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

July 30, 2010

EA-10-136

Mr. Michael J. Pacilio
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 AND INDEPENDENT SPENT FUEL STORAGE INSTALLATION INTEGRATED INSPECTION REPORT 05000237/2010-003; 05000249/2010-003; 07200037/2010-001

Dear Mr. Pacilio:

On June 30, 2010, 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 results of this inspection, which were discussed on July 14, 2010, with Mr. S. Marik, 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, three NRC-identified and two self-revealed findings of very low safety significance were identified. Three of the 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, the NRC is treating the issues as non-cited violations (NCVs) in accordance with Section VI.A.1 of the NRC Enforcement Policy. Additionally, a licensee-identified violation is listed in Section 4OA7 of this report.

If you contest the subject or severity of this 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 Nuclear Power Station. In addition, if you disagree with the cross-cutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region III, and the NRC Resident Inspector at the Dresden Nuclear Power Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records System (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Website 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; 72-037
License Nos. DPR-19; DPR-25

Enclosure: Inspection Report 05000237/2010-003; 05000249/2010-003; 07200037/2010-001 w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	05000237; 05000249; 07200037
License Nos:	DPR-19; DPR-25
Report No:	05000237/2010-003; 05000249/2010-003; 07200037/2010-001
Licensee:	Exelon Generation Company, LLC
Facility:	Dresden Nuclear Power Station, Units 2 and 3
Location:	Morris, IL
Dates:	April 1 through June 30, 2010
Inspectors:	C. Phillips, Senior Resident Inspector D. Meléndez Colón, Resident Inspector J. Draper, Reactor Engineer D. Sand, Reactor Engineer A. Dunlop, Senior Reactor Engineer, DRS B. Palagi, Senior Operations Engineer D. Reeser, Operations Engineer W. Slawinski, Senior Health Physicist J. Neurauter, Senior Reactor Engineer, DRS M. Learn, Lead ISFSI Inspector R. Edwards, Reactor Inspector
Approved by:	M. Ring, Chief Projects Branch 1 Division of Reactor Projects

Enclosure

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SUMMARY OF FINDINGS

IR 05000237/2010-003; 05000249/2010-003; 07200037/2010-001; 04/01/2010 -06/30/2010; Dresden Nuclear Power Station, Units 2 & 3; Heat Sink Performance, Operability Evaluations, Surveillance Testing, Identification and Resolution of Problems, and Follow-Up of Events, and Notices of Enforcement Discretion.

This report covers a three-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors, including a routine inspection by regional inspectors of operational activities associated with an Independent Spent Fuel Storage Installation. Three Green findings were identified by the inspectors and two Green findings were self-revealed. Three of these findings were considered non-cited violations (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. The inspectors identified a NCV of Technical Specification (TS) 5.4.1 for the failure to provide an adequate procedure for the verification of correct installation and restoration of equipment during an instrument maintenance surveillance test in May 2010. The licensee's corrective actions included a task to add requirements for an independent verification of the removal of volt-ohm meters (VOMs) in the ohm meter mode in 17 affected procedures. The licensee entered this finding into the corrective action program as issue report (IR) 1068559.

  Using IMC 0612, Appendix E, "Examples of Minor Violations," issued on September 20, 2007, the inspectors determined that there were no similar examples to this finding in Appendix E. The inspectors referenced IMC 0612, Appendix B, "Issue Screening," dated January 1, 2010. The inspectors determined that the finding was more than minor based on Block 9, Figure 2, paragraph 2.b, "If left uncorrected, would the finding become a more significant safety concern." The inspectors determined that the failure to perform an independent verification that a testing configuration had been returned to normal could result in the inability of a system or component to perform its function which would be a more significant safety concern. No systems had been incorrectly returned to service as a result of the inadequate procedure and, therefore, this violation had very low safety significance. This finding had a cross-cutting aspect in the area of Problem Identification and Resolution - Corrective Actions because the licensee did not address a previously identified safety issue in a timely manner. P.1(d) (Section 1R22)

- Green. On January 22, 2010, a finding of very low safety significance was self-revealed for failure to perform an adequate inspection of the grease condition of the 3-4402-C valve actuator HBC gear box, which was contrary to the requirements of MA-AA-723-301,"Periodic Inspection of Limitorque Model SMB/SB/SBD-000," Revision 3. No violation of regulatory requirements occurred because valve 3-4402-C

was a nonsafety-related component. The licensee planned to drill inspection ports into and/or replace the HBC gear boxes for valves 2/3-34403-A(B)(C)(D) and 2/3-34402-A(B)(C)(D) and 2-34402-C and change the preventive maintenance requirement to perform a 12 year mechanical inspection of the HBC gear box. This finding was placed in the licensee's corrective action program as IR 1034444, "Failure of the 3-4402-C Condenser Inlet Valve."

The finding was determined to be more than minor because the finding could be reasonably viewed as a precursor to a significant event. Specifically, valve 3-4402-C acted as an inlet in the circulating water system for the south water box. When the valve failed, it was almost completely closed. Had the valve failed open, circulating water would have been diverted from the condenser potentially causing a loss of vacuum that would have resulted in a reduction in power and/or a turbine trip and reactor trip. 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, Initiating Events Cornerstone Column, Transient Initiators question 1, does the finding contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available, was answered "no," and, therefore, screened as Green. This finding had no cross-cutting aspect due to the issues involved in this valve failure were not indicative of current performance. (Section 4OA2.6)

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 (Green) related to whether the equipment in the low pressure coolant injection (LPCI) corner rooms would remain within environmental qualification limits during a design basis loss-of-coolant-accident. Specifically, the licensee's heat-up calculation contained discrepancies and failed to evaluate the worst case effects post-extended power uprate when determining the heat-up in the LPCI corner rooms. This finding was entered into the licensee's corrective action program as AR00763663, AR00742158, AR00883207, AR01055863, and AR01060243, and an operability call performed by the licensee concluded that there are sufficient conservatisms in the calculation that equipment in the corner rooms remained operable.

  The finding was more than minor 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. This finding is of very low safety significance (Green) because the design deficiency was confirmed not to result in loss of operability or functionality. The inspectors did not identify a cross-cutting aspect associated with this finding because this was a legacy design issue and, therefore, was not reflective of current performance. (1R07.2.b.)

- Green. A finding of very low safety significance was identified by the inspectors on May 3, 2010, for the failure to monitor Unit 3 (U3) drywell temperature per commitments made to the NRC within LER 88-22, Supplement 2 to ensure that equipment in the drywell was operating within environmental qualification limits. No violation of regulatory requirements occurred. The licensee's corrective actions were to reinstate the temperature monitoring of both drywells, perform walkdowns of both drywells for correct placement and to verify functionality of drywell thermocouples, improve administrative

requirements on plant engineering turnover from one engineer to another, and train engineers to focus on identifying and validating assumptions when performing or reviewing technical products. This finding was placed in the licensee's corrective action program as IR 1064681.

The finding was determined to be more than minor because the finding was associated with the Mitigating Systems Cornerstone attribute of Equipment Performance and affected the cornerstone objective of ensuring the reliability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to monitor drywell temperatures and evaluate those temperatures against environmental qualification limits would have resulted in the motor operator for valve 3-1301-1 to exceed its qualified life without the licensee's knowledge after D3R21. 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 Systems Cornerstone. The inspectors were able to answer the first question, is the finding a design or qualification deficiency confirmed not to result in loss of operability or functionality, "yes" and, therefore, the finding screened as Green. The inspectors did not identify a cross-cutting aspect associated with this finding because the age of the finding did not reflect current performance. (Section 1R15)

- Green. A finding of very low safety significance (Green) and associated NCV of Title 10 of the Code of Federal Regulations, Part 50, Appendix B, Criterion XI, "Test Control," was self-revealed on November 16, 2009, for the failure to perform adequate post-modification testing which allowed an improper wiring condition to exist for years before an inappropriate action by a nuclear station operator exposed the problem. The licensee generated issue report 994101 to address this issue. Corrective actions implemented by the licensee to address this issue included correcting the improper wiring configuration and creating a training request to have enhanced post-maintenance testing requirement training given to all plant engineers, design engineers and maintenance work planners.

  Using IMC 0612, Appendix B, "Issue Screening," the inspectors determined that the finding was more than minor because it impacted the Mitigating Systems Cornerstone attribute of procedure quality to ensure the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors completed a Phase 1 significance determination of this issue using IMC 0609, Attachment 0609.04, "Phase 1 – Initial Screening and Characterization of Findings." The inspectors answered "no" to all questions in the Mitigating Systems Cornerstone column of Table 4a, "Characterization Worksheet for IE, MS, and BI Cornerstones." Therefore, the finding screened as Green (very low safety significance). Because the behavior leading to this event occurred in 1986, the inspectors determined that this event was not indicative of current performance and, therefore, no cross-cutting area was affected. (Section 4OA3.1)

B. Licensee-Identified Violations

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

REPORT DETAILS

Summary of Plant Status

Unit 2

On May 26, 2010, power was reduced to approximately 97 percent electrical output for an environmental derate due to high intake temperatures. The unit returned to full power the same day.

On May 29, 2010, power was reduced to approximately 68 percent electrical output for a control rod sequence change, control rod drive scram timing and turbine valve testing. The unit returned to full power on May 30, 2010.

Unit 3

On May 22, 2010, power was reduced to approximately 59 percent electrical output for a control rod sequence change, control rod drive scram timing and turbine valve testing. The unit returned to full power on May 24, 2010.

On May 25, 2010, power was reduced to approximately 92 percent electrical output for an environmental derate due to high intake temperatures. The unit returned to full power on May 29, 2010.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 Readiness of Offsite and Alternate AC Power Systems

a. Inspection Scope

The inspectors verified that plant features and procedures for operation and continued availability of offsite and alternate alternating current (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 this report. The inspectors also reviewed corrective action program (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 constituted 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.

1R04 Equipment Alignment (71111.04)

.1 Quarterly Partial System Walkdowns

a. Inspection Scope

The inspectors performed partial system walkdowns of the following risk-significant systems:

- Unit 3 LPCI/CCSW [low pressure coolant injection/containment cooling service water] Division 2 when Unit 3 LPCI/CCSW Division 1 was out-of-service on April 13, 2010;
- Unit 3 LPCI/CCSW Division 1 when Unit 3 LPCI/CCSW Division 2 was out-of-service on May 24,2010; and
- Unit 2 HPCI [high pressure coolant injection].

The inspectors selected these systems based on their risk significance relative to the Reactor Safety Cornerstones at the time they were inspected. 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, Updated Final Safety Analysis Report (UFSAR), Technical Specification (TS) requirements, outstanding work orders (WOs), condition reports, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the systems incapable of performing their intended

functions. The inspectors also walked down accessible portions of the systems 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 three partial system walkdown samples as defined in IP 71111.04-05. Also, additional activities were performed during this system walkdown that were associated with TI 2515/177, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems." These activities are described in bullet .2 of this section.

b. Findings

No findings of significance were identified.

.2 System Walkdown Associated with Temporary Instruction (TI) 2515/177, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal and Containment Spray Systems (NRC Generic Letter 2008-01)"

a. Inspection Scope and Documentation

On June 16, 2010, the inspectors conducted a walkdown of Unit 2 High Pressure Coolant Injection (HPCI) System in sufficient detail to reasonably assure the acceptability of the licensee's walkdowns (TI 2515/177, Section 04.02.d). The inspectors also verified that the information obtained during the licensee's walkdown was consistent with the items identified during the inspector's independent walkdown (TI 2515/177, Section 04.02.c.3).

The inspectors verified that Piping and Instrumentation Diagrams (P&IDs) accurately described the subject systems, that they were up-to-date with respect to recent hardware changes, and any discrepancies between as-built configurations and the P&IDs were documented and entered into the CAP for resolution (TI 2515/177, Section 04.02.b).

In addition, the inspectors reviewed the licensee's isometric drawings that describe the HPCI system configurations to verify that the licensee had acceptably confirmed the accuracy of the drawings (TI 2515/177, Section 04.02.a). The inspectors considered the following related to the isometric drawings:

- High point vents were identified;
- High points that do not have vents were acceptably recognizable;
- Other areas where gas can accumulate and potentially impact subject system operability, such as at orifices in horizontal pipes, isolated branch lines, heat exchangers, improperly sloped piping, and under closed valves, were acceptably described in the drawings or in referenced documentation;
- Horizontal pipe centerline elevation deviations and pipe slopes in nominally horizontal lines that exceed specified criteria were identified;

- All pipes and fittings were clearly shown;
- The drawings were up-to-date with respect to recent hardware changes and that any discrepancies between as-built configurations and the drawings were documented and entered into the CAP for resolution.

The licensee indicated that, even though they possess isometric drawings of the HPCI system, they do not rely upon any isometric drawings for gas management in that system. Therefore, the inspectors were unable to verify the above consideration.

Documents reviewed are listed in the Attachment to this report.

This inspection effort counts towards the completion of TI 2515/177, which will be closed on a later inspection report.

b. Findings

Introduction: The inspectors identified an unresolved item (URI) regarding the impact of HPCI booster pump bearing oil levels being outside the acceptable band on the operability of the HPCI pump.

Description: During a walkdown of the Unit 2 HPCI system, the inspectors identified that the preferred level band indicators on the sightglasses for the HPCI booster pump inboard and outboard bearing oil reservoirs were not reliable since the indicators consisted of wires that could be slid up and down the sightglass. When informed by the inspectors, the licensee checked the indicators and identified that they do not conform to the type and position of the markings described in the HPCI booster pump maintenance procedure (MA-AB-734-448). The maintenance procedure defined the oil level band as ½" between the high and low levels. Since the sightglasses on all bearings in the field had a band of approximately 1", they were all non-conservative in at least one direction. Upon further questioning by the inspectors, the licensee measured the oil levels in each of the HPCI booster pump bearing oil level sightglasses for both Units 2 and 3. From these measurements, the inspectors identified that one oil level, the Unit 3 outboard bearing oil level, was below the low level described in the maintenance procedure. When the inspectors asked the licensee about the impact on operability of the pump with this oil level, the licensee indicated that they do not consider the oil band contained in the maintenance procedure as defining the operability of the pump, and they determined that the oil level was not low enough to consider the pump inoperable. When asked at what levels the licensee would consider the pump inoperable, licensee staff were unable to provide an answer.

Further information is required from the licensee to determine whether the inaccurate oil level bands caused the oil for the bearings to be at levels such that the HPCI pump was inoperable. This is considered an URI pending further NRC review.
(URI 05000237/2010003-01; 05000249/2010003-01)

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 11.2.1, Unit 2 southwest corner room, elevation 476';
- Fire Zone 11.2.2, Unit 2 southeast corner room, elevation 476';
- Fire Zone 9.0B, Unit 3 emergency diesel generator room, elevation 517';
- Fire Zone 7.0B, Unit 3 battery room, elevation 551'.

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 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 to this report, 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.

