report documents the inspection findings, which were discussed on January 13, 2009, with Mr. D. Wozniak and other members of your staff.

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

Based on the results of this inspection, two NRC-identified findings of very low safety significance were identified. These findings involved violations of NRC requirements. However, because of their very low safety significance, and because the issues were entered into your corrective action program (CAP), the NRC is treating the issues as Non-Cited Violations (NCVs) in accordance with Section VI.A.1 of the NRC Enforcement Policy.

If you contest the subject or severity of a NCV, 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 a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - 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 Resident Inspector Office at the Dresden Station.

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

Sincerely,

/RA/

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

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

Enclosure: Inspection Report 05000237/2008-005; 05000249/2008-005
w/Attachment: Supplemental Information

cc w/encl: Site Vice President - Dresden Nuclear Power Station
Plant Manager - Dresden Nuclear Power Station
Manager Regulatory Assurance – Dresden Nuclear Power Station
Senior Vice President - Midwest Operations
Senior Vice President - Operations Support
Vice President - Licensing and Regulatory Affairs
Director - Licensing and Regulatory Affairs
Manager Licensing - Clinton, Dresden, and Quad Cities
Associate General Counsel
Document Control Desk - Licensing
Assistant Attorney General
J. Klinger, State Liaison Officer,
Illinois Emergency Management Agency
Chairman, Illinois Commerce Commission

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Letter to C. Pardee from M. Ring dated February 2, 2009

SUBJECT: DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3 INTEGRATED INSPECTION REPORT 05000237/2008-005; 05000249/2008-005

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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:	05000237/2008-005; 05000249/2008-005
Licensee:	Exelon Generation Company
Facility:	Dresden Nuclear Power Station, Units 2 and 3
Location:	Morris, IL
Dates:	October 1 through December 31, 2008
Inspectors:	C. Phillips, Senior Resident Inspector D. Meléndez-Colón, Resident Inspector F. Ramirez, Resident Inspector, LaSalle Station W. Slawinski, Senior Health Physicist C. Acosta Acevedo, Reactor Engineer J. Draper, Reactor Engineer J. Bozga, Reactor Inspector C. Brown, Reactor Engineer R. Murray, Reactor Engineer R. Winter, Reactor Inspector T. Bilik, Reactor Inspector E. Sanchez-Santiago, Observer
Approved by:	M. Ring, Chief Projects Branch 1 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000237/2008-005, 05000249/2008-005; 10/01/2008 - 12/31/2008; Dresden Nuclear Power Station, Units 2 & 3; Post-Maintenance Testing, Heat Sink Performance.

This report covers a three-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. Two Green findings were identified by the inspectors. The findings were considered NCVs of NRC regulations. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealed Findings

Cornerstone: Initiating Events

- Green. A finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified by NRC inspectors on November 12, 2008, when the licensee had declared a freeze seal established prior to meeting the requirements of procedure MA-AA-736-610, "Application of Freeze Seal to All Piping," Revision 3. The licensee took corrective actions that included counseling the first line supervisor and the engineer involved in the work.

  The finding was determined to be more than minor because the finding, if left uncorrected, would become a more significant safety concern. Specifically, the inspectors determined that the licensee had determined the freeze seal to be acceptable before it was allowed by procedure. Had there been a problem with the freeze seal, there may not have been adequate time to react and implement any required contingency actions. The inspectors concluded this finding was associated with the Initiating Events Cornerstone. This finding has a cross-cutting aspect in the area of Human Performance, H.1.b, because the licensee did not make a conservative assumption in decision making. (Section 1R19)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance involving the low pressure coolant injection (LPCI) heat exchangers' cooling capability during a design basis loss of coolant accident (LOCA). Specifically, the licensee failed to evaluate the effects of higher containment pressure post power up-rate on the LPCI heat exchangers' differential pressure set-point calculation. In response to the issue, the licensee implemented compensatory actions including updating various calculations and performing several operability evaluations.

  This finding was more than minor because there was reasonable doubt concerning the operability of the LPCI heat exchangers and if left uncorrected, these heat exchangers had the potential to be inoperable during the summer months. This finding was of very low safety significance because the inspectors determined that the LPCI heat exchangers were in a non-conforming but operable condition and the issue screened as Green using the SDP Phase 1 screening worksheet. (Section 1R07)

B. Licensee-Identified Violations

No violations of significance were identified.

REPORT DETAILS

Summary of Plant Status

Unit 2

The unit began this period recovering from a forced outage to replace the 2D3 feedwater heater extraction line expansion bellows. The unit returned to full power on October 1, 2008.

On December 13, 2008, power was reduced to 59 percent power to perform turbine valve testing, control rod drive scram time testing, control rod pattern adjustment, and other activities. The unit returned to full power on the same day.

Unit 3

The unit began this period continuing with fuel coastdown.

On November 3, 2008, the unit was taken offline to perform its regularly scheduled refueling outage and various other activities. The unit returned to full power on November 23, 2008.

On December 6, 2008, power was reduced to 66 percent to gather reactor recirculation data and perform a control rod pattern adjustment. The unit returned to full power on the same day.

1. REACTOR SAFETY

1R01 Adverse Weather Protection (71111.01)

.1 Readiness of Offsite and Alternate AC Power Systems – This inspection sample was completed by May 30, 2008, second quarter of the inspection year, but was not included in the inspection report (report no. 2008-003).

a. Inspection Scope

The inspectors verified that plant features and procedures for operation and continued availability of offsite and alternate AC power systems during adverse weather were appropriate. The inspectors reviewed the licensee's procedures affecting these areas and the communications protocols between the transmission system operator (TSO) and the plant to verify that the appropriate information was being exchanged when issues arose that could impact the offsite power system. Examples of aspects considered in the inspectors' review included:

- The coordination between the TSO and the plant during off-normal or emergency events;
- The explanations for the events;
- The estimates of when the offsite power system would be returned to a normal state; and
- The notifications from the TSO to the plant when the offsite power system was returned to normal.

The inspectors also verified that plant procedures addressed measures to monitor and maintain availability and reliability of both the offsite AC power system and the onsite alternate AC power system prior to or during adverse weather conditions. Specifically, the inspectors verified that the procedures addressed the following:

- The actions to be taken when notified by the TSO that the post-trip voltage of the offsite power system at the plant would not be acceptable to assure the continued operation of the safety-related loads without transferring to the onsite power supply;
- The compensatory actions identified to be performed if it would not be possible to predict the post-trip voltage at the plant for the current grid conditions;
- A re-assessment of plant risk based on maintenance activities which could affect grid reliability, or the ability of the transmission system to provide offsite power; and
- The communications between the plant and the TSO when changes at the plant could impact the transmission system, or when the capability of the transmission system to provide adequate offsite power was challenged.

Documents reviewed are listed in the Attachment to inspection report no. 2008-003. The inspectors also reviewed CAP items to verify that the licensee was identifying adverse weather issues at an appropriate threshold and entering them into their CAP in accordance with station corrective action procedures.

This inspection constitutes one readiness of offsite and alternate AC power systems sample as defined in Inspection Procedure (IP) 71111.01-05.

b. Findings

No findings of significance were identified.

.2 Winter Seasonal Readiness Preparations

a. Inspection Scope

The inspectors conducted a review 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. Documentation for selected risk-significant systems was reviewed to ensure that these systems would remain functional when challenged by inclement weather. During the inspection, the inspectors focused on plant specific design features and the licensee's procedures used to mitigate or respond to adverse weather conditions. Additionally, the inspectors reviewed the Updated Final Safety Analysis Report (UFSAR) and performance requirements for systems selected for inspection, and verified that operator actions were appropriate as specified by plant specific procedures. Cold weather protection, such as heat tracing and area heaters, was verified to be in operation where applicable. The inspectors also reviewed CAP items to verify that the licensee was identifying adverse weather issues at an appropriate threshold and entering them into their CAP in accordance with station corrective action procedures. Specific documents reviewed during this inspection are listed in the Attachment to this report. The inspectors' reviews focused specifically on the following plant systems due to their risk-significance or susceptibility to cold weather issues:

- Isolation Condenser Makeup Pumps;
- and 2/3 Emergency Diesel Generator.

This inspection constituted one winter seasonal readiness preparations sample as defined in IP 71111.01-05.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Quarterly Partial System Walkdowns

a. Inspection Scope

The inspectors performed a partial system walkdown of the following risk-significant system:

- 2A Containment Cooling Service Water (CCSW) - 2B CCSW out-of-service for pipe leak repair.

The inspectors selected this system based on its risk-significance relative to the Reactor Safety Cornerstones. The inspectors attempted to identify any discrepancies that could impact the function of the system, and, therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, system diagrams, UFSAR, TS requirements, outstanding work orders, condition reports, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the system incapable of performing its intended functions. The inspectors also walked down accessible portions of the system to verify system components and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no obvious deficiencies. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the CAP with the appropriate significance characterization. Documents reviewed are listed in the Attachment to this report.

These activities constituted one partial system walkdown sample as defined in IP 71111.04-05.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Routine Resident Inspector Tours (71111.05Q)

a. Inspection Scope

The inspectors conducted fire protection walkdowns which were focused on availability, accessibility, and the condition of firefighting equipment in the following risk-significant plant areas:

- Fire Zone 8.2.2.A, elevation 495', Unit 2 containment cooling service water and control rod drive pump area;
- Fire Zone 8.2.6.A, elevation 534', Unit 2 turbine building switchgear area;
- Fire Zone 18.7.1, elevation 517', Isolation condenser pump house, north cubicle and Fire Zone 18.7.2 Isolation condenser pump house, south cubicle; and
- Fire Zone 8.2.5.A, elevation 517', Unit 2 reactor feed pump area.

The inspectors reviewed areas to assess if the licensee had implemented a fire protection program that adequately controlled combustibles and ignition sources within the plant, effectively maintained fire detection and suppression capability, maintained passive fire protection features in good material condition, and had implemented adequate compensatory measures for out of service, degraded or inoperable fire protection equipment, systems, or features in accordance with the licensee's fire plan. The inspectors selected fire areas based on their overall contribution to internal fire risk as documented in the plant's Individual Plant Examination of External Events with later additional insights, their potential to impact equipment which could initiate or mitigate a plant transient, or their impact on the plant's ability to respond to a security event. Using the documents listed in the Attachment, the inspectors verified that fire hoses and extinguishers were in their designated locations and available for immediate use; that fire detectors and sprinklers were unobstructed, that transient material loading was within the analyzed limits; and fire doors, dampers, and penetration seals appeared to be in satisfactory condition. The inspectors also verified that minor issues identified during the inspection were entered into the licensee's CAP. Documents reviewed are listed in the Attachment to this report.

These activities constituted four quarterly fire protection inspection samples as defined in IP 71111.05-05.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07T)

.1 Triennial Heat Sink Performance

a. Inspection Scope

Unresolved item (URI) 5000237/2008002–01; 05000249/2008002–01, was opened during the 2008 triennial heat sink performance inspection due to deficiencies and discrepancies in the low pressure coolant injection (LPCI) heat exchangers' design calculation. During this inspection period, the inspectors reviewed related documents to determine the adequacy of the licensee's past operability evaluation. This URI is closed under Section 4OA5.3 of this report. This review did not represent an inspection sample.

b. Findings

LPCI Heat Exchangers' Design Calculation Deficiencies and Discrepancies

Introduction: The inspectors identified a NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance (Green) related to the LPCI Heat Exchangers' ability to remove heat during design basis loss of coolant accident (LOCA) conditions. Specifically, the licensee failed to evaluate the effects of higher containment pressure post power up-rate on the LPCI heat exchangers' differential pressure set-point calculation.

Description: While completing actions associated with a concern raised during the 2007 component design basis inspection, the licensee discovered that the design input of containment over pressurization was incorrect. The containment pressure had not been updated post power up-rate and the actual value was higher than that assumed in the differential pressure set-point calculation across the LPCI heat exchangers.

The design requirement of this differential pressure set point is to prevent radioactive water from leaking into the containment cooling service water (CCSW) and being discharged to the Kankakee River. In order to achieve this, the pressure across the LPCI heat exchanger is maintained at a set-point of 20 psid. The higher containment pressure identified through this issue resulted in a higher LPCI system pressure. Therefore, in order to maintain 20 psid across the heat exchanger, the CCSW system needed to operate at a higher system pressure than assumed in the calculation.

The licensee determined that the required throttling of the CCSW system flow in order to maintain the higher operating pressure would result in a reduced CCSW system flow rate in all cases lower than the minimum required flow rate to mitigate a design basis accident (5,000 gpm). In addition, during a routine thermal performance test on March 21, 2008, the licensee determined that the 3B LPCI heat exchanger's heat removal capability was below the design heat removal requirement.

The licensee initiated assignment report (AR) 763663 and determined that the heat exchangers were operable at a maximum river temperature of 85 degrees Fahrenheit (°F), which is lower than their maximum design river temperature of 95 °F. The inspectors opened this unresolved item pending completion of the licensee's past operability determination and the inspectors' review of this evaluation.

Consequently, the licensee completed several operability evaluations, revised various calculations and took some actions to evaluate these degraded and non-conforming conditions. These actions included the following:

- A revision to the CCSW flow balance which used a more accurate computer program model and eliminated some conservatism included in the calculation.

- A revision to the differential pressure set-point calculation across the LPCI heat exchanger which, eliminating conservatisms as well, determined the actual required value to prevent potential radioactive water from leaking into the CCSW system and discharging into the Kankakee River.

- An updated calculation on the containment pressure value during a design basis LOCA. This input removed some conservatism and resulted in a lower containment pressure than previously calculated.

- To address the degraded condition of the 3B LPCI heat exchanger, the licensee cleaned the component and re-performed the thermal performance test which had successful results. During the operability evaluation related to the performance test of the heat exchanger, the licensee also determined that, removing some conservatisms and utilizing the updated flow balancing model, the heat exchanger would be operable at a maximum temperature of 93 °F on average during the summer months.

The inspectors reviewed the licensee’s actions and had no further concerns. In addition, the licensee verified that the temperature during the summer months did not exceed this maximum temperature, therefore, the 3B LPCI heat exchanger remained operable during the duration of the degraded condition.

<u>Analysis</u>: The inspectors determined that failure to evaluate the effects of higher containment pressure post power up-rate on the LPCI heat exchangers’ differential pressure set-point calculation was a performance deficiency. The inspectors determined that the finding was more than minor in accordance with IMC 0612, “Power Reactor Inspection Reports,” Appendix B, “Issue Dispositioning Screening,” because it was associated with the attribute of design control, which affected the Mitigating Systems Cornerstone objective of ensuring the availability and reliability of safety-related systems. Specifically, the degraded LPCI heat exchangers would have resulted in a loss of suppression pool cooling capability when operating at design conditions.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, “Significance Determination Process,” Attachment 0609.04, “Phase 1 - Initial Screening and Characterization of findings,” Table 4a for the Mitigating System Cornerstone. The inspectors concluded the LPCI heat exchangers had been operable because river temperatures between the last two successful thermal performance tests were less than the maximum needed to support operability. The inspectors determined that the LPCI heat exchangers were in a non-conforming but operable condition; therefore, concluded the issue was of very low safety significance (Green).

The inspectors did not identify a cross-cutting aspect associated with this finding because the failure to demonstrate the ability to meet the design basis occurred several years ago and was not reflective of current performance.

<u>Enforcement</u>: Title 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” required in part, that design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program.

Contrary to the above, as of December 2001, the licensee's design control measures failed to verify the adequacy of design of the LPCI heat exchangers and the CCSW system. Specifically, the licensee failed to evaluate the effects of higher containment pressure post power up-rate in the differential pressure set-point across the heat exchangers allowing a lower than required CCSW system flow. The licensee entered the finding into the CAP as AR 763663 to correct the nonconforming conditions. Because this violation was of very low safety significance, and it was entered into the licensee's CAP, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. **(NCV 05000237/2008005–01; 05000249/2008005–01)**

1R08 Inservice Inspection (ISI) Activities 71111.08G

From November 3, 2008, through November 6, 2008, the inspectors conducted a review of the implementation of the licensee's ISI Program for monitoring degradation of the reactor coolant system (RCS), risk-significant piping, and components and containment systems.

The inspections described in Sections 1R08.1 and 1R08.2 below count as one inspection sample as defined in IP 71111.08-05.

.1 Piping Systems ISI

a. Inspection Scope

The inspectors observed the following non-destructive examinations mandated by the American Society of Mechanical Engineers (ASME), Section XI Code to evaluate compliance with the ASME Code Section XI, as well as Section V requirements, and if any indications and defects were detected, to determine if these were dispositioned in accordance with the ASME Code or an NRC approved alternative requirement.

- Ultrasonic Examination (UT) of the isolation condenser (ISCO) nozzle to shell weld (weld 12-8); and
- Magnetic Particle Examination (MT) of the ISCO nozzle to shell weld (weld 12-8).

During the prior outage non-destructive surface and volumetric examinations, the licensee did not identify any relevant/recordable indications. Therefore, no NRC review was completed for this inspection procedure attribute.

The inspectors reviewed the following pressure boundary weld completed for risk-significant systems since the beginning of the last refuelling outage to verify that the welding and any associated non-destructive examinations were performed in accordance with the Construction Code and ASME Code, Section XI.

- Weld repair of High Pressure Coolant Injection (HPCI) 1" steam line drain (Line 3-2322-1"-B).

The inspectors also reviewed the welding procedure specification and supporting weld procedure qualification records for the above, to determine if the welding procedures were qualified in accordance with the requirements of the Construction Code and the ASME Code Section IX.

b. Findings

No findings of significance were identified.

.2 Identification and Resolution of Problems

a. Inspection Scope

The inspectors performed a review of ISI related problems entered into the licensee's CAP and conducted interviews with licensee staff to determine if:

- the licensee had established an appropriate threshold for identifying ISI related problems;
- the licensee had performed a root cause (if applicable) and taken appropriate corrective actions; and
- the licensee had evaluated operating experience and industry generic issues related to ISI and pressure boundary integrity.

The inspectors performed these reviews to evaluate compliance with 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements. The corrective action documents reviewed by the inspectors are listed in the Attachment to this report. In addition, the inspectors verified that the licensee correctly assessed operating experience for applicability to the ISI group.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11Q)

.1 Resident Inspector Quarterly Review (71111.11Q)

a. Inspection Scope

The inspectors evaluated the following areas:

The quality, timeliness, and completeness of remediation requirements that were the result of individual and crew failures of dynamic requalification examinations from the previous calendar quarter.

This inspection constitutes one quarterly licensed operator requalification program sample as defined in IP 71111.11.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Routine Quarterly Evaluations

a. Inspection Scope

The inspectors evaluated degraded performance issues involving the following risk-significant systems:

- Unit 2 control rod drive; and
- Unit 2 feedwater.

The inspectors reviewed events such as where ineffective equipment maintenance had resulted in valid or invalid automatic actuations of engineered safeguards systems and independently verified the licensee's actions to address system performance or condition problems in terms of the following:

- implementing appropriate work practices;
- identifying and addressing common cause failures;
- scoping of systems in accordance with 10 CFR 50.65(b) of the maintenance rule;
- characterizing system reliability issues for performance;
- charging unavailability for performance;
- trending key parameters for condition monitoring;
- ensuring 10 CFR 50.65(a)(1) or (a)(2) classification or re-classification; and
- verifying appropriate performance criteria for structures, systems, and components (SSCs)/functions classified as (a)(2) or appropriate and adequate goals and corrective actions for systems classified as (a)(1).

The inspectors assessed performance issues with respect to the reliability, availability, and condition monitoring of the system. In addition, the inspectors verified maintenance effectiveness issues were entered into the CAP with the appropriate significance characterization. Documents reviewed are listed in the Attachment to this report.

This inspection constitutes two quarterly maintenance effectiveness samples as defined in IP 71111.12-05.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed the licensee's evaluation and management of plant risk for the maintenance and emergent work activities affecting risk-significant and safety-related equipment listed below to verify that the appropriate risk assessments were performed prior to removing equipment for work:

- Unit 3 High Pressure Coolant Injection maintenance window.

These activities were selected based on their potential risk-significance relative to the Reactor Safety Cornerstones. As applicable for each activity, the inspectors verified that risk assessments were performed as required by 10 CFR 50.65(a)(4) and were accurate and complete. When emergent work was performed, the inspectors verified that the plant risk was promptly reassessed and managed. The inspectors reviewed the scope of maintenance work, discussed the results of the assessment with the licensee's probabilistic risk analyst or shift technical advisor, and verified plant conditions were consistent with the risk assessment. The inspectors also reviewed TS requirements and walked down portions of redundant safety systems, when applicable, to verify risk analysis assumptions were valid and applicable requirements were met.

These activities constituted one sample as defined in IP 71111.13-05.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the following issues:

- Issue Report (IR) 798208, “Need Fuel Evaluation for Fe & Zn from Westinghouse/NSF;”
- IR 820657, “Isolation Condenser Failed DOS 1300-01 Surveillance on 9/10/08;” and
- IR 821228, "NRC Inspector Questions on EC [engineering change] Evaluation for Small Bore Vent Lines," discovery date 9/22/08.