1R06 Flooding (71111.06)

.1 Internal Flooding

a. Inspection Scope

The inspectors reviewed selected risk-important plant design features and licensee procedures intended to protect the plant and its safety-related equipment from internal flooding events. The inspectors reviewed flood analyses and design documents, including the UFSAR, engineering calculations, and abnormal operating procedures to identify licensee commitments. The specific documents reviewed are listed in the Attachment to this report. In addition, the inspectors reviewed licensee drawings to

identify areas and equipment that may be affected by internal flooding caused by the failure or misalignment of nearby sources of water, such as the fire suppression or the circulating water systems. The inspectors also reviewed the licensee's corrective action documents with respect to past flood-related items identified in the corrective action program to verify the adequacy of the corrective actions. The inspectors performed a walkdown of the following plant area to assess the adequacy of watertight doors and verify drains and sumps were clear of debris and were operable, and that the licensee complied with its commitments:

- Unit 3 containment cooling service water vault.

This inspection constituted one internal flooding sample as defined in IP 71111.06-05.

b. Findings

No findings of significance were identified.

.2 Underground Vaults

a. Inspection Scope

The inspectors selected underground bunkers/manholes subject to flooding that contained cables whose failure could disable risk-significant equipment. The inspectors reviewed the licensee's corrective action documents with respect to past submerged cable issues identified in the corrective action program to verify the adequacy of the corrective actions. The licensee performed first time inspections of underground cable vaults in response to a violation written in Inspection Report 05000237/2010-002; 05000249/2010-002 for having a submerged low voltage Unit 3 diesel generator cooling water pump cable. The inspectors verified that the licensee inspected all submerged vaults and addressed the conditions identified. The inspectors performed a walkdown of the following underground bunkers/manholes subject to flooding:

- MH-2 (manhole 2);
- MH-1 SBO (manhole 1 Station Blackout);
- MH-2 SBO (manhole 2 Station Blackout).

This inspection constituted one underground vaults sample as defined in IP 71111.06-05.

b. Findings

No findings of significance were identified.

1R07 Annual Heat Sink Performance (71111.07Q and T)

.1 Heat Sink Performance

a. Inspection Scope

The inspectors reviewed the licensee's testing of the Unit 2 turbine building closed cooling water heat exchangers to verify that potential deficiencies did not mask the licensee's ability to detect degraded performance, to identify any common cause issues

that had the potential to increase risk, and to ensure that the licensee was adequately addressing problems that could result in initiating events that would cause an increase in risk. The inspectors reviewed the licensee's observations as compared against acceptance criteria, the correlation of scheduled testing and the frequency of testing, and the impact of instrument inaccuracies on test results. Inspectors also verified that test acceptance criteria considered differences between test conditions, design conditions, and testing conditions. Documents reviewed for this inspection are listed in the Attachment to this report.

This annual heat sink performance inspection constituted one sample as defined in IP 71111.07-05.

b. Findings

No findings of significance were identified.

.2 Triennial Heat Sink Performance

a. Inspection Scope

Unresolved Item (URI) 5000237/2008002–02; 05000249/2008002–02, was opened during the 2008 triennial heat sink performance inspection due to deficiencies and discrepancies in the low pressure coolant injection (LPCI) corner room heat-up design calculation. During this inspection period, the inspectors reviewed the revised LPCI room heat-up design calculation and related documents to verify the post-extended power uprate (EPU) temperatures in the LPCI Room after a design basis loss-of-coolant-accident (LOCA) without LPCI room cooling were within the equipment's environmental qualification limits. This URI is closed under Section 4OA5.3 of this report. This review did not represent an inspection sample.

b. Findings

LPCI Room Heat-up Calculation Deficiencies

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 whether the equipment in the LPCI rooms would remain within environmental qualification (EQ) limits during a design basis LOCA. Specifically, the licensee's heat-up calculation contained discrepancies and failed to evaluate the worst case effects post-EPU when determining the heat-up in the LPCI corner rooms.

Description: During the 2008 triennial heat sink performance inspection, the inspector identified a number of issues with the calculations used to verify that the temperature in the LPCI corner rooms after a design basis LOCA remained within the EQ limits of the equipment after implementation of EPU at Dresden. These issues, which were documented in Inspection Report 05000237/2008002; 05000249/2008002, Section 1R07.1.b.(2), included incorporating increased containment pressure post-EPU (AR00763663), ensuring EQ binders were updated to reflect increased post-EPU temperatures and correctly documenting reference to 22 degrees Fahrenheit (°F) temperature increase (AR00742158). These issues were adequately resolved during this inspection. The issue concerning increased heat from the motor due to

increased ambient temperatures was verified not to be a significant source that needed to be included in the calculation.

An additional inspector concern identified during the inspection of this URI was the use of a non-conservative analytical method used for determining EQ temperatures in the corner rooms. Basically, the methodology did not take into account the heat input coming from the reactor building due to the high, post-LOCA temperatures in the drywell. This concern was documented in AR00883207. Based on the inspectors concerns, the licensee revised calculation BSA-D-00-01, "Dresden 2/3 ECCS Room Temperature Response with Loss of Room Cooler," Revision 1A. Based on this change, the projected LPCI room temperatures went from 182.4°F to 185.2°F. This temperature was still within the rooms EQ limit of 189.4°F.

Subsequent to this revision, the licensee identified a non-conservative heat load used in calculation BSA-D-00-01. The heat load for the room cooler fan motors was assumed to be 2793 BTU/hour based on calculation DR-721-M-001, "Maximum Room Temperature for the Corner Pump Rooms Following a Postulated Event Outside these Rooms," Revision 0. However, a later revision to this calculation increased the fan motor heat load to 15,518 BTU/hour, an increase of 12,725 BTU/hour. This change was not incorporated into calculation BSA-D-00-01 when calculation DR-721-M-001 was revised. Preliminary review of this issue by the licensee determined that the corrected peak room temperature would increase to 186.2°F, which still remained within the EQ limit for the corner room.

While resolving previous inspector concerns with the licensee's scenario that having offsite power available during the event would result in the worst case temperature in the corner rooms and the heat loads assumed from the LPCI and core spray pumps, the inspector identified additional issues involving the modeled scenario. With offsite power available, station procedures would allow the operators to continue to run the second LPCI pump after 10 minutes. The licensee calculation shut off the second LPCI pump in the analysis, which removed a significant heat load (134,041 BTU/hour). Based on a simplistic calculation using this additional heat load and using the corner room temperature increase identified by the licensee in the previous paragraph, this change could result in an approximate 9°F increase in the corner room peak temperature, which would exceed the EQ limit for the corner room. As a result, the inspector had a concern with the operability of the equipment in the corner rooms to function post-LOCA based on potentially exceeding the equipment's EQ limit. The licensee initiated AR01055863 and AR01060243 to address the inspector's concern.

The corner room temperature rise discussed in the previous paragraph would not be as significant because having the second LPCI pump operating would increase containment cooling and reduce the torus water temperature. This would result in a lower heat load from the drywell to the reactor building and a lower heat load from the LPCI piping in the corner room, which takes suction from the torus. Although the magnitude of these affects on having the second pump operating could not be quantified until the analysis is updated, it did provide part of the basis that the temperature in the corner rooms would remain within the EQ limits. The licensee provided additional conservatisms within the calculations to support their operability call. These included the following: (1) heat exchanger K value (262 BTU/sec-°F) used as a design input in the calculation BSA-D-00-01 was based on Quad Cities, while the Dresden heat exchanger K value was 7 percent larger; and (2) the torus temperature profile used had a peak

value of 201.6°F, while the EPU Containment System Response analysis determined a peak temperature of 184.6°F. Inclusion of these and other conservatisms provide additional margin such that the temperature in the LPCI corner rooms would remain within their EQ limits. The inspector did not have any additional operability concerns.

The licensee was also reviewing a generic operating experience issue that was issued for an error identified in the GOTHIC code, which was the model used by the licensee to determine the heat-up in the reactor building. The licensee initiated AR01058534 to address this new issue. The licensee intends to incorporate all the necessary changes into the model and revise the calculation of record for the temperature rise in the LPCI corner rooms after a design basis LOCA as part of their corrective action to this issue.

Analysis: The inspectors determined that the failure to evaluate the effects of higher containment pressure/temperature post-EPU, ensuring design inputs were conservative, and ensuring the scenario was worst case when determining the peak corner room temperature 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 equipment in the LPCI room could have been exposed to temperatures above their EQ limit during a design basis LOCA that could have resulted in a loss of equipment necessary to mitigate the accident.

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 Systems Cornerstone. The inspectors concluded the LPCI pumps and other equipment within the corner rooms remained operable because the conservatisms in the calculation would offset the deficiencies identified with the analysis such that the corner room temperature would not exceed the EQ limit. The inspectors agreed with the licensee's operability call and concluded the issue was of very low safety significance (Green).

The inspectors did not identify a cross-cutting aspect associated with this finding because this was a legacy design issue and, therefore, was not reflective of current performance.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," in part, that measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions.

Contrary to the above, as of December 2001, the license failed to correctly translate applicable design basis into specifications. Specifically, the licensee's analysis failed to evaluate the effects of higher pressure/temperature post-EPU, ensuring design inputs were conservative, and ensuring the scenario was worst case when determining the peak corner room temperature remained within the LPCI corner room equipments' established EQ limits. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program as AR00763663, AR00742158, AR00883207, AR01055863, and AR01060243, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy.

(**NCV 05000237/2010003–02; 05000249/2010003–02**; LPCI Room Heat-Up Calculation Deficiencies).

1R11 Licensed Operator Requalification Program (71111.11Q and B)

Completion of Sections .2 through .10 constituted one biennial licensed operator requalification inspection sample as defined in IP 71111.11B.

.1 Resident Inspector Quarterly Review (71111.11Q)

a. Inspection Scope

On April 12, 2010, the inspectors observed a crew of licensed operators in the plant's simulator during licensed operator requalification examinations to verify that operator performance was adequate, evaluators were identifying and documenting crew performance problems, and training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

- licensed operator performance;
- crew's clarity and formality of communications;
- ability to take timely actions in the conservative direction;
- prioritization, interpretation, and verification of annunciator alarms;
- correct use and implementation of abnormal and emergency procedures;
- control board manipulations;
- oversight and direction from supervisors; and
- ability to identify and implement appropriate TS actions and Emergency Plan actions and notifications.

The crew's performance in these areas was compared to pre-established operator action expectations and successful critical task completion requirements. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings of significance were identified.

.2 Facility Operating History (71111.11B)

a. Inspection Scope

The inspectors reviewed the plant's operating history from January 1, 2009, through April 30, 2010, to identify operating experience that was expected to be addressed by the Licensed Operator Requalification Training (LORT) program. The inspector verified that the identified operating experience had been addressed by the facility licensee in accordance with the station's approved Systems Approach to Training (SAT) program to satisfy the requirements of 10 CFR 55.59(c). The documents reviewed during this inspection are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

.3 Licensee Requalification Examinations

a. Inspection Scope

The inspectors performed an inspection of the licensee's LORT test/examination program for compliance with the station's SAT program which would satisfy the requirements of 10 CFR 55.59(c)(4). The reviewed operating examination material consisted of two operating tests, each containing two dynamic simulator scenarios and six job performance measures (JPMs). The written examinations reviewed consisted of two written examinations, consisting of 35 questions for each examination. The inspectors reviewed the annual requalification operating test and biennial written examination material to evaluate general quality, construction, and difficulty level. The inspectors assessed the level of examination duplication from week-to-week for the operating test administered in 2010 and written examination material administered in 2009. The inspectors reviewed the methodology for developing the examinations, including the LORT program 2-year sample plan, probabilistic risk assessment insights, previously identified operator performance deficiencies, and plant modifications. The documents reviewed during this inspection are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

.4 Licensee Administration of Requalification Examinations

a. Inspection Scope

The inspectors observed the administration of a requalification operating test to assess the licensee's effectiveness in conducting the test to ensure compliance with 10 CFR 55.59(c)(4). The inspectors evaluated the performance of two operating crews in parallel with the facility evaluators during two dynamic simulator scenarios and evaluated various licensed crew members concurrently with facility evaluators during the administration of several JPMs. The inspectors assessed the facility evaluators' ability to determine adequate crew and individual performance using objective, measurable standards. The inspectors observed the training staff personnel administer the operating test, including conducting pre-examination briefings, evaluations of operator performance, and individual and crew evaluations upon completion of the operating test. The inspectors evaluated the ability of the simulator to support the examinations. A specific evaluation of simulator performance was conducted and documented in the section below titled, "Conformance with Simulator Requirements Specified in 10 CFR 55.46." The documents reviewed during this inspection are listed in the Attachment to this report.

b. Findings

Introduction: During the conduct of one of the dynamic simulator scenarios, the inspectors identified an unresolved item related to procedure implementation.

Description: During one of the dynamic simulator scenarios, conditions were simulated during an Anticipated Transient Without Scram (ATWS) that required the operating crew to lower reactor pressure vessel (RPV) water level in accordance with emergency operating procedure DEOP 400-5, "Failure to SCRAM." The purpose of lowering RPV water level is to reduce core inlet sub-cooling and thus reduce the potential for power oscillations. DEOP 400-5, directs the operators to "Terminate and Prevent" all injection flow into the RPV except for flow from the CRD and Standby Liquid Control (Boron) systems. Contrary to the BWR Owners' Group (BWROG) Emergency Procedure Guidelines (EPG) and Severe Accident Guidelines [Revision 2] – which states that failure to completely stop RPV injection flow (with the exception of CRD, RCIC, and Standby Liquid Control) would delay the reduction in core inlet sub-cooling, thus increasing the potential for flux oscillations – the crew was observed to implement this step in accordance with the licensee's expectations, by decreasing the FWLC SETPOINT to -40 inches in incremental steps, such that the set point was always less than actual level. Using this method, feedwater flow was not actually stopped but level was dropped to -35 inches in approximately 1.5 minutes. When asked why the licensee's procedural steps deviated from the BWROG EPG, the licensee stated that the deviation was necessary to prevent the loss of the Main Condenser heat sink (bypassing the Group 1 Isolation interlocks is performed in parallel and cannot be completed quick enough to prevent isolation of the Main Steam lines if flow is terminated completely). The BWROG EPG states that reducing reactor power and preventing power oscillations is of greater importance than preventing loss of the main condenser.

Technical Specification 5.4.1 requires, in part, that written procedures/instructions be established, implemented, and maintained covering the emergency operating procedures required to implement the requirements of NUREG-0737, "Clarification of TMI Action Plan Requirements," and NUREG-0737, Supplement 1, as stated in GenericLetter 82-33. NUREG-0737 and the associated Supplement 1 requires licensees to analyze transients and accidents, prepare emergency procedure technical guidelines, and develop symptom-based emergency operating procedures based on those technical guidelines. The BWROG EPG provides the technical basis for the development of the emergency operating procedures used by BWR licensees. Licensees are permitted to deviate from the BWROG guidelines provided they document the technical basis for the deviation. When asked to provide justification for the deviation from the BWROG EPG, the licensee was unable to do so. The licensee has initiated an engineering evaluation to provide the necessary basis for the deviation.