The inspectors selected these potential operability issues based on the risk-significance of the associated components and systems. The inspectors evaluated the technical adequacy of the evaluations to ensure that TS operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the TS and UFSAR to the licensee’s evaluations, to determine whether the components or systems were operable. Where compensatory measures were required to maintain operability, the inspectors determined whether the measures in place would function as intended and were properly controlled. The inspectors determined, where appropriate, compliance with bounding limitations associated with the evaluations. Additionally, the inspectors also reviewed a sampling of corrective action documents to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. Documents reviewed are listed in the Attachment to this report.

This inspection constitutes three samples as defined in IP 71111.15-05.

b. Findings

(Closed) Unresolved item 05000249/2008003-03; "Corrective Actions to Prevent the Blocking of the 3B Low Pressure Coolant Injection (LPCI) Heat Exchanger Tubes by Relic Clamshells"

Introduction: A finding of very low safety significance was identified by the licensee. Specifically, the licensee failed to inject biocide into the containment cooling service water (CCSW) pumps' intake during normally scheduled operability surveillances and sample to verify biocide residual concentration. This was contrary to the licensee's Generic Letter 89-13 Program commitments.

Description: On March 20-21, 2008, thermal performance data was collected for the 3B low pressure coolant injection (LPCI) heat exchanger. Data reduction was performed and the initial analysis indicated that the 3B LPCI heat exchanger thermal performance, when extrapolated to design conditions (95 degrees F inlet water temperature), was approximately 0.6 percent below the UFSAR value (70.586 vice 71 MBtu/hr). Further evaluation determined that with a heat removal capability of 70.586 MBtu/hr the maximum allowable inlet water temperature for the next six months from the original test was 92.5 degrees F. Actual CCSW temperatures were below the design basis parameter of 92.5 degrees F, therefore the licensee determined that the 3B LPCI heat exchanger, although degraded, was able to perform the required design functions. The 3B LPCI heat exchanger was cleaned and the thermal performance testing was re-performed on June 2008. The test results indicated a heat removal capability of 73.265 MBtu/hr at design conditions which is 3.1 percent above the design heat removal rate.

The cause of the degradation was unknown, but the most probable fouling mechanism was either macro (debris that block tubes) or micro (microbial/slime that inhibit heat transfer within the tubes). Due to the degraded heat removal capability the licensee initiated a root cause investigation.

Root Cause Report (RCR) 776598-08, "Dresden 3-1503-B, 3B Low Pressure Coolant Injection (LPCI) / Containment Cooling Heat Exchanger (HX) Failure to Meet Design Basis Heat Removal Capability Due to Inadequate Programmatic Control of Macrofoulants," attributed the failure of the 3B LPCI heat exchanger to meet the design basis heat removal capability to inadequate programmatic control of macrofoulants.

On April 28, 1994, Dresden issued GFSLTR 94-0070, "Dresden Nuclear Power Station Unit 2 and 3 Supplemental Response to Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment," dated July 18, 1989." For LPCI/CCSW the letter stated in part:

> 'A biocide will be injected into the CCSW Pump Intake during normally scheduled operability surveillances. Sampling will be performed periodically at the discharge of the system (LPCI Heat Exchanger) to verify the residual biocide concentrations.'

Procedure CY-DR-120-413, "Cooling and Service Water Chemical Injection System," permits CCSW operation without requiring a biocide chemical addition for each CCSW pump operation. The failure to perform a biocide chemical addition for each CCSW operation increases the probability of introducing macrofoulants to the 3B LPCI heat exchanger via the CCSW pump suction line. The licensee failed to perform biocide chemical addition for each CCWS normally scheduled operability surveillance, as stated in the Generic Letter response. Instead, biocide chemical addition was performed randomly.

Additionally, after discussions with Chemistry personnel, the licensee determined that sampling of the CCSW residual biocide concentrations was not being performed. No sample results were identified in the Chemistry Database.

Analysis: The inspectors determined that the failure to inject biocide into the CCSW pumps' intake for each CCSW pump surveillance and the failure to sample to verify biocide residual concentration, contrary to the licensee's GL 89-13 Program commitments, was a performance deficiency. Using the guidance contained in IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," dated September 20, 2007, the inspectors determined that the finding was more than minor because it affected the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of a system that responds to initiating events to prevent undesirable consequences (i.e., core damage).

The inspectors evaluated the finding using IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," Table 4a, "Characterization Worksheet for IE, MS, and BI Cornerstones," dated January 10, 2008. The inspectors answered "No" to all the questions on Table 4a, therefore the finding screened as Green (very low safety significance). Because this finding was licensee-identified no cross-cutting aspect was identified.

Enforcement: The licensee identified a finding of very low safety significance (Green) for the failure to perform Generic Letter (GL) 89-13 commitments. Specifically, the licensee failed to inject biocide into the containment cooling service water (CCSW) pumps' intake during each normally scheduled operability surveillance and sample to verify biocide residual concentration. The failure to perform these commitments caused the blocking of the 3B low pressure coolant injection (LPCI) heat exchanger tubes by relic clamshells, which resulted in the degraded thermal performance of the heat exchanger.

The licensee determined that the 3B LPCI heat exchanger, although degraded was able to perform the required design functions. The failure to perform GL 89-13 program commitments was not an activity required by license conditions or technical specifications. Therefore, while a performance deficiency existed, no violation of regulatory requirements occurred. The licensee documented this issue in IR 805955, "NRC Generic Letter 89-13 Commitment not being performed."

This URI is closed.

1R18 Plant Modifications (71111.18)

.1 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the following temporary modifications:

- Unit 3 Temporary Ventilation Fan Procedurally Controlled No Temporary Change Package Number; and
- TCCP 372873, “Vessel Head Drain Isolation seal weld/ freeze seal.”

The inspectors compared the temporary configuration changes and associated 10 CFR 50.59 screening and evaluation information against the design basis, the UFSAR, and the TS, as applicable, to verify that the modification did not affect the operability or availability of the affected system(s). The inspectors also compared the licensee’s information to operating experience information to ensure that lessons learned from other utilities had been incorporated into the licensee’s decision to implement the temporary modification. The inspectors, as applicable, performed field verifications to ensure that the modifications were installed as directed; the modifications operated as expected; modification testing adequately demonstrated continued system operability, availability, and reliability; and that operation of the modifications did not impact the operability of any interfacing systems. Lastly, the inspectors discussed the temporary modification with operations, engineering, and training personnel to ensure that the individuals were aware of how extended operation with the temporary modification in place could impact overall plant performance.

This inspection constituted two temporary modification samples as defined in IP 71111.18-05.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed the following post-maintenance (PM) activities to verify that procedures and test activities were adequate to ensure system operability and functional capability:

- Work Order 1173169-01, “U2 CCSW Leak to DIV 1 Heat Exchanger;”
- Work Order 1176072, “U2 DG [diesel generator] Air Start Regulator Air Leak;”
- Work Order 982022-01, “D3 Refuel PM Electromatic Relief – Replace Pilot,” and 977155-02, “D3 Refuel Maintenance/surveillance on 3-203-3D Electromatic Relief Valve;”
- Work Order 01105355, “Unit 3 SRM [source range monitor] 24 is noisy compared to other 3 SRMs;” and
- Work Order 1182950-08, “Bottom Head Drain Valve Leaking.”

These activities were selected based upon the structure, system, or component's ability to impact risk. The inspectors evaluated these activities for the following (as applicable): the effect of testing on the plant had been adequately addressed; testing was adequate for the maintenance performed; acceptance criteria were clear and demonstrated operational readiness; test instrumentation was appropriate; tests were performed as written in accordance with properly reviewed and approved procedures; equipment was returned to its operational status following testing (temporary modifications or jumpers required for test performance were properly removed after test completion), and test documentation was properly evaluated. The inspectors evaluated the activities against TS, the UFSAR, 10 CFR Part 50 requirements, licensee procedures, and various NRC generic communications to ensure that the test results adequately ensured that the equipment met the licensing basis and design requirements. In addition, the inspectors reviewed corrective action documents associated with post-maintenance tests to determine whether the licensee was identifying problems and entering them in the CAP and that the problems were being corrected commensurate with their importance to safety. Documents reviewed are listed in the Attachment to this report.

This inspection constitutes five samples as defined in IP 71111.19-05.

b. Findings

Introduction: A finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” was identified by NRC inspectors for the licensee’s failure to establish a freeze seal in accordance with procedure MA-AA-736-610, “Application of Freeze Seal to All Piping,” Revision 3.

Description: On November 12, 2008, inspectors observed performance of Work Order (WO) 1182950-09, “Perform Freeze Seal with LN2 on 2" pipe to support task 08," which was to perform a freeze seal on Bottom Head Drain Line 3-0207-2. When performing MA-AA-736-610, step 4.4.7, the licensee declared a freeze plug established without meeting criterion 1, which stated, “Criterion 1 must be met, failure to meet any of the remaining criteria is not necessarily a criteria to abort the freeze. (1) Temperature probe readings are less than or equal to (minus) - 100°F and steadily trending down.” The licensee declared a freeze plug established prior to any temperature probes reading less than or equal to (minus) – 100 degrees F. The inspectors questioned the engineer at the worksite why the temperature probes were not reading (minus) -100 degrees F. The engineer stated that the plug was established based on other criteria in the procedure.

Analysis: The inspectors determined that declaring a freeze seal established prior to the required temperature probe readings was contrary to procedure MA-AA-736-610, Revision 3, “Application of Freeze Seal to All Piping,” and was a performance deficiency.

The finding was determined to be more than minor because the finding, if left uncorrected, would become a more significant safety concern. Specifically, the inspectors determined that had there been a problem with the freeze seal, there may not be adequate time to react and implement any required contingency actions. The inspectors concluded this finding was associated with the Initiating Events Cornerstone.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, “Significance Determination Process,” Attachment 0609.04, “Phase 1 - Initial Screening and Characterization of findings,” Table 3a for the Initiating Events Cornerstone. The inspectors determined that this finding affected a Primary System LOCA initiator contributor. Using IMC 0609, Appendix G, Attachment 1, Table 8 the inspectors determined this finding did not require a quantitative assessment. Therefore this finding screens as Green, very low safety significance.

This finding has a cross-cutting aspect in the area of Human Performance, H.1.b, because the licensee did not make a conservative assumption in decision making. Specifically, the licensee declared a freeze seal had been established in a manner which did not follow procedure MA-AA-736-610, Revision 3, “Application of Freeze Seal to All Piping.” Licensee personnel stated that the -100 degree F criterion was unnecessary. However, when asked why the -100 degree F criterion existed, neither station engineering nor corporate engineering could answer that question. The inspectors concluded it was non-conservative to ignore a procedural requirement without understanding why the procedural requirement exists.

Enforcement: 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings.

Contrary to the above, on November 12, 2008, the licensee failed to establish a freeze seal in accordance with procedures. Specifically, the licensee had declared a freeze seal established prior to meeting the criteria of procedure MA-AA-736-610, Revision 3, “Application of Freeze Seal to All Piping.” The licensee’s corrective action included counseling the first line supervisor and the engineer involved in the work. Because this violation was of very low safety significance and it was entered into the licensee’s CAP as IR 844263, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. **(NCV 05000249/2008005-02)**

1R20 Outage Activities (71111.20)

.1 Refueling Outage Activities

a. Inspection Scope

The inspectors reviewed the Outage Safety Plan (OSP) and contingency plans for the Unit 3 refueling outage (RFO), conducted November 3, 2008, to November 23, 2008, to confirm that the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth. During the RFO, the inspectors observed portions of the shutdown and cooldown processes and monitored licensee controls over the outage activities listed below. Documents reviewed during the inspection are listed in the Attachment to this report.

- Licensee configuration management, including maintenance of defense-in-depth commensurate with the OSP for key safety functions and compliance with the applicable TS when taking equipment out-of-service.
- Implementation of clearance activities and confirmation that tags were properly hung and equipment appropriately configured to safely support the work or testing.
- Installation and configuration of reactor coolant pressure, level, and temperature instruments to provide accurate indication, accounting for instrument error.
- Controls over the status and configuration of electrical systems to ensure that TS and OSP requirements were met, and controls over switchyard activities.
- Monitoring of decay heat removal processes, systems, and components.
- Controls to ensure that outage work was not impacting the ability of the operators to operate the spent fuel pool cooling system.
- Reactor water inventory controls including flow paths, configurations, and alternative means for inventory addition, and controls to prevent inventory loss.
- Controls over activities that could affect reactivity.
- Maintenance of secondary containment as required by TS.
- Refueling activities, including fuel handling and sipping to detect fuel assembly leakage.
- Startup and ascension to full power operation, tracking of startup prerequisites, walkdown of the drywell (primary containment) to verify that debris had not been left which could block emergency core cooling system suction strainers, and reactor physics testing.
- Licensee identification and resolution of problems related to RFO activities.

This inspection constituted one RFO sample as defined in IP 71111.20-05.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

.1 Surveillance Testing

a. Inspection Scope

The inspectors reviewed the test results for the following activities to determine whether risk-significant systems and equipment were capable of performing their intended safety function and to verify testing was conducted in accordance with applicable procedural and TS requirements:

- WO 976136, “D3 30M/Refuel TS LLRT [local leak rate test] MSIV 203-1C & 203 2C Dry Test” (isolation valve);
- WO 1170657, “D2 Qtr TS CCSW [containment cooling service water] Pump Operability Test and IST [inservice testing] Surveillance;”
- WO 99027210, “D3 15Y TS Primary Containment Integrated Leak Rate Test;”
- WO 1150620-01, “D1 1M TSTR Diesel Fire Pump Operability Surveillance;” and
- DTS-300-06, “Control Rod Drive Friction Testing” Revision 24.

The inspectors observed in plant activities and reviewed procedures and associated records to determine the following:

- did preconditioning occur;
- were the effects of the testing adequately addressed by control room personnel or engineers prior to the commencement of the testing;
- were acceptance criteria clearly stated, demonstrated operational readiness, and consistent with the system design basis;
- plant equipment calibration was correct, accurate, and properly documented;
- as-left setpoints were within required ranges; and the calibration frequencies were in accordance with TSs, the USAR, procedures, and applicable commitments;
- measuring and test equipment calibration was current;
- test equipment was used within the required range and accuracy; applicable prerequisites described in the test procedures were satisfied;
- test frequencies met TS requirements to demonstrate operability and reliability; tests were performed in accordance with the test procedures and other applicable procedures; jumpers and lifted leads were controlled and restored where used;
- test data and results were accurate, complete, within limits, and valid;
- test equipment was removed after testing;
- where applicable for inservice testing activities, testing was performed in accordance with the applicable version of Section XI, ASME code, and reference values were consistent with the system design basis;
- where applicable, test results not meeting acceptance criteria were addressed with an adequate operability evaluation or the system or component was declared inoperable;
- where applicable for safety-related instrument control surveillance tests, reference setting data were accurately incorporated in the test procedure;
- where applicable, actual conditions encountering high resistance electrical contacts were such that the intended safety function could still be accomplished;
- prior procedure changes had not provided an opportunity to identify problems encountered during the performance of the surveillance or calibration test;
- equipment was returned to a position or status required to support the performance of its safety functions; and
- all problems identified during the testing were appropriately documented and dispositioned in the CAP.

Documents reviewed are listed in the Attachment to this report.

This inspection constituted five surveillance testing samples, one containment isolation valve sample, one inservice testing sample, no RCS leakage detection samples, and three routine samples as defined in IP 71111.22, Sections -02 and -05. A RCS leakage detection surveillance was observed by the inspectors and documented in Paragraph 4OA1 under the performance indicator verification of RCS leakage.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

.1 Training Observation

a. Inspection Scope

The inspectors observed a training evolution for licensed operators on October 22, 2008, which required emergency plan implementation by a licensee operations crew. This evolution was planned to be evaluated and included in performance indicator data regarding drill and exercise performance. The inspectors observed event classification and notification activities performed by the Technical Support Center staff. The inspectors also attended the post-evolution critique for the scenario. The focus of the inspectors' activities was to note any weaknesses and deficiencies in the staff's performance and ensure that the licensee evaluators noted the same issues and entered them into the CAP. As part of the inspection, the inspectors reviewed the scenario package and other documents listed in the Attachment to this report.

This training inspection constituted one sample as defined in IP 71114.06-05.

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 reviewed licensee controls and surveys in the following radiologically significant work areas within radiation areas, high radiation areas, and airborne radioactivity areas in the plant to determine if radiological controls including surveys, postings, and barricades were acceptable:

- Unit-3 Drywell (general areas);
- Reactor Water Cleanup Valve Room;
- Unit-3 Drywell Basement and Subpile Room;
- Refuel Floor (general areas) and Scorpion Platform; and
- Unit-3 Reactor Building (various areas).

The inspectors reviewed the radiation work permits (RWPs) and work packages used to access these areas and other high radiation work areas. The inspectors assessed the work control instructions and control barriers specified by the licensee. Electronic dosimeter alarm set points for both integrated dose and dose rate were evaluated for conformity with survey indications and plant policy. The inspectors interviewed workers to verify that they were aware of the actions required if their electronic dosimeters noticeably malfunctioned or alarmed.

The inspectors walked down and surveyed (using an NRC survey meter) some of these areas to verify that the prescribed RWP, procedure, and engineering controls were in place; that licensee surveys and postings were complete and accurate; and that air samples were properly located.

These samples were credited and documented in Inspection Report 05000237/2008003; 05000249/2008003; therefore, these supplemental reviews do not represent samples.

b. Findings

No findings of significance were identified.

.2 Problem Identification (PI) and Resolution

a. Inspection Scope

The inspectors reviewed corrective action reports related to access controls and any high radiation area radiological incidents (issues that did not count as PI occurrences identified by the licensee in high radiation areas less than 1R/hr). Staff members were interviewed and corrective action documents were reviewed to verify that follow-up activities were being conducted in an effective and timely manner commensurate with their importance to safety and risk based on the following:

- initial problem identification, characterization, and tracking;
- disposition of operability/reportability issues;
- evaluation of safety significance/risk and priority for resolution;
- identification of repetitive problems;
- identification of contributing causes;
- identification and implementation of effective corrective actions;
- resolution of NCVs tracked in the corrective action system; and
- implementation/consideration of risk-significant operational experience feedback.

This sample was credited and documented in Inspection Report 05000237/2008003; 05000249/2008003; therefore, this supplemental review does not represent a sample.

b. Findings

No findings of significance were identified.

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

.1 Inspection Planning

a. Inspection Scope

The inspectors reviewed the Unit-3 refuel outage work scheduled during the inspection period and associated work activity exposure estimates for the following work activities, which were likely to result in the highest personnel collective exposures or were otherwise radiologically significant activities:

- Drywell In-Service-Inspection Activities;
- Reactor Water Cleanup (RWCU) System Maintenance Activities;
- Reactor In-Vessel Visual Inspections;
- Main Condenser Maintenance;
- Turbine and Generator Maintenance;
- Scaffold Installation/Removal Activities (Balance of Plant and Drywell);
- Reactor Disassembly, Reassembly and Related Activities;
- Digital Electro-hydraulic Control (DEHC) Modification; and
- Drywell Main Steam Safety, Electromatic and Target Rock Valve Maintenance.

This inspection constitutes one required sample as defined in IP 71121.02–5.

The inspectors reviewed documents to determine if there were site-specific trends in collective exposures and source-term measurements.

This inspection constitutes one required sample as defined in IP 71121.02–5.

The inspectors reviewed procedures associated with maintaining occupational exposures as-low-as-is-reasonably-achievable (ALARA) and processes used to estimate and track work activity specific exposures.

This inspection constitutes one required sample as defined in IP 71121.02–5.

b. Findings

No findings of significance were identified.

.2 Radiological Work Planning

a. Inspection Scope

The inspectors evaluated the licensee's list of work activities ranked by estimated exposure that were in progress and reviewed the following five work activities of highest exposure significance:

- Scaffold Activities (Balance of Plant and Drywell);
- Drywell Main Steam Safety, Electromatic and Target Rock Valve Maintenance;
- Drywell In-Service Inspection;
- Reactor Disassembly, Reassembly and Related Activities; and
- DEHC Modification.

This inspection constitutes one required sample as defined in IP 71121.02–5.

For these five activities, the inspectors reviewed the ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements in order to verify that the licensee had established procedures and engineering and work controls that were based on sound radiation protection principles in order to achieve occupational exposures that were ALARA. The inspectors also determined if the licensee had reasonably grouped the radiological work into work activities, based on historical precedence, industry norms, and/or special circumstances.

This inspection constitutes one required sample as defined in IP 71121.02–5.

The inspectors compared the results achieved (including dose rate reductions and person-rem used) with the intended dose established in the licensee's ALARA planning for these five work activities and for other selected activities. Reasons for inconsistencies between intended and actual work activity doses were reviewed, as applicable.

This sample was credited and documented in Inspection Report 05000237/2008003; 05000249/2008003; therefore, this supplemental review does not represent a sample.