This issue is an URI pending further NRC review and completion of the licensee's actions to provide the necessary documentation to support the deviation: (**URI 05000237/2010003-03; 05000249/2010003-03**, Undocumented Technical Basis for Change to EOP ATWS Mitigation Strategy)

.5 Examination Security

a. Inspection Scope

The inspectors observed and reviewed the licensee's overall licensed operator requalification examination security program related to examination physical security (e.g., access restrictions and simulator considerations) and integrity (e.g., predictability and bias) to verify compliance with 10 CFR 55.49, "Integrity of Examinations and Tests." The inspectors also reviewed the facility licensee's examination security procedure and

the implementation of security and integrity measures (e.g., security agreements, sampling criteria, bank use, and test item repetition) throughout the examination process. The documents reviewed during this inspection are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

.6 Licensee Training Feedback System

a. Inspection Scope

The inspectors assessed the methods and effectiveness of the licensee's processes for revising and maintaining its LORT Program up-to-date, including the use of feedback from plant events and industry experience information. The inspectors reviewed the licensee's quality assurance oversight activities, including licensee training department self-assessment reports. The inspectors evaluated the licensee's ability to assess the effectiveness of its LORT program and their ability to implement appropriate corrective actions. This evaluation was performed to verify compliance with 10 CFR 55.59(c) and the licensee's SAT program. The documents reviewed during this inspection are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

.7 Licensee Remedial Training Program

a. Inspection Scope

The inspectors assessed the adequacy and effectiveness of the remedial training conducted since the previous biennial requalification examinations and the training from the current examination cycle to ensure that they addressed weaknesses in licensed operator or crew performance identified during training and plant operations. The inspectors reviewed remedial training procedures and individual remedial training plans. This evaluation was performed in accordance with 10 CFR 55.59(c) and with respect to the licensee's SAT program. The documents reviewed during this inspection are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

.8 Conformance with Operator License Conditions

a. Inspection Scope

The inspectors reviewed the facility and individual operator licensees' conformance with the requirements of 10 CFR Part 55. The inspectors reviewed the facility licensee's program for maintaining active operator licenses and to assess compliance with 10 CFR 55.53(e) and (f). The inspectors reviewed the procedural guidance and the process for tracking on-shift hours for licensed operators and which control room

positions were granted watch-standing credit for maintaining active operator licenses. The inspectors reviewed the facility licensee's LORT program to assess compliance with the requalification program requirements as described by 10 CFR 55.59(c). Additionally, medical records for 10 licensed operators were reviewed for compliance with 10 CFR 55.53(l). The documents reviewed during this inspection are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

.9 Conformance with Simulator Requirements Specified in 10 CFR 55.46

a. Inspection Scope

The inspectors assessed the adequacy of the licensee's simulation facility (simulator) for use in operator licensing examinations and for satisfying experience requirements as prescribed in 10 CFR 55.46, "Simulation Facilities." The inspectors also reviewed a sample of simulator performance test records (i.e., transient tests, malfunction tests, steady state tests, and core performance tests), simulator discrepancies, and the process for ensuring continued assurance of simulator fidelity in accordance with 10 CFR 55.46. The inspectors reviewed and evaluated the discrepancy process to ensure that simulator fidelity was maintained. Open simulator discrepancies were reviewed for importance relative to the impact on 10 CFR 55.45 and 55.59 operator actions as well as on nuclear and thermal hydraulic operating characteristics. The inspectors conducted interviews with members of the licensee's simulator staff about the configuration control process and completed the IP 71111.11, Appendix C checklist to evaluate whether or not the licensee's plant-referenced simulator was operating adequately as required by 10 CFR 55.46(c) and (d). The documents reviewed during this inspection are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

.10 Annual Operating Test Results (71111.11B)

a. Inspection Scope

The inspectors reviewed the overall pass/fail results of the biennial written examination administered in 2009, and the individual Job Performance Measure operating tests, and the simulator operating tests (required to be given per 10 CFR 55.59(a)(2)) administered by the licensee in May and June 2010, as part of the licensee's operator licensing requalification cycle. These results were compared to the thresholds established in IMC 609, Appendix I, "Licensed Operator Requalification Significance Determination Process (SDP)." The evaluations were also performed to determine if the licensee effectively implemented operator requalification guidelines established in NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," and IP 71111.11, "Licensed Operator Requalification Program." The documents reviewed during this inspection are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

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

- Unit 3 125V DC electrical; and
- U3 250V DC electrical.

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 that 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 constituted 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 isolation condenser unavailable same time as Unit 3 station black out diesel generator;
- Work Order 01125716-01, “D3 24M TS DIV I LPCI INJ Cont Spray, Torus & 2/3 Core Cov;”
- Unit 3 low pressure coolant injection/containment cooling water, Division 1; and
- Unit 2 125V DC battery charger out-of-service.

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 maintenance risk assessments and emergent work control activities constituted four samples 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:

- Op-Eval 10-002, “U2 and U3 HPCI [high pressure coolant injection] Lube Oil Cooler and Gland Seal Condenser Post LOCA [loss of coolant accident] Flow,” Revision 0;
- IR 1062334, “LPCI [low pressure coolant injection] 3-1501-38A Valve Moves Without Operator Input;”
- IR 1057049, “Air Void U2 HPCI Discharge Piping Above Acceptance Criteria;”
- IR 1058558, “Air Ribbon Discovered in 3B Core Spray Suction;”
- IR 1069168, “MOV 3-1301-01 EQ [environmental qualification] Life Affected by High DW [drywell] Temperature;”
- EC 379781, “Evaluation of air identified in U2 high pressure core injection (HPCI) discharge line;” and
- IR 1024796, “Adverse Temperature Trend in U2 DW 4th floor.”

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 Updated Safety Analysis Report (USAR) 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 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.

Also, additional activities were performed during the evaluation of EC 379781 that were associated with TI 2515/177, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems." These activities are described in bullet .2 of this section.

This operability inspection constituted seven samples as defined in IP 71111.15-05.

b. Findings

(1) Failure to Monitor Unit 3 Drywell Temperature

Introduction: A finding of very low safety significance (Green) was identified by the inspectors for the failure to monitor Unit 3 (U3) drywell temperature per commitments made to the NRC within LER 88-22, Supplement 2 to ensure that equipment in the drywell was operating within environmental qualification limits.

Description: The inspectors reviewed IRs 1024796, "Adverse Temperature Trend in U2 DW [drywell] 4th Floor," and 1054599, "Adverse Trend in AVG Temp 4th Floor U2 DW." These IRs discussed a limit of 187 degrees as a maximum continuous temperature for environmental qualification in the drywell. On May 3, 2010, the inspectors observed that the temperature on the Unit 3 drywell temperature recorder 3-0260-20B for point #3 DW Ambient 578' was 194 degrees. The inspectors questioned the potential environment qualification impact on equipment in that area of the drywell. This concern was documented in IR 1064681, "NRC Resident – Question on U3 DW Ambient Temperature."

The licensee determined that the environmental qualified life of valve MOV 3-1301-01 (the isolation condenser inboard steam outlet from the reactor) was impacted. The EQ life of 3-1301-1 was originally evaluated using a maximum continuous temperature of 187 degrees. The licensee determined that the temperature in the vicinity of 3-1301-1 was as high as 199 degrees for the last six years. The licensee performed a preliminary evaluation of EQ components on the 3-1301-1 valve due to the elevated temperatures and concluded that the valve remained environmentally qualified through the next U3 refueling outage (D3R21) scheduled for November 2010. However, the licensee determined that the valve motor will have to be replaced in D3R21, which was two years earlier than originally scheduled.

The licensee determined and documented in IR 1070338, "Failure to Monitor EQ Equipment in the DW for Adverse Temps," that a previous high drywell temperature event on Unit 2 had resulted in licensee commitments to the NRC to monitor drywell temperatures which were documented in LER 88-22, Supplement 2 and the UFSAR. The licensee's LER made the following commitment:

"For Unit 2, an enhanced drywell temperature monitoring system was installed and is being implemented under Dresden Technical Procedure DTP-24 Rev. 1 dated 02/10/89 on a weekly basis. Temperature monitoring for Unit 2 began on 02/17/89. Results of this recording are identified in Sargent and Lundy (S&L) calculation CQD-048589 Rev. 1 dated 10/29/90 (237-200-88-13721)."

"For Unit 3, an enhanced drywell temperature monitoring system has been installed. Dresden Technical Procedure DTP-5 Rev. 0 dated 01/30/89 has been prepared to monitor drywell temperatures on a weekly basis. Enhancement of the Unit 3 drywell temperature monitoring system consisted of relocating two thermocouples to the third and fourth elevations to provide for a more accurate drywell bulk average temperature. This was performed under modification M12-3-29-2A (237-200-88-13722)."

In addition, UFSAR Section 6.2.1.2.8 stated that the drywell monitoring system monitors drywell temperature at predetermined locations and records the temperatures so that the temperature effects on environmentally qualified equipment located in the drywell can be determined. The licensee stated in IR 1070338, "Failure to Monitor EQ Equipment in the DW For Adverse Temps," that it appeared that plant engineering had not been evaluating the temperature impacts on EQ related equipment since 2000.

Analysis: The inspectors determined that the failure to monitor drywell temperature and evaluate the impact on environmentally qualified equipment in the Unit 2 and Unit 3 drywell was contrary to commitments made to the NRC in LER 0500237/1988-022 and UFSAR Section 6.2.1.2.8 and was a performance deficiency.

The finding was determined to be more than minor because the finding was associated with the Mitigating Systems Cornerstone attribute of Equipment Performance and affected the cornerstone objective of ensuring the reliability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to monitor drywell temperatures and evaluate those temperatures against environmental qualification limits would have resulted in the motor operator for valve 3-1301-1 to exceed its qualified life without the licensee's knowledge after D3R21.

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 Systems Cornerstone. The inspectors were able to answer the first question, is the finding a design or qualification deficiency confirmed not to result in loss of operability or functionality, "yes" and, therefore, the finding screened as Green.

The inspectors did not identify a cross-cutting aspect associated with this finding because the age of the finding did not reflect current performance.

Enforcement: No violation of regulatory requirements occurred. The licensee's corrective actions were extensive and are documented in IR 1070338. In summary, the licensee planned to reinstate the temperature monitoring of both drywells, perform walkdowns of both drywells for correct placement and to verify functionality of drywell thermocouples, improve administrative requirements on plant engineering turnover from one engineer to another, and train engineers to focus on identifying and validating assumptions when performing or reviewing technical products.

(**FIN 05000237/2010003-04; 05000249/2010003-04**, Failure to Monitor Unit 3 Drywell Temperature)

.2 Operability Evaluations associated with Temporary Instruction (TI) 2515/177, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems."

a. Inspection Scope

The inspectors reviewed the following issues associated with the scope of GL 2008-01, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems":

- EC 379781, "Evaluation of air identified in U2 HPCI discharge line."

The inspectors assessed the licensee's efforts for identifying the gas intrusion mechanisms that apply to the plant (TI 2515/177, Section 04.02.e). In addition, the inspectors verified that the licensee's void acceptance criteria was consistent with the Office of Nuclear Reactor Regulations' (NRR) void acceptance criteria. If NRR's acceptance criteria were not met, then the inspectors verified that the licensee justified the deviations. Also, the inspectors confirmed that (1) the licensee addressed the effect of pressure changes during system startup and operation since such changes could significantly affect the void fraction from the initial value; and (2) the range of flow conditions evaluated by the licensee was consistent with the full range of design basis and expected flow rates for various break sizes and locations (TI 2515/177, Section 04.02.f).

The inspectors noted that the licensee's CAP did not capture the discovery of an incorrectly sloped pipe that was previously found to be acceptable by the licensee during their walkdowns associated with the resolution of GL 2008-01. Specifically, following an unsuccessful fill and vent of the HPCI system, the licensee discovered that the actual high point of the HPCI discharge pipe, 2-2304-14"-C, was on the opposite end of the pipe from the vent valves. The licensee adjusted the hangers to properly slope the pipe towards the high point vents. However, the licensee did not identify that this condition was overlooked by their GL 2008-01 walkdowns. Therefore, the CAP did not capture this instance where the licensee's GL 2008-01 walkdowns did not adequately assess the plant's condition with respect to gas accumulation. The licensee captured the inspector's observation in AR01081055.

Documents reviewed are listed in the Attachment to this report.

This inspection effort counts towards the completion of TI 2515/177, which will be closed on a later Inspection Report.

b. Findings

No findings of significance were identified.

1R18 Plant Modifications (71111.18)

.1 Permanent Plant Modifications

a. Inspection Scope

The following engineering design package was reviewed and selected aspects were discussed with engineering personnel:

- EC 372027, “Revise EDG [emergency diesel generator] Day Tank Level Switches Setpoints and Tolerances,” Revision 0.

This document and related documentation were reviewed for adequacy of the associated 10 CFR 50.59 safety evaluation screening, consideration of design parameters, implementation of the modification, post-modification testing, and relevant procedures, design, and licensing documents were properly updated. The inspectors observed ongoing and completed work activities to verify that installation was consistent with the design control documents. The modification changed the day tank level switches setpoints and tolerance. Documents reviewed in the course of this inspection are listed in the Attachment to this report.

This inspection constituted one permanent plant modification sample as defined in IP 71111.18-05.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

.1 Post-Maintenance Testing

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:

- WO [work order] 01136214-14, “VT-2 2/3 B DGCW [diesel generator cooling water] Piping;”
- WO 453730, “D3 8Y PM Lubricate Coupling, 3D CCSW [containment cooling service water] Pump;”
- WO 01071924-01, “3-1401-4 U3 ECCS [emergency core cooling system] Jockey Pump Seal Leak Needs Repair;”
- WO 01346169-01, “TS/Repair Loss of U3 ESS [essential service system] UPS [uninterrupted power supply] Normal AC Supply;” and
- WO 99019477, “D3 12Y PM Replace ‘A’ LPCI Pump Mechanical Seal.”

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 constituted five post-maintenance testing samples as defined in IP 71111.19-05.

b. Findings

A violation of very low safety significance (Green) was identified by the licensee and is discussed in Section 4OA7 of this report.

1R22 Surveillance Testing (71111.22)

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:

- DOS 2300-03, Revision 92, “U2 high pressure coolant injection in-service testing” (IST);
- DOP 2000-24, Revision 23, “Drywell Sump Operation Procedure” (RCS);
- WO 01125716-01, “D3 24M TS DIV I LPCI INJ Cont Spray, Torus & 2/3 Core Cov” (routine);
- WO 1122412-01, “OP D3 2Y REG FUEL CONSUMPTION TEST” (routine); and
- WO 1136229-01, “IM D3 24M TS Isolation Condenser Auto Actuation” (routine).

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 frequency 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, American Society of Mechanical Engineers (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 three routine surveillance testing samples, one inservice testing sample, and one reactor coolant system leak detection inspection sample, as defined in IP 71111.22, Sections -02 and -05.

b. Findings

Failure to Provide An Adequate Procedure for Several Instrument Maintenance Surveillance Tests

Introduction: The inspectors identified a NCV of TS Paragraph 5.4.1 for the failure to provide an adequate procedure for several instrument maintenance surveillance tests. This issue was determined to be of very low safety significance, “Green.”

Description: The inspectors observed the performance of DIS 1300-03, “Isolation Condenser Initiation and Isolation Logic System Functional Test,” Revision 19, on May 12, 2010. There were no verification requirements for the installation and removal of volt meters used in the resistance measuring mode in step 171 of the procedure. During the execution of DIS 1300-03 two volt-ohm-meters (VOMs) were used in the resistance mode during the surveillance test. If the VOM remained in the circuit and power was returned, the VOM could propagate an electrical short on the circuit and either cause unwanted plant reactions or challenge the plant circuit’s electrical protection, i.e., fuse or circuit breaker.