The inspectors assessed the integration of ALARA requirements into work procedures and radiological work planning documents to assess whether the licensee was implementing actions in radiological job planning in order to reduce dose.

This inspection constitutes one optional sample as defined in IP 71121.02–5.

The inspectors compared the person-hour estimates, provided by maintenance planning and other groups to the radiation protection group, with the actual work activity time requirements in order to evaluate the accuracy of these time estimates.

This inspection constitutes one optional sample as defined in IP 71121.02–5.

The inspectors evaluated if the licensee's planning for radiologically significant work activities included consideration of the benefits of dose rate reduction activities, such as shielding (provided by water filled components/piping), job scheduling, and shielding and scaffolding installation and removal activities.

This inspection constitutes one optional sample as defined in IP 71121.02–5.

b. Findings

No findings of significance were identified.

.3 Verification of Dose Estimates and Exposure Tracking Systems

a. Inspection Scope

The inspectors reviewed the assumptions and bases for the Unit-3 refuel outage (D3R20) exposure estimate, including the applicable procedures, in order to evaluate the licensee's method for estimating work activity-specific exposures and the intended dose outcome. Dose rate and man-hour estimates were evaluated for reasonable accuracy.

This inspection constitutes one required sample as defined in IP 71121.02–5.

The licensee's process for adjusting exposure estimates or re-planning work (when unexpected changes in scope, emergent work, or higher than anticipated radiation levels were encountered) was evaluated. This included determining whether adjustments to estimated exposure (intended dose) were based on sound radiation protection and ALARA principles or whether they resulted from failures to adequately plan or to control the work. The frequency of these adjustments was reviewed to evaluate the adequacy of the original ALARA planning process.

This sample was credited and documented in Inspection Report 05000237/2008003; 05000249/2008003; therefore, this supplemental review does not represent a sample.

The inspectors evaluated the licensee's exposure tracking system to determine whether the level of exposure tracking detail, exposure report timeliness, and exposure report distribution was sufficient to support control of collective exposures. The inspectors reviewed radiation work permits to determine if they covered too many work activities to allow work activity specific exposure trends to be detected and controlled. During the conduct of exposure significant work, the inspectors evaluated if licensee management was aware of the exposure status of the work and if management intervened if exposure trends increased beyond exposure estimates.

This inspection constitutes one optional sample as defined in IP 71121.02–5.

b. Findings

No findings of significance were identified.

.4 Job Site Inspections and ALARA Control

a. Inspection Scope

The inspectors observed the following five jobs that were being performed in radiation areas, airborne radioactivity areas, or high radiation areas to evaluate work activities that presented the greatest radiological risk to workers:

- Reactor Bottom Head Drain Valve Repairs in Drywell;
- Recirculation System Nozzle Insulation Activities in Drywell;
- Decontamination of Drywell Basement;
- Torus Desludge Filter Removal; and
- Hydrolazing of a Reactor Building Equipment Drain Tank Line.

The inspectors reviewed the licensee's use of ALARA controls for these and other work activities. 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, that sufficient shielding of radiation sources was provided, and that the dose expended to install/remove the shielding did not exceed the dose reduction benefits afforded by the shielding.

This inspection constitutes one required sample as defined in IP 71121.02–5.

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

The inspectors attended work briefings and observed ongoing work activities to determine if workers received appropriate on-the-job supervision to ensure the ALARA requirements are met. The inspectors assessed whether the first-line job supervisor ensured that the work activity was conducted in a dose efficient manner by minimizing work crew size and by ensuring that workers were properly trained and that proper tools and equipment were available when the job started.

This inspection constitutes two optional samples as defined in IP 71121.02-5.

The inspectors reviewed exposures of individuals from various work groups that were involved in similar work activities to evaluate any significant exposure variations among workers and to determine whether any significant exposure variations were the result of worker job skill differences or whether certain workers received higher doses because of poor ALARA work practices.

This inspection constitutes one optional sample as defined in IP 71121.02–5.

b. Findings

No findings of significance were identified.

.5 Radiation Worker Performance

a. Inspection Scope

Radiation worker and radiation protection technician performance was observed during work activities being performed in radiation areas, airborne radioactivity areas, and high radiation areas that presented the greatest radiological risk to workers. The inspectors evaluated whether workers demonstrated the ALARA philosophy by being familiar with the scope of the work activity and tools to be used, by utilizing ALARA low dose waiting areas, and by complying with work activity controls. Also, radiation worker training and skill levels were reviewed to determine if they were sufficient relative to the radiological hazards and the work involved.

This inspection constitutes one required sample as defined in IP 71121.02–5.

b. Findings

No findings of significance were identified.

.6 Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed the licensee’s self-assessments, audits, and Special Reports related to the ALARA program since the last inspection to determine if the licensee’s overall audit program’s scope and frequency for all applicable areas under the Occupational Cornerstone met the requirements of 10 CFR 20.1101(c).

This inspection constitutes one required sample as defined in IP 71121.02–5.

The inspectors reviewed corrective action reports related to the ALARA program and interviewed staff members to verify that follow-up activities had been conducted in an effective and 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;
- identification and implementation of effective corrective actions;
- resolution of NCVs tracked in the corrective action system; and
- implementation/consideration of risk-significant operational experience feedback.

This inspection constitutes one optional sample as defined in IP 71121.02–5.

The inspectors reviewed the licensee's CAP to determine if repetitive deficiencies and/or significant individual deficiencies in problem identification and resolution had been addressed.

This inspection constitutes one required sample as defined in IP 71121.02–5.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification (71151-05)

.1 Mitigating Systems Performance Index (MSPI) - Cooling Water Systems

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index - Cooling Water Systems PI for Units 2 and 3 for the period from the third quarter 2007 through the third quarter 2008. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the Nuclear Energy Institute (NEI) Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5, were used. The inspectors reviewed the licensee's operator narrative logs, issue reports, MSPI derivation reports, event reports and NRC Integrated Inspection Reports for the period of July 1, 2007, through September 30, 2008, to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

The inspectors also observed the performance of a RCS leakage detection surveillance test.

This inspection constituted two MSPI cooling water system samples as defined in IP 71151-05.

b. Findings

No findings of significance were identified.

.2 Reactor Coolant System (RCS) Leakage

a. Inspection Scope

The inspectors sampled licensee submittals for the RCS Leakage PI for Units 2 and 3 for the period from December 1, 2007, to December 5, 2008. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5, were used. The inspectors reviewed the licensee's operator logs, RCS leakage tracking data, issue reports, event reports and NRC Integrated Inspection Reports for the period from October 1, 2007, to September 30, 2008, to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. The inspectors also observed the performance of a RCS leakage detection surveillance test. Documents reviewed are listed in the Attachment to this report.

This inspection constituted two RCS leakage samples as defined in IP 71151-05.

b. Findings

No findings of significance were identified.

.3 Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspectors sampled licensee submittals for the Occupational Radiological Occurrences PI for the period from the third quarter 2007 through the third quarter 2008. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in NEI Document 99–02, "Regulatory Assessment Performance Indicator Guideline," Revision 5, were used. The inspectors reviewed the licensee's assessment of the PI for occupational radiation safety to determine if indicator related data was adequately assessed and reported. To assess the adequacy of the licensee's PI data collection and analyses, the inspectors discussed with radiation protection staff, the scope and breadth of its data review, and the results of those reviews. The inspectors independently reviewed electronic dosimetry dose rate and accumulated dose alarms, dose reports and the dose assignments for any intakes that occurred during the time period reviewed to determine if there were potentially unrecognized occurrences. The inspectors also conducted walkdowns of locked high radiation area entrances to determine the adequacy of the controls in place for these areas. Documents reviewed are listed in the Attachment to this report.

This inspection constitutes one occupational radiological occurrences sample as defined in IP 71151–05.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Routine Review of items Entered Into the CAP

a. Scope

As part of the various baseline inspection procedures 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 CAP at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. Attributes reviewed included: the complete and accurate identification of the problem; that timeliness was commensurate with the safety significance; that evaluation and disposition of performance issues, generic implications, common causes, contributing factors, root causes, extent of condition reviews, and previous occurrences reviews were proper and adequate; and that the classification, prioritization, focus, and timeliness of corrective actions were commensurate with safety and sufficient to prevent recurrence of the issue. Minor issues entered into the licensee's CAP as a result of the inspectors' observations are included in the attached List of Documents Reviewed.

These routine reviews for the identification and resolution of problems did not constitute any additional inspection samples. Instead, by procedure they were considered an integral part of the inspections performed during the quarter and documented in Section 1 of this report.

b. Findings

No findings of significance were identified.

.2 Daily Corrective Action Program (CAP) Reviews

a. Scope

In order to assist with the identification of repetitive equipment failures and specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's CAP. This review was accomplished through inspection of the station's daily condition report packages.

These daily reviews were performed by procedure as part of the inspectors' daily plant status monitoring activities and, as such, did not constitute any separate inspection samples.

b. Findings

No findings of significance were identified.

.3 Semi-Annual Trend Review

a. Scope

The inspectors performed a review of the licensee's CAP and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review was focused on repetitive equipment issues, 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. Specifically, the inspectors performed a review of the licensee's corrective actions program documents related to the areas of human performance, fire protection and plant modifications. The inspectors' review nominally considered IRs that were generated in the six month period of July 2008 through December 2008, although some examples expanded beyond those dates where the scope of the trend warranted. In addition to reviewing the IR documents for trends the inspectors compared their results with issues identified in the licensee's trending reports. A sample of the licensee IRs associated with trends was reviewed for corrective action adequacy.

This review constituted a single semi-annual trend inspection sample as defined in IP 71152-05.

b. Findings

No findings of significance were identified.

The inspectors identified a trend in the area of facility changes. Specifically, the inspectors identified five examples within the six month evaluation period where plant procedures were not properly updated or revised after a facility change was performed. An example was IR 849434, "Errors Noted in DOS 0010-16," where after performing an isolation condenser heat removal capability surveillance test, the maximum design value for the open position of the 2-1301-3 Unit 2 Isolation Condenser Condensate Outlet Outboard Isolation Valve was changed. Following the performance of the isolation condenser heat removal capability surveillance test, the new maximum open position value for 2-1301-3, Unit 2 Isolation Condenser Condensate Outlet Outboard Isolation Valve should have been updated in DOS 0010-16 to reflect the modified plant conditions. In this particular example, the error was identified during the review of DOS 0010-16, Unit 2(3) Isolation Condenser Safe Shutdown Valve Operability, prior to the pre-job brief and there was no negative consequence. This example was identified by the licensee and was minor because the procedure was not used before the change occurred.