The inspectors interviewed two instrument maintenance department management personnel and verified that if the volt meters were to be inadvertently left in place, it would have had a detrimental impact on isolation system logic. The Exelon Quality Assurance Topical Report, NO AA 10, Revision 81, Appendix C, Paragraph 1.1, commits the licensee to American Nuclear Standard ANS 3.2/ANSI N18.7 – 1988, which requires an independent verification of the installation and removal of temporary cables or jumpers that modify the circuit logic.

The inspectors informed the operations unit supervisor who documented the procedure deficiency in issue report (IR) 1068559, "NRC Questions the Need for Verification in Surveillance."

Analysis: The inspectors determined that the failure to independently verify the correct installation and restoration of equipment was a performance deficiency warranting a significance evaluation. Using IMC 0612, Appendix E, "Examples of Minor Violations," issued on September 20, 2007, the inspectors determined that there were no similar examples to this finding in Appendix E. The inspectors referenced IMC 0612, Appendix B, "Issue Screening," dated January 1, 2010. The inspectors determined that the finding was more than minor based on Block 9, Figure 2, paragraph 2.b, "If left uncorrected, would the finding become a more significant safety concern."
The inspectors determined that the failure to perform an independent verification that a testing configuration had been returned to normal, could result in the inability of a system or component to perform its function which would be a more significant safety concern.

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, Initiating Events Cornerstone Column, Transient Initiators question 1, does the finding contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available, was answered "no," and, therefore, screened as Green.

This finding had a cross-cutting aspect in the area of Problem Identification and Resolution – Corrective Actions because the licensee did not address a previously identified safety issue in a timely manner. Specifically, the licensee received a similar violation in Dresden Inspection Report, 05000237/249/2008004. That violation was entered in the licensee's CAP as IR 786449 and IR 805061. All the corrective actions associated with those two IRs were completed as of December 31, 2009. P.1(d)

Enforcement: Technical Specification 5.4.1, states, in part, written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978.

Regulatory Guide 1.33, Revision 2, Appendix A, February 1978, paragraph1.c lists equipment control as an applicable procedure.

The Exelon Quality Assurance Topical Report, NO AA 10, Revision 81, Appendix C, Paragraph 1.1, commits the licensee to American Nuclear Standard ANS 3.2/ANSI N18.7 – 1988.

American Nuclear Standard ANS 3.2/ANSI N18.7 – 1988, paragraph 5.2.6, Equipment Control, states in part, temporary modifications, such as temporary bypass lines, electrical jumpers, lifted electrical leads, and temporary trip point settings, shall be

controlled by approved procedures, which shall include requirements for the period of time for which the temporary modification is in effect, and a requirement for independent verification by either a second person or by a functional test which conclusively proves the proper installation or removal of the temporary modification. A log, or other documented evidence, shall be maintained of the current status of such temporary modifications.

Contrary to the above, on May 12, 2009, the licensee did not have written procedures established to perform independent verifications of the installation or restoration of VOM cables acting as electrical jumpers during the performance of surveillance test DIS 1300-03, "Isolation Condenser Initiation and Isolation Logic System Functional Test," Revision 19. This issue was entered in the licensee's CAP as IR 106559. The licensee's corrective actions included a task to add requirements for an independent verification of the removal of VOMs in the ohm meter mode in 17 affected procedures. 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 or the NRC Enforcement Policy. (**NCV 05000237/2010003-05; 05000249/2010003-05**; Failure to Provide An Adequate Procedure for Several Instrument Maintenance Surveillance Tests)

1EP6 Drill Evaluation (71114.06)

.1 Emergency Preparedness Drill Observation

a. Inspection Scope

The inspectors evaluated the conduct of a 2010 off-year exercise emergency drill on April 21, 2010, to identify any weaknesses and deficiencies in classification, notification, and protective action recommendation development activities. The inspectors observed emergency response operations in the control room simulator and technical support center to determine whether the event classification, notifications, and protective action recommendations were performed in accordance with procedures. The inspectors also attended the licensee drill critique to compare any inspector-observed weakness with those identified by the licensee staff in order to evaluate the critique and to verify whether the licensee staff was properly identifying weaknesses and entering them into the CAP. As part of the inspection, the inspectors reviewed the drill package and other documents listed in the Attachment to this report.

This emergency preparedness drill inspection constituted one sample as defined in IP 71114.06-05.

b. Findings

No findings of significance were identified.

.2 Training Observation

a. Inspection Scope

The inspector observed a simulator training evolution for licensed operators on March 15, 2010, and again on April 12, 2010, 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 crew. 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 crew's performance and ensure that the licensee evaluators noted the same issues and entered them into the corrective action program. As part of the inspection, the inspectors reviewed the scenario package and other documents listed in the Attachment to this report.

Note that one of the above samples was performed in the first quarter of 2010, but was omitted from Inspection Report 05000237/2010-002; 05000249/2010-002. These inspections of the licensee's training evolution with emergency preparedness drill aspects constituted two samples as defined in IP 71114.06-05.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

This inspection constituted one radiological hazard assessment and exposure control sample as defined in IP 71124.01-05.

.1 Inspection Planning (02.01)

a. Inspection Scope

The inspectors reviewed all licensee performance indicators (PIs) for the Occupational Exposure Cornerstone for follow-up. The inspectors reviewed the results of radiation protection program audits (e.g., licensee's quality assurance audits or other independent audits). The inspectors reviewed any reports of operational occurrences related to occupational radiation safety since the last inspection. The inspectors reviewed the results of the audit and operational report reviews to gain insights into overall licensee performance.

b. Findings

No findings of significance were identified.

.2 Radiological Hazard Assessment (02.02)

a. Inspection Scope

The inspectors determined if there have been changes to plant operations since the last inspection that may result in a significant new radiological hazard for onsite workers or members of the public. The inspectors determined whether the licensee assessed the potential impact of these changes and had implemented periodic monitoring, as appropriate, to detect and quantify the radiological hazard.

The inspectors reviewed recent radiological surveys from several selected plant areas. The inspectors evaluated the thoroughness and frequency of the surveys to determine if they were appropriate for the given radiological hazard.

The inspectors conducted walkdowns of the facility, including radioactive waste storage, and handling areas to evaluate material conditions and performed independent radiation measurements to verify conditions.

The inspectors selected the following radiologically risk-significant work activities that involved exposure to radiation.

- Unit 2 heater bay at power entry for leak inspection;
- Unit 2 noble metals skid valve repair;
- Unit 2 dryer/separator pit decontamination;
- Unit 2 cavity decontamination; and
- Unit 2 recirculation system valve breach.

For these work activities, the inspectors determined whether the pre-work surveys performed were appropriate to identify and quantify the radiological hazard and to establish adequate protective measures. The inspectors evaluated the radiological survey program to determine if hazards were properly identified, including the following:

- the identification of hot particles;
- the presence of alpha emitters;
- the potential for airborne radioactive materials, including the potential presence of transuranics and/or other hard-to-detect radioactive materials;
- the hazards associated with work activities that could suddenly and severely increase radiological conditions and whether the licensee established a means to inform workers of changes that could significantly impact their occupational dose; and
- the radiation dose gradients that can result in non-uniform exposures of the body.

The inspectors observed work in potential airborne areas and evaluated whether the air samples were representative of the breathing air zone. The inspectors evaluated whether continuous air monitors were located in areas with low background to minimize false alarms and were representative of actual work areas. The inspectors determined whether the licensee had a program for monitoring levels of loose surface contamination in areas of the plant with the potential for the contamination to become airborne.

The inspectors reviewed and inspected the postings and physical controls for high radiation areas (HRAs) and very high radiation areas (VHRAs), to the extent necessary to verify conformance with the Occupational Radiation Safety Performance Indicator.

b. Findings

No findings of significance were identified.

.3 Instructions to Workers (02.03)

a. Inspection Scope

The inspectors selected approximately 10 containers holding nonexempt licensed radioactive materials that may cause unplanned or inadvertent exposure of workers, and determined whether the containers were labeled and controlled in accordance with 10 CFR 20.1904, “Labeling Containers,” or otherwise met the requirements of 10 CFR 20.1905(g).

The inspectors reviewed the following RWPs used to access HRAs and evaluated the specified work control instructions or control barriers.

- RWP 10011754 – Unit 2 Online Noble Chemistry Applications;
- RWP 10010430 – Drywell Emergent Work Activities; and
- RWP 10010403 – Drywell Basement Decontamination Activities.

For these RWPs, the inspectors assessed whether allowable stay times or permissible dose (including from the intake of radioactive material) for radiologically significant work under each RWP were clearly identified. The inspectors evaluated whether electronic personal dosimeter (EPD) alarm set-points were in conformance with survey indications and plant policy.

The inspectors reviewed selected occurrences where a worker’s EPD noticeably malfunctioned or alarmed. The inspectors evaluated whether workers responded appropriately to the off-normal condition. The inspectors assessed whether the issue was included in the corrective action program and dose evaluations were conducted as appropriate.

b. Findings

No findings of significance were identified.

.4 Contamination and Radioactive Material Control (02.04)

a. Inspection Scope

The inspectors observed locations where the licensee monitors potentially contaminated material leaving the radiologically controlled area (RCA), and evaluated the methods used for control, survey, and release of materials from these areas. The inspectors observed the performance of personnel surveying and releasing material for unrestricted use and evaluated whether the work was performed in accordance with plant procedures. The inspectors also reviewed whether the procedures were sufficient to control the spread of contamination and prevent unintended release of radioactive materials from the site. The inspectors assessed whether the radiation monitoring instrumentation used for these surveys had appropriate sensitivity for the type(s) of radiation present.

The inspectors reviewed the licensee’s criteria for the survey and release of potentially contaminated material. The inspectors evaluated whether there was guidance on how to respond to an alarm that indicates the presence of licensed radioactive material.

The inspectors reviewed the licensee's procedures and records to verify that the radiation detection instrumentation was used at its typical sensitivity level based on appropriate counting parameters. The inspectors assessed whether or not the licensee had established a de facto "release limit" by altering the instrument's typical sensitivity through such methods as raising the energy discriminator level or locating the instrument in a high-radiation background area.

The inspectors selected several sealed sources from the licensee's inventory records and assessed whether the sources were accounted for and verified to be intact (i.e., they were not leaking their radioactive content).

The inspectors verified that any transactions, since the last inspection, involving nationally tracked sources were reported in accordance with 10 CFR 20.2207. No transactions had occurred.

b. Findings

No findings of significance were identified.

.5 Radiological Hazards Control and Work Coverage (02.05)

a. Inspection Scope

The inspectors evaluated ambient radiological conditions (e.g., radiation levels or potential radiation levels) during tours of the facility. The inspectors assessed whether the conditions were consistent with applicable posted surveys, RWPs, and worker briefings.

The inspectors evaluated the adequacy of radiological controls, such as required surveys, radiation protection job coverage (including audio and visual surveillance for remote job coverage), and contamination controls. The inspectors evaluated the licensee's use of EPDs in high noise areas as HRA monitoring devices.

The inspectors assessed whether radiation monitoring devices were placed on the individual's body consistent with licensee procedures. The inspectors assessed whether the dosimeter was placed in the location of highest expected dose or that the licensee otherwise properly employed an NRC-approved method of determining effective dose equivalent.

The inspectors reviewed the application of dosimetry to effectively monitor exposure to personnel in high-radiation work areas with significant dose rate gradients.

The inspectors reviewed various work packages and associated RWPs for work conducted within airborne radioactivity areas with the potential for individual worker internal exposures. In particular, the inspectors reviewed RWP 10010430 for drywell emergent work performed in November 2009, which included recirculation system valve repairs.

For this RWP, the inspectors evaluated airborne radioactive controls and monitoring, including potential for significant airborne levels (e.g., grinding, grit blasting, system breaches, entry into tanks, cubicles, and reactor cavities). The inspectors assessed whether barrier (e.g., tent or glove box) integrity and temporary high-efficiency

particulate air (HEPA) ventilation system operation, for this and other potential airborne radioactive material work activities, was adequate.

The inspectors examined the licensee's physical and programmatic controls for highly activated or contaminated materials (nonfuel) stored within spent fuel and other storage pools. The inspectors assessed whether appropriate controls (i.e., administrative and physical controls) were in place to preclude inadvertent removal of these materials from the pool.

The inspectors inspected the posting and physical controls for selected HRAs and VHRAs, to verify conformance with the Occupational PI.

b. Findings

No findings of significance were identified.

.6 Risk-Significant High Radiation Area and Very High Radiation Area Controls (02.06)

a. Inspection Scope

The inspectors discussed with the Radiation Protection Manager (RPM) the controls and procedures for high-risk HRAs and VHRAs. The inspectors assessed whether any changes to licensee procedures substantially reduced the effectiveness and level of worker protection.

The inspectors reviewed special areas that have the potential to become VHRAs during certain plant operations (e.g., pressurized-water reactor (PWR) thimble withdrawal into the reactor cavity sump; boiling-water reactor (BWR) traversing in-core probe movement; BWR drywell fuel transfer slot area; spent fuel pool, cavity, or pit diving). The inspectors discussed these areas with first-line health physics (HP) supervisors (or equivalent positions having backshift HP oversight authority) to assess whether the communication protocols with the HP group would allow for corresponding timely actions to properly post, control, and monitor the radiation hazards including re-access authorization. The inspectors evaluated licensee controls for VHRAs, and areas with the potential to become a VHRA, to ensure adequate controls were in-place to prevent an individual from unauthorized access into the VHRA.

b. Findings

No findings of significance were identified.

.7 Radiation Worker Performance (02.07)

a. Inspection Scope

The inspectors observed radiation worker performance with respect to stated radiation protection work requirements. The inspectors assessed whether workers were aware of the significant radiological conditions in their workplace and the RWP controls/limits in place and that their performance reflected the level of radiological hazards present.

The inspectors reviewed several radiological problem reports since the last inspection that found the cause of the event to be human performance errors. The inspectors evaluated whether there was an observable pattern traceable to a similar cause.

The inspectors assessed whether this perspective matched the corrective action approach taken by the licensee to resolve the reported problems. The inspectors discussed with the RPM any problems with the corrective actions planned or taken.

b. Findings

No findings of significance were identified.

.8 Radiation Protection Technician Proficiency (02.08)

a. Inspection Scope

The inspectors observed the performance of radiation protection technicians with respect to radiation protection work requirements. The inspectors evaluated whether technicians were aware of the radiological conditions in their workplace and the RWP controls/limits and whether their performance was consistent with their training and qualifications with respect to the radiological hazards and work activities.

The inspectors reviewed several radiological problem reports since the last inspection that found the cause of the event to be radiation protection technician error.

The inspectors evaluated whether there was an observable pattern traceable to a similar cause. The inspectors assessed whether this perspective matched the corrective action approach taken by the licensee to resolve the reported problems.

b. Findings

No findings of significance were identified.