A trend was noted since the inspectors identified four more examples that were very similar to the one just described, where after changing conditions of plant equipment, the appropriate plant procedures were not updated. None of the examples identified resulted in any plant consequence. However, in aggregate they demonstrated a trend that could lead to a more significant safety concern. Four of the examples were identified by the licensee late in the year in either November or December 2008. Therefore, the inspectors determined that the licensee had not yet had time to identify this trend. Consequently, the inspectors did not consider the failure to identify this trend to be a violation of 10 CFR 50, Appendix B, Criterion XVI, “Corrective Actions.” The licensee wrote IR 860946, “NRC Identified A Potential Adverse Trend,” and assigned the performance of a Common Cause Analysis as a method to determine what corrective actions were necessary.

.4 Selected Issue Follow-Up Inspection: Inadvertent Control Rod Movement while Shutdown

a. Scope

During a review of items entered in the licensee’s CAP, the inspectors recognized a corrective action item documenting inadvertent control rod withdrawal while shutdown. Issue report (IR) 00839678, “U3 Multiple Rods Unexpectedly Withdraw During D3R20 [Dresden 3 refueling outage 20],” dated November 3, 2008, documented that three control rods had inadvertently partially withdrawn from the reactor core during the performance of a clearance order to isolate all the control rod drive mechanisms. The inspectors reviewed the circumstances surrounding the event including the procedures used, the operators’ training, the use of operating experience (OE), pre-job briefings, plant conditions, possible consequences, and licensee event notification to the NRC.

This review constituted one in-depth problem identification and resolution sample as defined in IP 71152-05.

b. Findings

Introduction: The inspectors identified two Unresolved Items relating to the inadvertent control rod movement event.

Description: On November 3, 2008, Unit 3 was in Day 1 of the D3R020 refueling outage and the operations department was performing multiple tasks to support removing systems from service. The plant was shutdown with all control rods fully inserted into the core. One of the scheduled tasks was alignment of the control rod drive system in preparation for hydro-lazing the Unit 3 scram discharge volume. Non-licensed operators (NLOs) were in the process of isolating the control rod drive (CRD) mechanisms per a clearance order that directed using Procedure DOP 0500-05, “Discharging of CRD Accumulators with Mode Switch in Shutdown or Refuel,” when a control rod drift alarm was received in the control room at 10:42 a.m. Over the next few minutes, multiple rod position indication system (RPIS) indications went to green double-dashes (- -), indicating the control rod had slightly inserted beyond the full-in position. The reactor operators notified the control room supervisor and the shift manager. Ensuing discussion between these individuals and the operations staff supervisor came to the decision that instrument maintenance individuals in the auxiliary electrical equipment room (AEER) had probably caused the interruption in the RPIS indications. The operations staff supervisor was dispatched to the AEER to determine if the work there had disrupted the RPIS indication.

Over about 17 minutes, seven control rod indications sequentially went from full-in to over-travel in. Four of the indications settled back to indicated full-in; however, three control rod indications drifted out from the full-in position (D-7 to position 06, E-7 to position 18, and E-6 to position 16). Until the three control rods drifted out, the reactor operators had not recognized that the seven affected control rods were actually moving and had not taken any action to prevent possible outward rod motion. The control room operators then entered Technical Specification (TS) 3.1.1, Condition D, Procedure DOA 0300-12, "Mispositioned Control Rod," and referenced DGA 7, "Unexpected Reactivity Addition," stopped multiple clearance orders involving the CRDs, verified no work was in progress on RPIS and notified the qualified nuclear engineer. The operators subsequently discovered the control rods had drifted due to increasing differential pressure between the CRDs and the reactor when NLOs had sequentially shut the insert riser isolation valve (101) and the withdraw riser isolation valve (102) to each CRD, isolating the related CRD. Operations department staff returned the three control rods to full-in by opening the related 101 valve until the control rod moved fully in to the over-travel position. When the related 101 valve was re-shut, each of the control rods settled to the full-in position.

Inspector interviews revealed that the control room operators were not in communication with the NLOs who were isolating CRDs and did not try to establish communication via the plant announcing system when the indications started to change; did not believe the indication that control rods were actually moving; did not take actions to prevent outward motion (SCRAM the plant or go to shutdown on the Mode switch) before the three rods started to drift out; and had not discussed the possibility of a reactivity addition or control rod motion during the pre-job briefing.

The licensee completed a Prompt Investigation Report on this event (AR 839678) and determined that industry operating experience (OE) existed specific to this event. Institute of Nuclear Power Operations (INPO) Significant Event Notice (SEN) 264, "Unplanned BWR Control Rod Withdrawals While Shutdown," dated April 10, 2007, detailed historical events at several BWRs between 1978 and 2000 where single or multiple control rods unexpectedly moved out of the core without a deliberate withdrawal signal. The reactor had become critical at two plants, one of which had the reactor vessel head removed.

The key lessons from SEN 264 were:

1. The isolation of multiple hydraulic control units (HCUs) with the control rod drive pumps in operation can cause higher-than-normal cooling and exhaust header pressures that may be a precursor to inadvertent rod motion (insert or withdraw) if a sufficient number of HCUs are not in service or if alternate system flow paths are not established.

2. Station Procedures should specify the minimum number of HCUs to be kept in service while the control rod drive pump is in service, to prevent inadvertent control rod movement when HCUs are being isolated and restored, particularly during outage conditions.

3. Reactor operators should monitor control rod drive system pressures, rod positions, and alarms during outages when the system is being manipulated to identify changing conditions that could lead to inadvertent control rod movement.

4. Personnel who operate valves to isolate and restore HCUs should be aware that their actions directly affect control rod drive system pressures that can lead to inadvertent control rod movement.

When SEN 264 was originally received at Dresden, the HCU system manager and an operations technical superintendent performed the subject matter expert review of the SEN under Action Tracking Item (ATI) 616696-04. A qualified nuclear engineer also reviewed SEN 264; however, he stated that it was unlikely that he would have reviewed any operations procedures independent of the one procedure identified in ATI 616696-04 because he is not an operations procedures expert. The licensee incorporated the SEN 264 information into the “300 Series” procedure on the control rod drive system, specifically, to monitor the cooling and exhaust header pressures every 10 HCUs after 50 HCUs had been isolated. This change was intended to alert operators to the potential increase in pressure in the CRD system so that operators could take actions to reduce pressure and avoid an unplanned control rod withdrawal event similar to SEN 264. However, the inspectors determined the licensee had not reviewed all procedures that isolated the HCUs. Specifically, the information was not entered into the “500 Series” procedures that applied to the reactor protection system.

During the performance of the clearance order to isolate the control rod drive mechanisms on November 3, 2008, the non-licensed operators were using a “500 series” procedure, Procedure DOP 500-05, “Discharging CRD Accumulators with Mode Switch in Shutdown or Refuel,” Revision 5, when the last three control rods, (D-7, E-7, and E-6) drifted out of the core to positions 06, 18, and 16. The inspectors reviewed the event and determined that more than three control rods could have moved out and that the control rods would have continued moving out continuously until the 102 valve to the related HCU was closed. Therefore, the inspectors concluded it was possible that the three (or more) control rods could have moved to full out – position 48.

The licensee analyzed the shutdown margin for the reactor for the following possible conditions:

- The actual position of the three rods at the actual temperature and xenon conditions -- the reactor was 4.5 percent subcritical.

- Three drifted rods at 48, actual temperature and xenon conditions – 3.1 percent subcritical.

- Cold conditions (actual rod positions, 68°F, and zero xenon) – 1 percent subcritical.

- Design shutdown margin (actual rod pattern plus 1 rod full out, 68°F, and zero xenon) – critical.

However, the licensee had not analyzed the shutdown margin for the three drifted rods if they were full out at cold conditions. After the inspectors requested the results of those conditions, the licensee's analysis showed that the reactor would have been critical under those conditions.

The temperature of the reactor coolant, the amount of xenon in the core, the order in which the control rod mechanisms were isolated, the pressure in the control rod drive system, and the time between when the inset valve was shut and the withdraw valve was shut were key parameters for this event. The procedure in use, DOP 500-05, did not appear to control any of these parameters. The inspectors were concerned that under different conditions the inadvertent, unplanned control rod movement could have caused the reactor to go critical. Additionally, the inspectors were concerned that the licensee had not reported the event to the NRC in accordance with 10 CFR 50.72. In discussions with the licensee on this topic, the inspectors learned that the licensee interpreted the guidance in NUREG 10-22, "Event Reporting Guidelines 10 CFR 50.72 and 50.73," to not require immediate notification because the reactor remained sub-critical at the time of discovery. Subsequent to these discussions, the licensee made a 50.72 report to the NRC on November 18, 2008. In order to resolve a difference of opinion regarding whether this event should have been promptly reported under 50.72, the inspectors planned to request assistance from the Office of Nuclear Reactor Regulation (NRR). Pending additional clarification from NRR, the inspectors considered the reporting of this event to be an URI **(URI 05000249/2008005-03)**.

Subsequent to a preliminary exit meeting on site on November 18, 2008, the inspectors received additional information regarding the circumstances and activities surrounding the unplanned control rod withdrawal event in the form of the licensee's root cause investigation report. This information was provided to the inspectors on January 7, 2009. The issues associated with this event are considered an URI **(URI 05000249/2008005-04)** pending inspector review and evaluation of the new information in the root cause report.

4OA3 Event Follow-up (71153)

.1 (Closed) LER 237/2008-003-00, "Control Room Emergency Ventilation Air Conditioning System Inoperable Due to Excessive Vibration"

On April 23, 2008, the licensee identified that the Control Room Emergency Ventilation Air Conditioning System compressor had excessive vibration. The vibration was caused by damage to several compressor pistons which was caused by a lack of lubrication. The lack of lubrication was caused by liquid refrigerant flowing back into the compressor after it shutdown. The refrigerant diluted the lubrication in the compressor. The inspectors reviewed the licensee's equipment apparent cause report and the actions associated with that report. The lubrication dilution occurred because of design deficiencies in the compressor inlet and outlet piping. The inspectors determined that there was no performance deficiency on the part of the licensee associated with the equipment failure. The inspectors reviewed the licensee's corrective actions. The inspectors had no issues with the licensee's corrective actions and determined that they were completed or had an acceptable time table for completion.

This represented one inspection sample. This LER is closed.