.9 Problem Identification and Resolution (02.09)

a. Inspection Scope

The inspectors evaluated whether problems associated with radiation monitoring and exposure control were being identified by the licensee at an appropriate threshold and were properly addressed for resolution in the licensee's corrective action program. The inspectors assessed the appropriateness of the corrective actions for a selected sample of problems documented by the licensee that involved radiation monitoring and exposure controls. The inspectors assessed the licensee's process for applying operating experience to their plant.

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety

2RS8 Radioactive Solid Waste Processing and Radioactive Material Handling, Storage and Transportation (71124.08)

This inspection constituted one partial sample as defined in IP 71124.08-05.

.1 Shipment Preparation (02.05)

a. Inspection Scope

The inspectors reviewed the circumstances surrounding an April 29, 2010, radioactive waste shipment problem that involved a sea land container used to transport contaminated scrap metal to a waste processor. The sea land container was placed in an overpack (bag) because the container had rusted. During the shipment, the zipper seal of the shipment overpack failed potentially impacting shipment integrity.

The inspectors evaluated the licensee's use of the overpack, reviewed packaging certifications and the licensee's overall shipment readiness to determine whether the packaging along with its overpack were used consistent with the package manufacturer's recommendations and procedures. Radiological surveys completed before and after shipment were reviewed by the inspectors. Corrective actions were also evaluated for adequacy.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Mitigating Systems Performance Index - Emergency AC Power System

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index (MSPI) - Emergency AC Power System performance indicator (PI) for Unit 2 and Unit 3 for the period from the third quarter 2009 through the second quarter 2010. 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 6, were used. The inspectors reviewed the licensee's operator narrative logs, issue reports, event reports and NRC Inspection Reports for the period of April 2009 through March 2010 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.

This inspection constituted two MSPI emergency AC power system samples as defined in IP 71151-05.

b. Findings

No findings of significance were identified.

.2 Mitigating Systems Performance Index - High Pressure Injection Systems

a. Inspection Scope

The inspectors sampled licensee submittals for the MSPI - High Pressure Injection Systems performance indicator for Unit 2 and Unit 3 for the period from the third quarter 2009 through the first quarter 2010. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, dated October 2009, 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 April 2009 through March 2010 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.

This inspection constituted two MSPI high pressure injection system samples as defined in IP 71151-05.

b. Findings

No findings of significance were identified.

.3 Reactor Coolant System Specific Activity

a. Inspection Scope

The inspectors sampled licensee submittals for the Reactor Coolant System (RCS) Specific Activity performance indicator for Units 2 and 3 for the period from April 2009, through April 2010. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, dated October 2009, was used. The inspectors reviewed the licensee's RCS chemistry samples including isotopic analyses to determine if the greatest dose equivalent iodine values obtained during steady state operations corresponded to the values reported to the NRC. The inspectors also reviewed issue reports and licensee event reports for the period specified above to validate the accuracy of the submittals. Additionally, the inspectors 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. In addition to record reviews, the inspectors observed a chemistry

technician obtain and analyze a RCS sample. Documents reviewed are listed in the Attachment to this report.

This inspection constituted two RCS specific activity samples 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 Corrective Action Program

a. Inspection 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 Reviews

a. Inspection 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. Inspection 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. The inspectors' review nominally considered the six month period of January 1, 2010, through June 30, 2010, although some examples expanded beyond those dates where the scope of the trend warranted.

The review also included issues documented outside the normal CAP in major equipment problem lists, repetitive and/or rework maintenance lists, departmental problem/challenges lists, system health reports, quality assurance audit/surveillance reports, self assessment reports, and Maintenance Rule assessments. The inspectors compared and contrasted their results with the results contained in the licensee's CAP trending reports. Corrective actions associated with a sample of the issues identified in the licensee's trending reports were reviewed for adequacy.

The inspectors identified two trends:

(1) Alarms coming in with immediate resets with no sequence of event recorder acknowledgement

IR 1074392, 903-3 G-12 HPCI Control Power Failure Alarm Received;
IR 1074868, 903-3 Panel Alarm E-7 (LPCI Pump Overload);
IR 1074867, Alarm 903-3 Panel E-7 (CCSW Pump Trip).

(2) Several Examples where IRs were not written
The inspectors identified several examples during the inspection period where IRs should have been written and were not. The licensee entered this into theIr CAP as IR 1080695, "Some Instances of IRs Not Written for Events/Conditions."

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

b. Findings

No findings of significance were identified.

.4 Annual Sample: Review of Operator Workarounds (OWAs)

a. Inspection Scope

The inspectors evaluated the licensee's implementation of their process used to identify, document, track, and resolve operational challenges. Inspection activities included, but were not limited to, a review of the cumulative effects of the OWAs on system availability

and the potential for improper operation of the system, for potential impacts on multiple systems, and on the ability of operators to respond to plant transients or accidents.

The inspectors performed a review of the cumulative effects of OWAs. The documents listed in the Attachment to this report were reviewed to accomplish the objectives of the inspection procedure. The inspectors reviewed both current and historical operational challenge records to determine whether the licensee was identifying operator challenges at an appropriate threshold, had entered them into their CAP and proposed or implemented appropriate and timely corrective actions which addressed each issue. Reviews were conducted to determine if any operator challenge could increase the possibility of an Initiating Event, if the challenge was contrary to training, required a change from long-standing operational practices, or created the potential for inappropriate compensatory actions. Additionally, all temporary modifications were reviewed to identify any potential effect on the functionality of Mitigating Systems, impaired access to equipment, or required equipment uses for which the equipment was not designed. Daily plant and equipment status logs, degraded instrument logs, and operator aids or tools being used to compensate for material deficiencies were also assessed to identify any potential sources of unidentified operator workarounds.

This review constituted one operator workaround annual inspection sample as defined in IP 71152-05.

b. Findings

No findings of significance were identified.

.5 Selected Issue Follow-Up Inspection: Human Performance Improvement Initiatives

a. Inspection Scope

The corrective action plan associated with IR 1039023, “NRC Identifies Substantive Cross-Cutting Issue in Human Performance” was reviewed as follow-up to a previous substantive cross-cutting issue in the area of Human Performance - Work Practices as described in IMC 0310, “Components Within The Cross-Cutting Areas,” dated February 23, 2010, (H.4.(a)). The quality of the action closures was reviewed as well as the thoroughness of the on-going actions. During the review, inspectors monitored implementation of programs put in place to reinforce human performance improvement initiatives both in the office and in the power plant. Although the licensee continued to be challenged by human performance issues in different areas, the inspectors found no new significant issues involving the failure to use Human Performance Tools.

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

b. Findings

No findings of significance were identified.

.6 Selected Issue Follow-Up Inspection: Failure to Perform An Adequate Inspection:

a. Inspection Scope

During a review of items entered in the licensee's CAP, the inspectors reviewed a corrective action evaluation item documenting the failure of a circulating water directional valve. The inspectors reviewed the licensee's equipment apparent cause evaluation (EACE), including associated documentation described in the EACE, and attended the management review committee meeting. This review constituted one in-depth problem identification and resolution sample as defined in IP 71152-05.

b. Findings

(1) Failure To Perform An Adequate Inspection of Circulating Water Valve 3-4402-C

Introduction: A finding of very low safety significance (Green) was self-revealed for failure to perform an adequate inspection of the grease condition of the 3-4402-C valve actuator HBC gear box, which was contrary to the requirements of MA-AA-723-301,"Periodic Inspection of Limitorque Model SMB/SB/SBD-000", Revision 3, step 4.3.8.

Description: On January 22, 2010, operations department personnel performed a reversal of circulating water flow through the Unit 3 condenser. This evolution was performed weekly to maximize condenser heat exchange capability. Valve 3-4402-C started to close, but did not fully close and, after valve movement stopped, the valve could not be moved in either the open or closed direction. Valve 3-4402-C acted as an inlet in the circulating water system for the south water box. When the valve failed, it was almost completely closed. Had the valve failed open, circulating water would have been diverted from the condenser, potentially causing a loss of vacuum that would have resulted in a reduction in power and/or a turbine trip and reactor trip. The 3-4402-C valve was operated by a Limitorque SMB-3 operator attached to a HBC-6 gearbox attached to a 120 inch Pratt Butterfly Valve.

The inspectors reviewed apparent cause evaluation (ACE) 1034444. The ACE referenced Failure Analysis (DRE-86577), which showed that a piece of the actuator segment gear had broken off and the broken piece lodged between the worm gear and the segment gear. The licensee documented, within the ACE, that the failure was caused due to the deterioration of the grease within the HBC gear box housing. The ACE also stated that the external gear box was found cracked. The gear box was destroyed in the investigation and that a cause for the failure of the gear box was neither identified nor speculated.

The Failure Analysis stated that the worm gear bearings failed due to degraded grease. This caused the worm gear to put a stress on the segment gear. The segment gear failed due to stress corrosion cracking. The stress corrosion cracking was most likely caused by the stress imparted on the segment gear from the mis-aligned worm gear and that the degraded grease could have caused the corrosive environment within the gear box. The Failure Analysis stated that the corrosive conditions could also have been caused by an intrusion of water. The inspectors determined that water intrusion could have been a symptom of the HBC gear box failure.

The licensee stated in the ACE that the HBC gear box for valve 3-4402-C did not have inspection ports. The valve was scheduled for inspection every 12 years. Licensee maintenance procedure MA-AA-723-301, ”Periodic Inspection of Limitorque Model SMB/SB/SBD-000”, Revision 3, step 4.3.8 allowed the grease condition in the HBC gear box to be assumed the same as the Limitorque gear box. This assumption was based on identical service conditions for the two gear boxes and no indications of external grease leakage.

However, the Limitorque gear box was replaced in 1995 and the HBC gear box was not. Per the equipment ACE 1034444, when the valve was inspected in 2008 under work order (WO) 758827-01, the grease condition in the HBC gear box was assumed to be the same as the grease condition in the Limitorque gear box. This assumption was incorrect due to the replacement of the Limitorque gear box in 1995.

In addition, the inspectors reviewed a copy of the inspection results for the 3-4402-C performed in accordance procedure with MA-AA-723-301 under WO 758827 on November 3, 2008. The inspection results were documented in MA-AA-723-301, Attachment 2. The inspection result for step 4.2.1, “no external signs of grease leakage” was marked as “UNSAT.” There was no documentation in the work order of any actions that were taken due to the “UNSAT” inspection result. This information was not identified in ACE 1034444.

The inspectors searched for and identified that IR 839819, “3-4402-C – Actuator U3 CDSR Inlet VLV,” was written on November 3, 2008, and stated that grease was leaking from the 3-4402-C pointer cap, which could indicate that the drive sleeve o-ring needed to be replaced. The licensee wrote Work Request 839819 on November 4, 2008, to fix the grease leakage.

Analysis: The inspectors determined that the failure to perform an adequate inspection of the grease condition of the 3-4402-C actuator HBC gear box was contrary to the requirements of MA-AA-723-301, ”Periodic Inspection of Limitorque Model SMB/SB/SBD-000”, Revision 3, step 4.3.8 and was a performance deficiency.

The finding was determined to be more than minor because the finding could be reasonably viewed as a precursor to a significant event. Specifically, valve 3-4402-C acted as an inlet in the circulating water system for the south water box. When the valve failed, it was almost completely closed. Had the valve failed open, circulating water would have been diverted from the condenser potentially causing a loss of vacuum that would have resulted in a reduction in power and/or a turbine trip and reactor trip. 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 4a, Initiating Events Cornerstone Column, Transient Initiators question 1, does the finding contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available, was answered “no,” and, therefore, screened as Green.

This finding had no cross-cutting aspect due to the issues involved in this valve failure were not indicative of current performance. Limitorque Maintenance Update 92-1 recommended that two inspection ports be drilled and tapped 90 degrees apart on the

HBC gear-head center line. The licensee did not act on this recommendation. However, this decision was made years ago and was not indicative of current performance. The licensee's inspection procedure, MA-AA-723-301, stated that if there were no inspection ports on the HBC gear box (and there were not), the assumption could be made that the Limitorque gear box and the HBC gear box had the same grease quantity and quality if there was no external grease leakage because of similar service conditions. The licensee's ACE stated the cause of this problem was the Limitorque gear box was replaced in 1995 and the HBC gear box was not, therefore, the service conditions were no longer the same for the two gear boxes at the time of the 2008 valve inspection. However, the failure to document this change was made years ago and was not indicative of current performance.

The inspectors identified an additional issue with the 2008 valve inspection documentation performed under WO 758827 on November 3, 2008. The inspection determined that the grease leakage was unsatisfactory with no additional documentation in the work order of whether the grease leakage was from the Limitorque gear box or the HBC gear box, and what if anything was done about the unsatisfactory condition. This information was not in ACE 1034444.

If the HBC gear box leakage was observed to be unsatisfactory, then the performer of the valve inspection could not assume that the grease conditions were the same. The inspectors did a word search of issue reports for a seven day period following the 2008 inspection, which was conducted on November 3, 2008. The inspectors found an IR 839819, "3-4402-C – Actuator U3 CDSR Inlet VLV," which stated that the grease was leaking from the 3-4402-C pointer cap, which could indicate that the drive sleeve o-ring needed to be replaced. The licensee wrote work request 839819 on November 4, 2008, to fix the grease leakage, but the work request was not performed prior to the HBC segment gear failure. Based on this additional information, the inspectors concluded that if the licensee had attempted to repair the grease leakage identified in November 2008, that the failure of the 3-4402-C in January of 2010 might have been avoided. However, since the valve was neither safety-related nor augmented quality, there was no regulatory or programmatic requirement to address the degraded condition in a manner more timely than what was planned. The inspectors were unable to determine what role the cracked HBC gearbox (which was not documented in 2008 inspection) had on the failure of the HBC segment gear.

Enforcement: No violation of regulatory requirements occurred because valve 3-4402-C was a nonsafety-related component. The licensee planned to drill inspection ports into and/or replace the HBC gear boxes for valves 2/3-34403-A(B)(C)(D) and 2/3-34402-A(B)(C)(D) and 2-34402-C and change the preventive maintenance requirement to perform a 12 year mechanical inspection of the HBC gear box. This finding was placed in the licensee's corrective action program as IR 1034444, "Failure of the 3-4402-C Condenser Inlet Valve." (**FIN 05000237/2010003-06; 05000249/2010003-06;** Failure To Perform An Adequate Inspection of Circulating Water Valve 3-4402-C)

4OA3 Follow-Up of Events and Notices of Enforcement Discretion (71153)

.1 (Closed) Unresolved Item (URI) 05000237, 05000249/2009005-04 "2/3 Emergency Diesel Generator (EDG) Overvoltage During Division 1 Undervoltage"

Introduction: A finding of very low safety significance (Green) and associated NCV of Title 10 of the Code of Federal Regulations (CFR), Part 50, Appendix B, Criterion XI, "Test Control," was self-revealed for the failure to perform an adequate post-modification test, which allowed an improper wiring condition to exist for years before an inappropriate action by a nuclear station operator (NSO) exposed the problem.