.2 (Closed) LER 237/2007-003-00, "Unit 2 High Pressure Coolant Injection (HPCI) System Declared Inoperable"

On July 26, 2007, the Unit 2 HPCI system was declared inoperable due to a through wall leak on the system inlet drain pot elbow. The cause of the through wall leak was liquid impingement erosion on the exterior curve at a 90 degree elbow. This line had been replaced in 1997 with A335 P11 Chrome Moly Alloy Steel to reduce the line's susceptibility to flow accelerated corrosion (FAC). Although chromium alloy steels are immune to FAC, they are still susceptible to wall loss from mechanical thinning mechanisms such as liquid impingement. The licensee replaced the piping. Non-destructive testing inspections were performed for the Unit 2 and 3 HPCI drain pot drain lines that have the highest susceptibility to impingement erosion. Piping replacement as a result of this testing was completed or planned. Other high-pressure safety piping which is not included in the FAC Program was evaluated for susceptibility. The inspectors determined that there was no performance deficiency on the part of the licensee. The inspectors reviewed the licensee's corrective actions. The inspectors had no issues with the licensee's corrective actions and determined that they were completed or had an acceptable time table for completion.

This represented one inspection sample. This LER is closed.

.3 Issue Report (IR) 833561, " U3 Reactor Building Ventilation Trip"

On December 10, 2008, the inspectors reviewed the circumstances surrounding the U3 reactor building ventilation trip event that took place on October 20, 2008. In addition to IR 833561, the inspectors reviewed IR 833555, "2/3 Reactor Building Ventilation Flow Indication," Technical Specifications and the UFSAR.

This represented one inspection sample. No findings of significance were identified.

4OA5 Other Activities

.1 Implementation of Temporary Instruction (TI) 2515/176, "Emergency Diesel Generator Technical Specification Surveillance Requirements Regarding Endurance and Margin Testing"

a. The objective of TI 2515/176 was to gather information to assess the adequacy of nuclear power plant emergency diesel generator endurance and margin testing as prescribed in plant-specific TS. The inspectors reviewed the licensee's TS, procedures, and calculations and interviewed licensee personnel to complete the TI. The information gathered for this TI was forwarded to the Office of Nuclear Reactor Regulation for further review and evaluation on December 17, 2008. This TI is complete at Dresden Nuclear Power Station; however, this TI 2515/176 will not expire until August 31, 2009. Additional information may be required after review by the Office of Nuclear Reactor Regulation.

b. Findings

No findings of significance were identified.

.2 Quarterly Resident Inspector Observations of Security Personnel and Activities

a. Inspection Scope

During the inspection period, the inspectors conducted observations of security force personnel and activities to ensure that the activities were consistent with licensee security procedures and regulatory requirements relating to nuclear plant security. These observations took place during both normal and off-normal plant working hours.

These quarterly resident inspector observations of security force personnel and activities did not constitute any additional inspection samples. Rather, they were considered an integral part of the inspectors' normal plant status review and inspection activities.

The inspectors reviewed Exelon's self-assessment of the survey results relative to its site security organization safety conscious work environment performed in 2008. The self-assessment was reviewed to determine whether any deficiencies, strengths or recommendations were identified and if any corrective actions were taken to address any of these.

b. Findings

No findings of significance were identified.

.3 (Closed) Unresolved Item (URI) 05000237/2008002–01; 05000249/2008002–01 LPCI Heat Exchangers' Design Calculation Deficiencies and Discrepancies

This item is discussed in Section 1R07 of this report. The inspectors identified an NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." This URI is closed.

4OA6 Management Meetings

.1 Exit Meeting Summary

On January 13, 2009, the inspectors presented the inspection results to Mr. D. Wozniak, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

.2 Interim Exit Meetings

Interim exits were conducted for:

- Inservice Inspection results were presented to the Plant Manager, Mr. T. Hanley, and others on November 6, 2008. The licensee acknowledged the issues presented. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

- The results of the Heat Sink Performance URI review were presented to Mr. S. Taylor and members of his staff on November 14, 2008. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

- Interim exit was conducted for the inadvertent control rod withdrawal event with Mr. T. Hanley and others on November 18, 2008. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

- Occupational Radiation Safety ALARA program inspection with Mr. T. Hanley and others on November 18, 2008. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

- A telephone exit for TI 2515/176 was conducted with Mr. R. Rybak, Regulatory Assurance, and other licensee staff on December 3, 2008. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
D. Wozniak, Site Vice President
T. Hanley, Plant Manager
K. Aleshire, Exelon Corporate Emergency Preparedness Manager
C. Barajas, Operations Director
H. Bush, Radiation Protection Manager
H. Do, Corporate ISI
B. Finely, Security Manager
D. Galanis, Design Engineering Manager
D. Glick, Shipping Specialist
G. Graff, Operations Training Manager
J. Griffin, Regulatory Assurance - NRC Coordinator
D. Gronek, Work Management Director
J. Hansen, Corporate Licensing
L. Jordan, Training Director
R. Kalb, Chemistry
P. Karaba, Maintenance Director
J. Kish, ISI Coordinator
M. Kluge, Design Engineer
D. Leggett, Nuclear Oversight Manager
R. Luburn, Radiation Protection
M. McDonald, Mechanical Maintenance
J. Miller, Corporate NDE Coordinator
T. Mohr, Maintenance Planning
M. Overstreet, Lead Radiation Protection Supervisor
G. Petrovic, Maintenance
C. Podczerwinski, Maintenance Rule Coordinator
P. Quealy, Emergency Preparedness Manager
E. Rowley, Chemistry
R. Rybak, Regulatory Assurance
J. Sipek, Engineering Director
N. Starcevich, Radiation Protection Instrumentation Coordinator
J. Strmec, Chemistry, Environmental and Radwaste Manager
S. Taylor, Regulatory Assurance Manager
S. Vercelli, Work Management

NRC
M. Ring, Chief, Division of Reactor Projects, Branch 1
A. M. Stone, Branch Chief
J. Benjamin, Reactor Engineer
G. O'Dwyer, Reactor Engineer

IEMA
R. Zuffa, Illinois Emergency Management Agency
R. Schulz, Illinois Emergency Management Agency

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened:

05000237/2008-005-01; 05000249/2008-005-01	NCV	LPCI Heat Exchangers' Design Calculation Deficiencies and Discrepancies (1R07)
05000249/2008-005-02	NCV	Freeze Seal Established Prior to Meeting the Requirements of Procedure MA-AA-736-610 (1R19)
05000249/2008-005-03	URI	Reportability for Inadvertent Rod Withdrawal
05000249/2008-005-04	URI	Inadvertent Control Rod Withdrawal

Closed:

05000237/2008-005-01; 05000249/2008-005-01	NCV	LPCI Heat Exchangers' Design Calculation Deficiencies and Discrepancies
05000249/2008-005-02	NCV	Freeze Seal Established Prior to Meeting the Requirements of Procedure MA-AA-736-610
05000237/2008-002-01; 05000249/2008-002-01	URI	LPCI Heat Exchangers' Design Calculation Deficiencies and Discrepancies
05000249/2008-003-03	URI	Corrective Actions to Prevent Blocking of the 3B Low Pressure Coolant Injection (LPCI) Heat Exchanger Tubes by Relic Clamshells
05000237/2007-003-00	LER	Unit 2 High Pressure Coolant Injection System Declared Inoperable
05000237/2008-003-00	LER	Control Room Emergency Ventilation Air Conditioning System Inoperable Due to Excessive Vibration

Discussed:

None.

LIST OF DOCUMENTS REVIEWED

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

1R01 Adverse Weather Protection (71111.01)

- IR 800442 "'A' IC Makeup Diesel Jacket Water Heater is Overheating"
- MSDS 10577, "Cat ELC Premix 50/50," revision 11
- DOA 6600-03, "Diesel Generator Keep Warm System Failure," Revision 7

1R04 Equipment Alignment (71111.04)

- DOP 1500-M1 "Unit 2 LPCI and Containment Cooling Valve Checklist" Revision 38
- DOP 1500-E1 "Unit 2 LPCI and CCSW Electrical" Revision 12

1R05 Fire Protection (71111.05)

- IR 833760, Bus 21 Top Cooling Fan Fire
- IR 849611, Oily Rags Accumulating in the IC Diesel Makeup Pump House
- Dresden Unit 2 Fire Pre-Plans

1R07 Heat Sink Performance (71111.07T)

- AR 763663; EPU Project Did Not Evaluate the Effect of Higher Overpressure; dated April 14, 2008
- EC 370130; 3B LPCI Heat Exchanger March 20, 2008 Thermal Performance Test dated March 31, 2008
- EC 371356; 3B LPCI Heat Exchanger June 26, 2008 Thermal Performance Test; dated July 3, 2008
- EC 371675; CCSW Flowrates Through the LPI/CCSW While Maintaining the Required Differential Pressure Between CCSW and LPCI Systems
- OpEv 08-003; 3B LPCI Heat Exchanger (3-1503-B); Revision 1
- OpEv 08-004; Containment Cooling Service Water (CCSW) System; Revision 005*

1R08 In-service Inspection Activities (ISI) 71111.08G

- AR00598719; HPCI Inlet Drain Pot Outlet Piping Down Stream of 3-2301-55 VL; dated March 2, 2007
- AR00686157; NDE Rejectable Rounded Indication in Pre-Weld Area; dated October 18, 2007
- AR00614057; NOS ID Loose Piping Support U-bolt; dated April 6, 2007
- AR00573308; Unit 3 Isolation Condenser Flange Leak; dated December 27, 2006
- AR00695265; RPV Bushing 81 Sticking Out of Outer Vessel Flange; dated November 6, 2007
- AR00839260; Indication Found During MT on 3A LPCI Heat Exchanger; dated November 2, 2008
- AR00598940; U3 HPCI Extent of Condition NDE from Pipe Leak; dated March 3, 2007
- AR00840261; Enhancement to GE Lighting Demonstration; dated November 3, 2008
- AR00712677; Degraded HPCI 90 Degree Elbows Identified; dated December 17, 2007

- AR00654263; U2 HPCI Inlet Drain Pot Piping Leak; dated July 26, 2007
- WO01006173; HPCI Inlet Drain Pot Outlet Piping Downstream of 3-2301-55 VL; dated March 3, 2007
- WO01059284; EP Extent of Condition for HPCI Drain Pot Line Leak; dated December 17, 2007
- WO00920390; Replace Degraded Service Water Piping; dated September 23, 2006
- WO01108194; Degraded Unit 3 CCSW Piping Identified; dated March 31, 2008
- GEH-PDI-UT-1; PDI Generic Procedure for the Ultrasonic Examination of Ferritic Pipe Welds; Revision 6
- GE-MT-100; Procedure for Magnetic Particle Examination (Dry Particle, Color Contrast or Wet Particle, Fluorescent); Revision 8
- ER-AA-335-003; Magnetic Particle Examination; Revision 3
- ER-AA-336-1008; Code Acceptance and Recording Criteria for Nondestructive(NDE) Surface Examination; Revision 1
- ER-AA-335-004; Manual Ultrasonic Measurement of Material Thickness and Interfering Conditions; Revision 3
- WPS 1-1-GTSM-PWHT; ASME Welding Procedure Specification Record Manual GTAW/SMAW; Revision 1
- A-001; Procedure Qualification Record for WPS 1-1-GTSM-PWHT; Revision 0
- A-002; Procedure Qualification Record for WPS 1-1-GTSM-PWHT; Revision 0
- 1-50C; Procedure Qualification Record for WPS 1-1-GTSM-PWHT; Revision 0

1R12 Maintenance Effectiveness (71111.12)

- IR 826528 “CRD Discharge MOV Not Stopping Flow”
- IR 844840 “2B CRD Pump Unavailability
- IR 856598, “2C RFP Inboard Seal Leak Has Degraded”
- IR 857125 “NRC Inspector Raises Concern Regarding FW Check Valve Tests”

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

- IR 837804 “NRC Inspector IDs Incorrect Statement in WC-AA-101”
- WC-AA-101, “On-Line Work Control Process,” Revision 14

1R15 Operability Evaluations (71111.15)

- BTP 08-0964, “Evaluation of changes in Dresden water chemistry conditions”
- IR 821401, “U3 Feedwater Fe Representativeness Problems”
- EC# 350134, “Dresden Nuclear Power Station – Unit 03 Implement Westinghouse Optima2 Nuclear Fuel”
- EC 371270, Rev. 0, "Qualify Small-Bore Vent Lines for CC-DR-405"
- Calc. No. D2-ISCO-01C, Minor Revision 005B, "Piping Analysis D2-ISCO-01C"
- IR 839009, “Unexpected Tech Spec Entry: RBCCW PCIV Inop”

1R18 Plant Modifications (71111.18)

- IR 832007, “NOS Ids Inadequate Procedural Control of Temporary Fan”
- IR 834940, “3A Hydrogen Cooler Leak”
- DOS 5300-01, “Unit 2(3) Hydrogen Survey,” Revision 15
- CC-AA-12, “Temporary Configuration Changes,” Revision 12

1R19 Post-Maintenance Testing (71111.19)

- EC 372445, “Calculate code and operability minimum wall thickness for CCSW line 2-1510-16”-D”
- IR 827334, “NRC ID Rust on Previous LPCI Pipe Repair”
- WO 1167193, “Op D2 1M TS Unit Diesel Generator Operability”
- IR 840969, “Weak Spring on 3D ERV”
- IR 841006, “3D ERV Pilot Valve Sticking”
- WO 1125591 - “Replace U3 SRM 24 detector”

1R22 Surveillance Testing (71111.22)

- WO 976490, “D3 30M/RFL TS LLRT MSIV 203-1B & 203-2B Dry Test”
- WO 977606, “D3 30M/RFL TS LLRT MSIV 203-1A & 203-2A Dry Test”
- WO 976132, “D3 30M/RFL TS LLRT MSIV 203-1D & 203-2D Dry Test”
- WO 975575, “D3 30M/RFL TS LLRT MSIV 203-2C Wet Test”
- WO 975576, “D3 30M/RFL TS LLRT MSIV 203-B Wet Test”
- WO 975566, “D3 30M/RFL TS LLRT MSIV 203-2A Wet Test”
- WO 975574, “D3 30M/RFL TS LLRT MSIV 203-2D Wet Test”
- IR 803315, “U1 DFP Day Tank Level Indicator Stuck at ¾”
- DOS 1500-02, “Containment Cooling Service Water Pump Test and Inservice Test (IST),” Revision 68
- IR 799916, “Quad Cities ILRT Benchmark Trip PIRS”
- DTP 47, “Leak Rate Testing Program,” Revision 17
- ER-AA-380, “Primary Containment Leakrate Testing Program,” Revision 5
- ER-AA-380-1002, “Integrated Leakage Rate Test Planning and Implementation Guide,” Revision 0
- IR 848349, “Conservative Error Found in ILRT Procedure DTS 1600-07 R. 33”
- IR 803315, “U1 DFP Day Tank Level Indicator Stuck at ¾”
- IR 856490, “Unit 3 B-04 Troubleshooting Results – Possible Channel Bow”
- DOS-300-16, “Fuel Channel Distortion Monitoring” Revision 4

2OS1 Access Control to Radiologically Significant Areas; and
2OS2 As-Low-As-Is-Reasonably-Achievable (ALARA) Planning and Controls

- RWP 10009335 and Associated ALARA Plan; D3R20 Refuel Floor In-Vessel-Inspection Activities; Revision 1
- RWP 10009261and Associated ALARA Plan; D3R20 Scaffold Activities (excluding Drywell); Revision 0
- RWP 10009293 and Associated ALARA Plan; D3R20 Drywell Main Steam Valve Maintenance Activities; Revision 0
- RWP 10009306 and Associated ALARA Plan; D3R20 Drywell In-Service Inspection Activities; Revision 0
- RWP 10009333 and Associated ALARA Plan; D3R20 Reactor Disassembly/Reassembly and Related Activities; Revision 0
- RWP 10009344 and Associated ALARA Plan; D3R20 DEHC Modification; Revision 0
- RP-AA-401; Operational ALARA Planning and Controls; Revision 9
- Focused Area Self-Assessment Report; ALARA Planning and Controls Outage Readiness and Preparation; dated September 10, 2008
- TEDE ALARA Evaluations for RWP 10009333, 10009293, 10010274 and various other D3R20 work activities; various dates

- Worker Daily Dose Reports for RWPs 10009344, 10009315, 10009306, and 10009293; dated November 4–17, 2008
- AR 00842533/00842347; Worker Briefed on Correct RWP but Logs-In on Incorrect RWP; both dated November 8, 2008
- RWP 10010274 and Associated ALARA Plan; D3R20 Bottom Head Drain Valve Repair Activities; Revision 0
- AR 00843817; RWP Outage Scaffold Used Pre-outage; dated November 3, 2008
- ALARA Work In Progress Review for RWP 10009327; D3R20 Turbine/Generator Activities; dated November 9, 2008
- ALARA Work In Progress Review for RWP 10009293; D3R20 ERV, SRV and Target Rock Activities; dated November 10, 2008
- ALARA Work In Progress Review for RWP 10009261; D3R20 Scaffold Installation/Removal Activities (Excluding Drywell); dated November 11, 2008
- ALARA Work In Progress Review for RWP 10009344; D3R20 DEHC Modification; dated November 10, 2008
- ALARA Work In Progress Review for RWP 10009306; D3R20 Drywell ISI Activities; dated November 10, 2008
- ALARA Work In Progress Review for RWP 10009286; D3R20 Drywell Scaffold Installation/Removal Activities; dated November 6, 2008
- AR 00764789; Gates to Unit-2 Torus Basement Do Not Stay Closed; dated April 16, 2008
- AR 08398926; Operator in Drywell with 45K Particle under Chin; dated November 3, 2008
- AR 00840644; Boilermaker 20K On the Side of Face; dated November 5, 2008
- AR00841054; 5K Found Around Nostril Area; November 5, 2008
- AR00842068; Level 1 PCEs For Drywell Under-vessel Workers; dated November 7, 2008
- AR 00840182; Individual Received Dose Rate Alarm; dated November 3, 2008

4OA1 Performance Indicator Verification (71151)

- LS-AA-2140, Attachment 1; Monthly Data Elements for NRC Occupational Exposure Control Effectiveness; September 2007 – October 2008
- RP-DR-4010; Electronic Dosimetry Alarm Response Form; various forms completed between September 2007–October 2008
- AR 00828068; Electronic Dosimetry Alarm During Spent Fuel Pool Filter Job; dated October 6, 2008
- AR 00818442; Individual Received Dose Rate Alarm of 1073 mrem/hr; dated September 16, 2008
- AR 00696216; Unit 2A RWCU Entry Door Defective; dated November 8, 2007
- AR 00770226; Radwaste Barreling Area Door Will not Self-Lock; dated April 30, 2008

4OA5 Other Activities

- Apparent Cause Report 654273, “Unit 2 HPCI Drain Line to Condenser Through-Wall Leak”
- IR 833555, “2/3 Rx Bldg Vent Flow Indication”
- Technical Specification 3.6.4.1, “Secondary Containment”
- UFSAR 6.2.3.2.4, “Secondary Containment Isolation System”
- UFSAR 9.4.5, “Reactor Building Ventilation System”
- DOS 6600-12; Diesel Generator Tests Endurance and Margin/Full Load Rejection/ECCS/Hot Restart; Revision 43
- Analysis No. 9389-46-19-2; Calculation for Diesel Generator 2 Loading Under Design Bases Accident Conditions; Revision 003

LIST OF ACRONYMS USED

AC	Alternating Current
AEER	Auxiliary Electrical Equipment Room
ALARA	As-Low-As-Is-Reasonably-Achievable
AR	Assignment Report
ASME	American Society of Mechanical Engineers
ATI	Action Tracking Item
CAP	Corrective Action Program
CCSW	Containment Cooling Service Water
CFR	Code of Federal Regulations
CRD	Control Rod Drive
DEHC	Digital Electro-hydraulic Control
DRP	Division of Reactor Projects
EC	Engineering Change
F	Fahrenheit
FAC	Flow Accelerated Corrosion
GL	Generic Letter
HCU	Hydraulic Control Unit
HPCI	High Pressure Coolant Injection
HX	Heat Exchanger
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IR	Inspection Report
IR	Issue Report
ISCO	Isolation Condenser
ISI	In-Service-Inspection
LER	Licensee Event Report
LLRT	Local Leak Rate test
LOCA	Loss of Coolant Accident
LPCI	Low Pressure Coolant Injection
MSPI	Mitigating Systems Performance Index
MT	Magnetic Particle Examination
NCV	Non-Cited Violation
NDE	Nondestructive Examination
NEI	Nuclear Energy Institute
NLO	Non-Licensed Operator
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
OE	Operating Experience
OSP	Outage Safety Plan
PARS	Publicly Available Records
PI	Performance Indicator or Problem Identification
PI&R	Problem Identification and Resolution
PM	Planned or Preventative Maintenance, or Post-Maintenance
RCR	Root Cause Report
RCS	Reactor Coolant System
RWCU	Reactor Water Cleanup
RFO	Refueling Outage
RPS	Reactor Protection System
RPIS	Rod Position Indication Systems

RWP	Radiation Work Permit
SDP	Significance Determination Process
SSC	Structures, Systems, and Components
TI	Temporary Instruction
TLD	Thermoluminescent Dosimeters
TS	Technical Specification
TSO	Transmission System Operator
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
UT	Ultrasonic Examination
WO	Work Order