Description: On November 16, 2009, a nuclear station operator was performing DOS 6600-06, "Bus Undervoltage and ECCS Integrated Functional Test for Unit 2/3 Diesel Generator to Unit 2," per Work Order (WO) 1082066, "D2 24M/RFL TS Bus 23-1 UV and ECCS Integrated Functional Tests." At the time, the NSO was attempting to synchronize Bus 23-1 to Bus 23. The operator stated that as KVARs just started to lower, a loud noise was heard coming from the 902-3 panel in the control room. The operator noticed that the 23-1/24-1 digital voltage meter read around 5600 volts.

As a result, various components of electrical equipment were subjected to an overvoltage condition. Fuses were blown, various control room panel instrument indications were lost or went downscale and some control room annunciators did not reset. In addition, Technical Specification 3.8.1, "AC Sources – Operating," Condition B, "One required DG inoperable," planned completion time was extended for Unit 3 due to the 2/3 EDG being inoperable for an extended period of time due to the broken voltage regulator.

Troubleshooting under WO 1286397, "2/3 EDG Voltage Transient," later revealed that wiring from the local voltage control switch on the 2/3 EDG Neutral Grounding Compartment to the motorized voltage regulator potentiometer (2/3-6601-MVR) was landed on incorrect terminals. The wires were historically miswired at the local control switch, which, in turn, removed over-travel protection on the motorized voltage regulator potentiometer (designated MVR) when it was operated beyond its limits. The wire that should have been connected to point 18T was found connected to point 21, and the wire that should have been connected to point 21 was found connected to point 18T.

The EDG had been in this condition since modification M12-2/3-83-14 was installed on June 4, 1986, to add controls for the 2/3 EDG's automatic voltage regulator (AVR) locally in the 2/3 EDG room to mitigate a fire scenario. The MVR limit switches were adjusted from the local switch, Diesel 2/3 Voltage Adjust Switch, so that the MVR limit switches kept the wiper arm from riding off the coil as long as the MVR was operated from the local control switch (the MVR wiper arm is designed to be stopped by limit switches before operator/system demand runs the wiper arm off the potentiometer windings). However, operating the MVR using the control room switch resulted in the possibility of running the wiper arm off the MVR, since the limits were not aligned for operation from the control room switch due to the wiring error.

The wiring problem was found inside a cabinet that had been procured as a pre-wired fixture that would facilitate installation in the system. The post-modification testing was reviewed and voltage control was demonstrated from both the control room and locally while the EDG was operating, but the MVR limits were not validated using both switches.

Thus, it was a historical problem not found during post-modification testing that was caused by an error during fabrication of the cabinet by a vendor.

The licensee's equipment apparent cause evaluation (EACE) also revealed that on November 16, 2009, the NSO lowered EDG output voltage out of the normal range (greater than 3952 volts and lower than 4368 volts per DOS 6600-06), challenging the MVR limit switches during operation of the EDG. The operator lowered emergency diesel generator output voltage to approximately 3700 volts over a 7 second period. Then, the MVR wiper arm was driven off the coil. When the wiper arm left the coil, voltage reference was lost to the EDG AVR. When the AVR saw no reference voltage, it increased excitation to maximum. With EDG excitation at maximum, the EDG terminal voltage went to approximately 5800 volts for about 22 seconds, until the NSO opened the EDG output breaker. The peak voltage observed on the plant process computer 1-second sample rate was 5755 volts and the lowest voltage observed was 3681 volts.

Lowering EDG voltage below the normal range challenged the limits on the MVR and revealed the improper wiring. This problem would not have been revealed had the EDG been operated in accordance with DOS 6600-06, if voltage had not been lowered below the normal range.

Analysis: The inspectors determined that the failure to perform adequate post-modification testing and ensure that the equipment installed in the plant was in accordance with the design documentation was a performance deficiency warranting significance evaluation. Using IMC 0612, Appendix B, "Issue Screening," issued on December 24, 2009, the inspectors determined that the performance deficiency was more than minor because it impacted the Mitigating Systems Cornerstone attribute of procedure quality to ensure the availability, reliability and capability of system that respond to initiating events to prevent undesirable consequences. The improper wiring error had an impact on safety equipment (i.e., 2/3 EDG) and it resulted in an actual equipment problem. As discussed above in the Description section, wiring from the local voltage control switch on the 2/3 EDG Neutral Grounding Compartment to the voltage regulator potentiometer (2/3-6601-MVR) was landed on incorrect terminals. The wires were historically miswired at the local control switch, which in turn removed over-travel protection on the voltage regulator potentiometer when it was operated beyond its limits. The wire that should have been connected to point 18T was found connected to point 21, and the wire that should have been connected to point 21 was found connected to point 18T. Because the behavior leading to this event occurred in 1986, the inspectors determined that this event was not indicative of current performance and therefore no cross-cutting area was affected.

The inspectors completed a Phase 1 significance determination of this issue using IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," dated January 10, 2008. The inspectors answered "no" to all questions in the Mitigation Systems Cornerstone column of Table 4a, "Characterization Worksheet for IE, MS, and BI Cornerstones." Therefore, the finding screened as Green (very low safety significance).

Enforcement: Title 10 of the Code of Federal Regulations, Part 50, Appendix B, Criterion XI, "Test Control," requires, in part, that a test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written

test procedures which incorporate the requirements and acceptable limits contained in applicable design documents.

Contrary to the above, on June 4, 1986, the licensee failed to perform adequate post-modification testing, which allowed an improper wiring condition to exist for years before an inappropriate action by a NSO exposed the problem. Specifically, modification M12-2/3-83-14 was installed without confirmation that the equipment was in accordance with design documentation. The licensee generated IR 994101 to address this issue. Part of the corrective actions implemented by the licensee to address this issue included correcting the improper wiring and creating a training request to have enhanced post-maintenance testing requirement training given to all plant engineers, design engineers and maintenance work planners. Other planned corrective actions include evaluate whether EDG procedures need to have additional guidance regarding reactive load control to the NSO when operating the EDG during the undervoltage test. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program, this violation is being treated as and NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy.
(**NCV 05000237/2010003-07/05000249/2010003-07**; 2/3 Emergency Diesel Generator (EDG) Overvoltage during Division I Undervoltage)

This URI is closed. This event follow-up review constituted one sample as defined in IP 71153-05.

.2 <u>(Closed) Licensee Event Report (LER) 05000237/2009-004-00 "Unit 2 Shutdown Cooling System Isolation During Cooldown Due to Temperature Instrumentation Failure"</u>

On November 2, 2009, during shutdown for the Dresden Unit 2 Fall refueling outage, the Shutdown Cooling (SDC) System high temperature isolation logic actuated causing the SDC isolation valves to shut, resulting in a loss of cooling to the reactor vessel.
The apparent cause was a failure at the soldered thermocouple (TC) junction and the resistance temperature detector (RTD) element-to-lead solder joint attributed to corrosion and oxidation at the solder connection. The safety significance of this event is minimal as cooling flow was restored to the vessel in approximately 40 minutes.
Plant response and operation actions were consistent with plant procedures.
The licensee has submitted a License Amendment Request (LAR), dated February 4, 2010, to the NRC to modify SDC isolation logic from recirculation loop cold-leg temperatures to steam dome pressures, consistent with other industry BWR plants. The inspectors reviewed the licensee's corrective actions and had no further concerns. The inspectors did not identify a performance deficiency and there was no violation of regulatory requirements.

This LER is closed.

This event follow-up review constituted one sample as defined in IP 71153-05.

4OA5 Other Activities

.1 (Open) NRC TI 2515/177, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal and Containment Spray Systems (NRC Generic Letter 2008-01)"

As documented in Section 1R04, the inspectors confirmed the acceptability of the described licensee's actions. This inspection effort counts towards the completion of TI 2515/177, which will be closed on a later Inspection Report.

.2 Inspection of Procedures and Processes for Managing Fatigue (TI 2515/180)

a. Inspection Scope

The objective of this Temporary Instruction is to determine if the licensees' implementation procedures and processes required by 10 CFR Part 26, Subpart I, "Managing Fatigue" are in place to reasonably ensure the requirements specified in Subpart I are being addressed. The Temporary Instruction applies to all operating nuclear power reactor licensees but is intended to be performed for one site per utility. The inspector interfaced with the appropriate station staff to obtain and review station policies, procedures and processes necessary to complete all portions of this Temporary Instruction.

b. Findings

No findings of significance were identified.

.3 (Closed) Unresolved Item (URI) 05000237/2008002–02; 05000249/2008002–02: LPCI Room Cooler 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.

.4 (Closed) Unresolved Item (URI) 05000237/2009007-04: Potential Non-Conservative Fatigue Analysis

A concern was identified regarding a potential non-conservative analytical methodology described in NRC Regulatory Issue Summary (RIS) 2008-30, "Fatigue Analysis of Nuclear Power Plant Components." Specifically, the licensee's fatigue analysis of the feedwater nozzle did not consider six stress components as discussed in ASME Code, Section III, Subsection NB, Subarticle NB-3200. In RIS 2008-30, NRC staff requested that licensees who had received renewed licenses prior to the RIS issuance perform confirmatory analyses to demonstrate that the simplified analyses provide acceptable results. The inspectors opened this URI pending review of licensee's confirmatory analysis.

During this inspection period, the inspectors reviewed the licensee's confirmatory analysis for the feedwater nozzle that retained six stress components in accordance with the ASME Code, Analysis No: SIA File No. 0901269.303, Dresden 2 Feedwater Nozzle Confirmatory Analysis – Environmentally Assisted Fatigue Analysis of Feedwater Nozzle, Revision 0. In addition, inspectors reviewed the simplified fatigue analysis, Analysis No: SIA File No. EXLN-15Q-302, Dresden and Quad Cities – Environmental Assisted Fatigue Analysis – Updated Fatigue Analysis of Feedwater Nozzle, Revision 1.

Because Analysis No: SIA File No. 0901269.303 removed conservatism used in Analysis No: SIA File No. EXLN-15Q-302, the inspectors could not conclude that the simplified single stress analysis would have provided acceptable results.

However, because the confirmatory Analysis No: SIA File No. 0901269.303 demonstrated that the fatigue cumulative usage factor for the feedwater nozzle was acceptable for the period of extended operation, this unresolved item is closed. No finding of significance was identified and this item is closed.

.5 Operation of an Independent Spent Fuel Storage Facility Installation (ISFSI) at Operating Plants (60855.1)

a. Operations of an ISFSI

Inspection Scope

The inspectors observed and evaluated select licensee loading, processing, and transfer operations of the third canister during the licensee's dry fuel storage 2010 campaign to verify compliance with the applicable certificate of compliance (CoC) conditions, Technical Specifications (TS), and procedures. Specifically, the inspectors observed loading and independent verification of fuel assemblies, lifting of the transfer cask from the spent fuel pool, decontamination and surveying of the transfer cask, non-destructive evaluations of the lid to shell weld, hydrostatic testing, draining of water, vacuum drying, and transfer of the multi-purpose canister (MPC) from the transfer cask to the storage cask.

A tour was conducted of the East and West ISFSI pads to assess the condition of the ISFSI. The storage casks and pad were observed to be in good condition. No flammable or combustible materials were observed inside the ISFSI cask storage area, and inspectors reviewed evaluations of flammable materials outside the ISFSI cask storage area. Inspectors independently reviewed environmental radiation levels around the ISFSI, and reviewed the licensee's radiation monitoring program for the ISFSI.

In addition, the inspectors reviewed a number of condition reports and the associated follow-up actions since the last ISFSI inspection. The inspectors reviewed 72.48 screenings and evaluations and changes to the licensee's 72.212 report. Inspectors reviewed the licensee's fuel selection procedure and fuel selection documentation for MPC-68-260 to verify that TS criteria were being met. Inspectors reviewed the licensee's crane inspection documentation for compliance with industry standard ASME B30.2.

Enforcement Guidance Memorandum EGM-09-006 – Enforcement Discretion for Violations of 10 CFR Part 72, Implementation of Certificate of Compliance (CoC) Amendments to Previously Loaded Spent Fuel Casks

Current NRC regulations require that for 10 CFR Part 72 general licensees, any cask loaded with spent fuel must comply with the terms, conditions, and specifications (TCSs) of the CoC under which that cask was loaded. The NRC staff has determined that a change described by a later CoC amendment cannot be applied to a cask loaded under the original CoC or an earlier amendment without prior NRC approval, if such a change results in a change to the TCSs of the CoC under which the cask was loaded.

Therefore, under current regulations, general licensees that want to apply any changes from a later CoC amendment to a previously loaded cask must request an exemption from the NRC, pursuant to 10 CFR 72.7.

On September 15, 2009, the NRC issued a proposed rule for public comment entitled "License and Certificate of Compliance Terms," RIN 3150-AI09. The proposed rule will allow general licensees to implement, without prior NRC approval, changes described by a CoC amendment listed in 10 CFR 72.214 to a previously loaded cask, provided that the cask, after the changes have been applied, conforms to all TCSs of the updated CoC amendment. Partial or selective application of some of the authorized changes, but not others, would continue to require prior NRC approval through an exemption request.

Additionally, the NRC issued Enforcement Guidance Memorandum EGM-09-006 to address the general licensees who have implemented changes described by a CoC amendment to a previously loaded cask without prior NRC approval, prior to issuance of the proposed rule.

The licensee records indicated that canister loading operations had been completed in accordance with each Holtec HI-STAR and HI-STORM Amendment. However, the licensee had elected to the revise their 72.212 evaluation to implement the updated cask monitoring requirements for seventeen casks that had been placed on the East ISFSI pad.

At the time of the inspection, the licensee was found to be implementing the cask monitoring requirements for casks that had been loaded under HI-STORM Amendments 0, and 1, per the requirements of HI-STORM Amendment 1 as of May 2003 and HI-STAR Amendments 0, 1, and 2 per the requirements of HI-STAR Amendment 2 as of April 2002.

The inspectors reviewed a Dresden evaluation that had been performed of the CoC/TS changes that had been made from amendment to amendment. The licensee determined that the conditions of CoC HI-STAR Amendment 2 could be applied to the previously loaded casks and HI-STORM Amendment 1 could be applied to the previously loaded casks.

The 72.212 evaluation was completed and implemented within the 4 month time limit imposed by EGM-09-006. No significant findings were observed by the inspectors during the documentation review.

The NRC has determined that a potential violation of 10 CFR Part 72, Subpart K, was identified regarding Exelon Generation Company, which holds a general license pursuant to 10 CFR 72.210. The licensee has applied all of the changes described by CoC amendments listed in 10 CFR 72.214 to previously loaded casks prior to issuance of the proposed rule entitled "License and Certificate of Compliance Terms," RIN 3150-AI09, and without obtaining NRC approval.

Because the licensee has performed a 10 CFR 72.212 evaluation for its application of all of the changes described by HI-STAR Certificate Number 1008 Amendment 2 and HI-STORM Certificate Number 1014 Amendment 1 to the previously loaded casks by January 15, 2010, which is 4 months after the issuance of the proposed rule RIN 3150-AI09, the NRC is exercising enforcement discretion in accordance with

Section VII.B.6 of the NRC Enforcement Policy and, therefore, is not issuing any enforcement action for this potential violation. (EA 10-136)

The following table provides information required by EGM-09-006, on the Holtec HI-STORM 100 CoC 1014 and Holtec HI-STAR CoC 1008 cask number, date loaded and which amendment that the cask was loaded under:

Cask Serial No.	Canister Serial No.	Date Stored	CoC Amendment - Loaded	CoC Amendment - Current
HI-STAR 100-004	68F-005	7/12/2000	1008 - 0	1008 - 2
HI-STAR 100-005	68F-021	11/26/2001	1008 - 1	1008 - 2
HI-STAR 100-006	68F-029	1/15/2002	1008 - 1	1008 - 2
HI-STAR 100-007	68F-020	11/1/2001	1008 - 1	1008 - 2
HI-STORM 100-001	68F-006	6/16/2001	1014 - 0	1014 - 1
HI-STORM 100-002	68-008	7/17/2001	1014 - 0	1014 - 1
HI-STORM 100-003	68-016	9/5/2001	1014 - 0	1014 - 1
HI-STORM 100-004	68-009	8/15/2001	1014 - 0	1014 - 1
HI-STORM 100-009	68-017	9/18/2001	1014 - 0	1014 - 1
HI-STORM 100-010	68-018	10/1/2001	1014 - 0	1014 - 1
HI-STORM 100-011	68F-019	10/16/2001	1014 - 0	1014 - 1
HI-STORM 100-012	68-007	4/27/2002	1014 - 0	1014 - 1
HI-STORM 100-013	68-034	5/7/2002	1014 - 0	1014 - 1
HI-STORM 100-014	68F-027	10/25/2001	1014 - 0	1014 - 1
HI-STORM 100-027	68-035	5/17/2002	1014 - 0	1014 - 1
HI-STORM 100-036	68-037	6/4/2002	1014 - 0	1014 - 1
HI-STORM 100-037	68-036	5/28/2002	1014 - 0	1014 - 1

b. Findings

No findings of significance were identified.

.6 Operation of an ISFSI at Operating Plants (60855.1)

a. Control of Heavy Loads

Inspection Scope

During a weekly teleconference with the licensee's Byron Station and the NRC on February 9, 2010, the NRC inspectors noted concerns with the seismic analysis of the proposed free-standing stack configuration during transfer activities of the MPC from HI-TRAC transfer cask into HI-STORM storage cask.

Noting similarities in the analysis performed for the Dresden free-standing stack configuration, the licensee entered the concern into its corrective action program, AR 1028862, "Questions Raised by the NRC Regarding Cask Stackup," dated February 10, 2010. In addition, Exelon initiated AR 1031363, "NRC Questions on Byron Specific Calculation," dated February 12, 2010. As interim corrective action to support Dresden's dry fuel storage 2010 campaign, the licensee elected to design and install a physical seismic restraint to the stack configuration during transfer activities of the MPC from HI-TRAC transfer cask into HI-STORM storage cask.

The inspectors reviewed the licensee's design calculations associated with the physical restraint of the stack configuration during transfer activities of the MPC from the HI-TRAC transfer cask into the HI-STORM storage cask. In addition, the inspectors reviewed changes to the calculation that supported stability of the HI-TRAC transfer cask and the HI-STORM storage cask while on the low profile transporter if in the reactor building during a postulated design basis seismic event.

b. Findings

(1) Potential Safety Significance of Free-Standing Stack Configuration during MPC Transfer from HI-TRAC Transfer Cask into HI-STORM Storage Cask

Introduction: As part of review of the licensee's control of heavy loads associated with the design and installation of a physical restraint to the stack configuration during transfer activities of the MPC from the HI-TRAC transfer cask into the HI-STORM storage cask, the inspectors identified an unresolved item (URI) related to the potential safety significance of a free-standing stack configuration during a postulated design basis seismic event. Specifically, the licensee had utilized a free-standing stack configuration during prior dry fuel storage campaigns.

Description: During a weekly teleconference with the licensee's Byron Station and the NRC on February 9, 2010, the NRC inspectors noted concerns with the seismic analysis of the proposed free-standing stack configuration during transfer activities of the MPC from HI-TRAC transfer cask into HI-STORM storage cask. Noting similarities in the analysis performed for the Dresden free-standing stack configuration, the licensee entered the concern into its corrective action program, AR 1028862, "Questions Raised by NRC Regarding Cask Stackup," dated February 10, 2010. In addition, Exelon initiated AR 1031363, "NRC Questions on Byron Specific Calculation," dated February 12, 2010.

Corrective actions are being processed through Exelon AR 1031363 to ensure future safe handling of the cask. Based on final corrective actions, the safety significance of

the historical use of a free-standing stack configuration may need to be addressed, as noted in AR 1028862.

Pending NRC review of the licensee's associated corrective actions, this concern will be tracked as an unresolved item **(URI 05000237/2010003-08; 05000249/2010003-08)**.

4OA6 Management Meetings

.1 Exit Meeting Summary

On July 14, 2010, the inspectors presented the inspection results to Mr. S. Marik, 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:

- Unresolved item associated with the LPCI room cooler calculation deficiencies with Mr. T. Loch on April 29, 2010.
- Radiological Hazard Assessment and Exposure Control inspection with Mr. Tim Hanley and others on May 7, 2010.
- The results of the licensed operator requalification training program inspection with Mr. T. Hanley, Site Vice President, on May 28, 2010.
- Temporary Instruction 2515/180 Inspection of Procedures and Processes for Managing Fatigue with T. Hanley, Site Vice President, on June 18, 2010.
- The licensed operator requalification training biennial annual operating test results with Mr. M Otten, Operations Training Manager, via telephone on June 28, 2010.
- The ISFSI operational inspection conducted interim exit meetings on June 11, 2010, and June 30, 2010. The inspectors presented the inspection results to members of the licensee management and staff.

The inspectors confirmed that none of the potential report input discussed was considered proprietary. Proprietary material received during the inspection was returned to the licensee.

- The closure of Unit 2 NRC post-approval inspection for license renewal URI 05000237/2009007-04 with the Nuclear Oversight Manager, Mr. D. Leggett, and other members of the licensee's staff via telephone on June 30, 2010.
- The ISFSI control of heavy loads inspection conducted an interim exit with the Nuclear Oversight Manager, Mr. D. Leggett, and other members of the licensee's staff via telephone on June 30, 2010.

The inspectors confirmed that none of the potential report input discussed was considered proprietary. Proprietary material received during the inspection will be shredded.

4OA7 Licensee-Identified Violations

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

.1 Failure to Document a Post-Maintenance Test Failure

On April 16, 2010, a plant engineer performed an ultrasonic examination (UT) of the 3B core spray pump suction piping and identified that an air bubble existed in the piping. The purpose of the UT was to determine post-maintenance operability. The test procedure required an examination within one foot of the vent valve on the suction piping. The engineer found no air in that location and called the test satisfactory. However, the engineer performed additional UT examination of the piping farther away from the vent valve and found air in that location. The licensee declared the 3B core spray subsystem operable after the completion of the UT examination. The engineer neither evaluated the size of the air bubble for operability nor contacted operations personnel or wrote an issue report so that operability could be evaluated until April 19, 2010. The air bubble was eventually documented in IR 1058558, "Air Ribbon Discovered in 3B Core Spray Suction," and was determined to not impact the operability of the 3B Core Spray subsystem. The fact that the air bubble was identified and that no IR was written until three days later was documented in IR 1058966, "UT Inspection Process Improvement." Exelon Procedure LS-AA-125, "Corrective Action Program (CAP) Procedure," Revision 14, Section 4.1.2, states, in part, that at any time any question of current or past operability arises, then initiate an Issue Report. Air in the suction piping of emergency core cooling system piping raised a question of operability. This violation was not greater than Green because the air bubble did not result in the inoperability of the 3B core spray subsystem. The adequacy of the PMT procedure as addressed in IR 1057049, "Air Void U2 HPCI Discharge Piping Above Acceptance Criteria."

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

T. Hanley, Site Vice President
S. Marik, Station Plant Manager
H. Bush, Radiation Protection Manager
D. Doggett, Emergency Preparedness Coordinator
D. Dura, Dry Fuel Storage Project Manager
B. Finlay, Security Manager
J. Fox, Design Engineering
D. Glick, Shipping Specialist
J. Griffin, Regulatory Assurance - NRC Coordinator
D. Gronek, Operations Director
J. Hansen, Corporate Licensing
K. Hunter, Dry Fuel Storage Engineer
M. Johnson, Operations Training
L. Jordan, Training Director
R. Kalb, Chemistry
P. Karaba, Maintenance Director
J. Kish, Engineering Programs
M. Kucek, Nuclear Over Sight Assessor
D. Leggett, Nuclear Oversight Manager
R. Laburn, Radiation Protection
T. Loch, Senior Manager, Design Engineering
P. Mankoo, Chemistry Supervisor
M. Marchionda, Regulatory Assurance Manager
P. O'Connor, Licensed Operator Requalification Training Lead
M. Otten, Operations Training Manager
M. Overstreet, Manager of Radiation Engineering
C. Podczerwinski, Maintenance Rule Coordinator
P. Quealy, Emergency Preparedness Manager
J. Reda, Mechanical/Structural Design Engineering Manager
E. Rowley, Chemistry
R. Rybak, Regulatory Assurance Engineer
J. Sipek, Engineering Director
N. Starcevich, Radiation Protection Instrumentation Coordinator
J. Strmec, Chemistry Manager
S. Vercelli, Work Management Director

Nuclear Regulatory Commission

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

IEMA

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

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000237/2010003-01; 05000249/2010003-01	URI	High Pressure Coolant Injection (HPCI) Booster Pump Bearing Oil Levels (Section 1R04)
05000237/2010003-02; 05000249/2010003-02	NCV	LPCI Room Heat-up Calculation Deficiencies (Section 1R07.2)
05000237/2010003-03; 05000249/2010003-03	URI	Undocumented Technical Basis for change to EOP ATWS Mitigation Strategy (Section 1R11.4)
05000237/2010003-04; 05000249/2010003-04	FIN	Failure to Monitor Unit 3 Drywell Temperature (Section 1R15.1)
05000237/2010003-05; 05000249/2010003-05	NCV	Failure to Provide An Adequate Procedure for Several Instrument Maintenance Surveillance Tests (Section 1R22)
05000237/2010003-06; 05000249/2010003-06	FIN	Failure To Perform An Adequate Inspection of Circulating Water Valve 3-4402-C (Section 4OA2.6)
05000237/2010003-07; 05000249/2010003-07	NCV	2/3 Emergency Diesel Generator (EDG) Overvoltage during Division I Undervoltage (Section 4OA3.1)
05000237/2010003-08; 05000249/2010003-08	URI	Potential Safety Significance of Free-Standing Stack Configuration during MPC Transfer from HI-TRAC Transfer Cask into HI-STORM Storage Cask (Section 4OA5.6)
2515/177	TI	Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal and Containment Spray Systems (NRC Generic Letter 2008-01) (Section 1R04)

Closed

05000237/2010003-02; 05000249/2010003-02	NCV	LPCI Room Heat-up Calculation Deficiencies (Section 1R07.2)
05000237/2010003-04; 05000249/2010003-04	FIN	Failure to Monitor Unit 3 Drywell Temperature (Section 1R15.1)
05000237/2010003-05; 05000249/2010003-05	NCV	Failure to Provide An Adequate Procedure for Several Instrument Maintenance Surveillance Tests (Section 1R22)
05000237/2010003-06; 05000249/2010003-06	FIN	Failure To Perform An Adequate Inspection of Circulating Water Valve 3-4402-C (Section 4OA2.6)

05000237/2010003-07; 05000249/2010003-07	NCV	2/3 Emergency Diesel Generator (EDG) Overvoltage during Division I Undervoltage (Section 4OA3.1)
05000237/2008002–02; 05000249/2008002–02	URI	LPCI Room Cooler Calculation Deficiencies and Discrepancies (Section 1R07)
05000237/2009005-04 05000249/2009005-04	URI	2/3 Emergency Diesel Generator (EDG) Overvoltage During Division 1 Undervoltage (Section 4OA3.1)
05000237/2009-007-04	URI	Potential Non-Conservative Fatigue Analysis (Section 4OA3.4)
05000237/2009-004-00	LER	Unit 2 Shutdown Cooling System Isolation During Cooldown Due to Temperature Instrument Failure (Section 4OA3.2)

Discussed

2515/180	TI	Managing Fatigue (Section 4OA5.3)

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)

- DOA 6500-12, “Low Switchyard Voltage,” Revision 20
- WC-AA-101, “On-Line Work Control Process,” Revision 17
- OP-AA-108-107-1001, “Station Response to Grid Capacity Conditions,” Revision 3
- OP-AA-108-107-1002, “Interface Procedure Between ComEd/PECO and Exelon Generation (Nuclear/Power) for Transmission Operations,” Revision 5
- EGC/AmerGen 60-Day Response to NRC Generic Letter 2006-02, “Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power”
- IR 1075931, “Unplanned DOA 0010-02 Entry”
- IR 1075941, “Entered DOA 0040-03 due to Loss of 34KV L1263”
- IR 1075930, “Procedure Conflict Between DOA 10-02 and OP-AA-108-111-1001”
- OP-AA-108-111-1001, “Severe Weather and Natural Disaster Guidelines,” Revision 5
- DOA 0010-02, “Tornado Warning/Severe Winds,” Revision 14

1R04 Equipment Alignment (71111.04)

- DOP 1500-M1, “Unit 3 LPCI and Containment Cooling Valve Checklist,” Revision 31
- DOP 1500-E1, “Unit 3 LPCI and CCSW System Electrical,” Revision 13
- DOP 0900-E1, “Unit 2(3) Control Room Panels,” Revision 20
- DOP 1500-01, “Preparation of Low Pressure Coolant Injection for Automatic Start,” Revision 15
- M-360, “Diagram of L.P. Coolant Injection System,” Revision VO
- DOP 0900-E1, “Unit 2(3) Control Room Panel,” Revision 20
- DOP 1500-01, “Preparation of Low Pressure Coolant Injection for Automatic Start,” Revision 15
- DOP 2300-M1/E1, “Unit 2 HPCI System Checklist,” Revision 38
- DOP 2300-01, “High Pressure Coolant Injection (HPCI) System Standby Operation,” Revision 44
- MA-AB-734-448, “HPCI Booster Pump Maintenance,” Revision 0
- M-51, “Diagram of High Pressure Coolant Injection Piping,” Revision CM
- ISI-510, Sheet 2, “System Pressure Test Walkdown Isometric H.P. Coolant Injection Piping,” Revision D
- ISI-513, Sheet 1, “System Pressure Test Walkdown Isometric Emergency Core Cooling Torus Ring Header,” Revision B
- M-1151A, Sheet 10, “High Pressure Coolant Injection System Dresden Unit #2,” Revision F
- M-1151A-2, Sheet 1, “Computer Piping Isometric High Pressure Coolant injection System Dresden Nuclear Station Unit 2,” Revision 0
- M-1151A, Sheet 9, “High Pressure Coolant Injection System Dresden Unit #2,” Revision E
- IR 1081187, “NRC Concerns – MCC 29-1 Light, U2 HPCI Oil Sightglass”
- IR 1081578, “NOS IDs IR 1081187 Contains Incorrect Information”
- IR 957698, “MCC 29-1 'bus Energized' Light Is Not Lit”
- IR 1081750, “NRC Concern – U2 HPCI Booster Pump Oil Level”

- IR 1085361, “Unit 3 HPCI Booster Pump Sight Glass Markings”
- IR 1085422, “NRC Inspector Questions HPCI Booster Pp Oil Level Impact”
- IR 161709, “Condensate Booster Thrust Bearing Damaged”
- EC 371153, “NRC GL 2008-01 HPCI System Evaluation,” Revision 3
- IR 1084147, “U2 HPCI Discharge Piping Air Void Increase”
- IR 1081055, “NRC Question: GL 2008-01 HPCI Piping”

1R05 Fire Protection (71111.05)

- WO 1237655-01, “D2/3 An TSTR Linear Thermal Detection Surv (Rx Bldg and 2/3 Crib House)”
- Fire Hazards Analysis Sections 4.2 and 6.3
- IR1070316, “NRC Identifies XL3 Testing Oversight”
- IR 1056773, “Oil Around 3A Starting Air Compressor”
- IR 1045393, “Enhancement for Cardox System Operability Testing”
- IR 1057594, “NRC-Identified Scaffolding and Temp Lighting in U3 DG Room”
- WO 1141619, “D3 18M TSTR/COM EMER D/G CARDOX SYSTEM MAINTENANCE TEST”
- “Low Pressure CO2 Fire Extinguishing System Functional Operability and Full Discharge Tests for the Diesel Generator and Day Tank Rooms, Dresden Nuclear Station, Commonwealth Edison Company,” Revision 0, dated May 31, 1989
- Dresden Station Units 2 and 3 Commonwealth Edison Company, Fire Protection Program Documentation Package, Volume 8, “NFPA Code Conformance”
- Dresden Station Units 2 and 3 Commonwealth Edison Company, Fire Protection Program Documentation Package, Volume 9, “NFPA Code Conformance (continued)”
- IR 1063187, “NRC has Question on CO2 Pressure in U2/3 EDG Room”

1R06 Flooding (71111.06)

- IR 01071699, “U3 CCSW Vault Door Latch Spring Needs Adjustment”
- WO 01168289, “18M U3 CCSW PMP VAULT FLR DRN CK VLV”
- WO 01127379, “18M U3 CCSW PMP VAULT Penetration Seal Testing”
- WO 01191674, “18M US CCSW PMP VAULT Water Tight Door Leak Test”
- IR 1060187, “Water Reappeared in Manhole #2”
- IR 1060198, “Degraded Cable Tray in Man Hole #2 – one support was very rusty and supported by a 4x4 piece of wood”
- IR 1054298, “Water Discovered in Cable Manhole/Vault #2”
- IR 1081708, “Submerged Cables Found in Manholes”
- IR 975308, “NRC Concerns on Submerged Electrical Cables”
- IR 962651, “Need to Remove Cable/Duct Seals”

1R07 Annual Heat Sink Performance (71111.07)

- ER-AA-340-2000, Rev 5, “Balance-Of-Plant Heat Exchanger Inspection, Testing, and Maintenance”
- Generic Letter 89-13, Dresden Station Program Basis Document
- WO 99182584 D2 6Y PM TBCCW (8/20/07) 2B HX
- WO 99063274 D2 6Y PM TBCCW (3/22/05) 2A HX
- IR 01073314, “U2TBCCW inlet temp high out of tolerance”
- IR 01001567, “U2 TBCCW temp high”
- IR 00993445, “U2 TBCCW HX inlet header erroneous"

- BSA-D-00-01; Dresden 2/3 ECCS Room Temperature Response with Loss of Room Cooler; Revisions 1 and 1A
- DRE97-0214; Reactor Building Post-LOCA Temperature Analysis; Revision 1
- DR-721-M-001; Maximum Room Temperature for the Corner Pump Rooms Following a Postulated Event Outside these Rooms; Revision 1
- AR00742158; Calculation Basis not Specifically Described; February 28, 2008
- AR00763663; EPU Project did not Evaluate the Effect of Higher Overpressure; dated April 15, 2008
- AR00777935; 2008-01 NRC Quarterly Exit URI Documentation; May 21, 2008
- AR00883207; Non-Conservative Model used for EQ Temps; February 20, 2009
- AR01042748; Non-Conservative Heat Load used in Area Temp Calculation; March 15, 2010
- AR01058534; Gothic 7 Computer Code may Affect Station Calcs; April 19, 2010
- AR01055863; Add'l Proof Needed to Support Worst Case was Evaluated; April 13, 2010
- AR01060243; Additional Information Regarding IR 1055863; April 22, 2010
- EQ Binder for LPCI Corner Room

1R11 Licensed Operator Requalification Program (71111.11Q and B)

- TQ-AA-150; Operator Training Programs; Revision 4
- TQ-AA-201; Examination Security and Administration; Revision 11
- OP-AA150-102; NRC Active License Maintenance; Revision 9
- LORT Requal Operating Test Weeks 2 and 5, 2010
- LORT Requal Written Exam Weeks 1 and 2, 2009
- LORT 2010 Curriculum Review Committee Meeting Minutes 3/26/10 and 2/12/10
- FASA Self-Assessment Report, Dresden Station 2010 Per 71111.11 Inspection Assignment #870609
- TQ-AA-150-F06; Simulator Evaluation Form – Shift Manager; 5/26/10 (2 forms)
- TQ-AA-150-F08; Simulator Evaluation Form – Individual; 5/26/10 (8 forms)
- TQ-AA-150-F09; Simulator Evaluation Form – Crew; 5/26/10 (2 forms)
- TQ-AA-306; Simulator Management; Revision 1
- TQ-AA-306-JA-02; Simulator Testing Report Update [03/19/09 – 03/18/10]; 04/22/2010
- Computer Real Time Test [11/04/2009]; Revision 1
- Steady State Test [11/03/2009]; Revision 2
- Normal Operating Test – DGP 02-02 Unit Shutdown [11/17/2009]; Revision 134
- TT6; Turbine Trip from a Power Level Such That a SCRAM Does Not Occur [11/02/2009]; Revision 9
- TT7; Maximum Rated Power Ramp [11/08/2009]; Revision 8
- TT8; LOCA with a Loss of Off-Site Power [10/08/2009]; Revision 7
- TT9; Maximum Size Unisolable Steam Line Break [11/08/2009]; Revision 9
- CCSWPAOC; 2A (B, C, D) CCSW Pump Over-current Trip [04/30/2009]; Revision 0
- CWMSCDEP; Traveling Screen Deposition of Solids [10/07/2009]; Revision 0
- HP1; Condenser Tube Leak [12/19/2009]; Revision 0
- HP2; Cond Demin Clogging Percentage [10/13/2009]; Revision 0
- IC1VBN; IC Valve 2-1301-1 Binds 0%-100% [05/12/2009]; Revision 0
- RADCCSW1; CCSW HX 1 Pinhole Tube Leak [10/07/2009]; Revision 0
- SPCMOCA; SBLC Pump A (B) Trip [10/08/2009]; Revision 0
- Simulator Review Board Minutes; 08/2008, 02/2009, 06/2009, 10/2009, 12/2009, 02/2010
- TQ-AA-150; Operator Training Programs; Revision 4

- Remedial Training Packages; various dates between March 2008 and April 2010
- a. Crew/Individual Simulator Evaluation Forms for “Failures” and “Pass with Remediation”
- b. Remedial Training Notification and Action on Failure packages
- c. Performance Review Committee Data Sheets (Back to Back OBE failures)
- DOA-01; Terminate and Prevent Injection (Hard Card); Attachment A; Revision 48
- OP-DR-103-102-1002; Strategies for Successful Transient Mitigation; Revision 4
- BWR Owners’ Group Emergency Procedure and Severe Accident Guidelines; Appendix B: Technical Basis; Revision 2
- NEDO-32164; Mitigation of BWR Core Thermal-Hydraulic Instabilities in ATWS; December 1992

1R12 Maintenance Effectiveness (71111.12)

- WO 1323537-01, “D3 1M TS Alt 125VDC Station Battery Inspection”
- IR 1066376, “Loose Bolts on Alternate Battery Rack”
- IR 01066343, “Corrosion on U3 125VDC Alt Batt Cell #31”

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

- WC-AA-101, “On-Line Work Control Process,” Revision 17
- WC-AA-1004, “On-Line Maintenance for Limiting Condition for Operation of Systems or Components,” Revision 4
- ER-AA-600-1042, “On-Line Risk Management,” Revision 6
- OP-AA-108-117, “Protected Equipment Program,” Revision 0
- Dresden Operations Department Policy No. 02, “Protected Equipment/Pathway Policy,” September 24, 2009
- DOP 6900-08, “125 VDC and 250 VDC Portable Battery Charger Use,” Revision 3
- ER-AA-600-1042, “On-Line Risk Management,” Revision 6
- WC-AA-101, “On-Line Work Control Process,” Revision 17
- WC-AA-101-1004, “On-Line Maintenance for Limiting Condition for operation of Systems or Components,” Revision 4
- IR 1062551, “NOS ID Paragon Program Execution Issues”

1R15 Operability Evaluations (71111.15)

- IR 01050101, “NOS IDS Op Eval 10-002 Did not Address all Required Aspects”
- IR 01053502, “NRC Questions Regarding IR 01053502”
- Design Analysis No. DRE03-0013, “Low Pressure Coolant Injection System Combined DBD and DP Calculation,” Revision 000
- EC 379781, “Evaluation of air identified in U2 HPCI Discharge line”
- IR 1058966, “UT Inspection Process Improvement”

1R18 Plant Modifications (71111.18)

- EC 353371, Revise the Setpoints for the EDG Day Tank Level Switch,” Revision 001
- NED-I-EIC-0083, “Diesel Generator Fuel Oil Day Tank Level Loop Accuracy Calculation,” Revision 000C
- IR 806157, “U2 DG Fuel Oil Day Tank Level Less Than Required”
- IR 1062257, “U3 EDG Diesel Day Tank Level Hi Alarm 903-8 G-7”
- IR 1062357, “Unexpected Alarm: U2/3 EDG Day Tank High Level”
- IR 1065734, “NRC CDBI: Vulnerability on EDG Fuel Oil Day Tank Setpoint”
- DOS 6600-14, “Diesel Oil Transfer Pump Operation and Fuel Consumption Test,” Revision 15

1R19 Post-Maintenance Testing (71111.19)

- IR 01021725, “U3 DGCW Flanges Less than B16.5 Minimum Thickness”
- IR 01027411, “2/3 EDG Htx Needs to be Inspected”
- EC 394142, “Input on Replacement of the Diesel Generator Cooling Water Flanges/Piping from Htx’s to the System Connection”
- ER-AA-335-015, “VT-2 Visual Examination,” Revision 9
- CC-AA-407, “Maintenance Specification: Evaluation and Repair of Piping and Equipment Flanges,” Revision 2
- DMP 1501-04, “Containment Cooling Service Water (CCSW) Pump Maintenance,” Revision 16
- MA-AA-716-230-1002, “Vibration Analysis/Acceptance Guideline,” Revision 2
- WO 01346169-01, “TS/Repair Loss of U3 ESS UPS Normal AC Supply”
- DOP 6800-01, “Essential Service System,” Revision 28
- DAN 902(3)-8 E-8, “ESS UPS on DC or Alternate AC,” Revision 08
- DAN 902(3)-8 F-8, “ESS UPS Trouble,” Revision 05
- DOS 0040-09, “Unit 3 Operating Power Sources and Distribution,” Revision 25
- Drawing 12E-3811B, “Uninterruptible Power Supply Panel 903-63,” Revision H
- IR 1056582, “3A LPCI PMP Seal Leak-off Line Crack”
- WO 1125680-02, “Op Perform PMT on 3A LPCI Motor”
- IR 1057262, “Procedure Enhancement Opportunity”

1R22 Surveillance Testing (71111.22)

- DOS 2300-07 Rev 41
- HPCI fast initiation test
- IR 00858582, “NRC SRI concerns on U2 HPCI flow indications”
- IR 00858112, “Unanticipated U2 HPCI flow while running U3 HPCI IST”
- WO 01271444, “U2 QTR TS HPCI Pump Oper Test and IST Surv” (12/15/09)
- WO 01299401, “U2 QTR TS HPCI Pump Oper Test and IST Surv”
- DOS 6600-14, “Diesel Oil Transfer Pump Operation and Fuel Consumption Test,” Revision 15
- IR 1059004, “Effect of EDG Frequency Tolerance on Fuel Oil Storage”

1EP6 Drill Evaluation (71114.06)

- Dresden 2010 Off-Year Exercise Evaluation Report

2RS1 Radiological Hazard Assessment and Exposure Controls

- Self-Assessment Report; Radiological Hazard and Exposure Controls; dated February 24, 2010
- Self-Assessment Report; Radiation Protection Dosimetry Check-In; dated August 17, 2009
- AR 00905751; Contaminated Sludge Vacuumed into Truck; dated April 3, 2009
- AR 00995554; Unit 3 Steam Sensitive Area Dose Rates Higher than Expected; dated November 19, 2009
- AR 00998898; Wrench with Purple Paint in Men’s Locker Room; dated November 28, 2009
- AR 00992680; Worker Received Dose Alarm in Shutdown Cooling Pump Room; dated November 12, 2009
- RP-AA-460-002; Additional High Radiation Exposure Control; Revision 0
- Semi-Annual Inventory and Leak Testing of Radioactive Sources; dated January 25, 2010
- RP-AA-800; Control, Inventory and Leak Testing of Radioactive Sources; Revision 6

- RP-AA-460, Attachment 8; Approval for HRA/LHRA Deviations; dated various periods in 2009 and 2010
- RP-AA-301; Radiological Air Sampling Program; Revision 3
- TID-2008-05 and 2009-001; Alpha Program Evaluation/Prospectus; dated October 20, 2008, and December 2009
- RP-AA-302; Determination of Alpha Levels and Monitoring; Revision 2
- Radiological Survey Results and Air Samples Analyses for Various Unit 2 Refuel Outage Activities; dated October – November 2009
- RP-AA-210; Dosimetry Issue, Usage and Control; Revision 17
- RP-AA-503; Unconditional release Survey Method; Revision 2
- RP-AA-460; Controls for High and Locked High Radiation Areas; Revision 19
- DFP 0800-39; Control of Material/Equipment Hanging in Units 2/3 Spent Fuel Pools; Revision 16
- Spent Fuel Pool Material Log/Inventory; dated January 2010

2RS8 Radioactive Solid Waste Processing and Radioactive Material Transportation

- AR 01063216; Failure of Shipping Container in Transit; dated April 29, 2010
- PacTec, Inc. Overpac Design Specifications and Operating Instructions; Undated
- PacTec, Inc. Certificate of Conformance for IP-1 Packaging; Item No. OP2199-18VF12Z (SP); dated January 28, 2010
- PacTec, Inc. Statement Regarding IP-1 Compliance; dated May 6, 2010
- Uniform Low Level Waste Manifest for Dresden Shipment No. DW-10-017; dated April 27, 2010 (and associated shipment survey information)

4OA1 Performance Indicator Verification (71151)

- IR 00926605, “Oil Leak on the 2/3 DG turbo Lube Oil Y-Strainer”
- IR 00937460, “IR 00926605 was a Maint Rule Functional Failure”
- IR 00932854, “OPS Screening for Reportability – EDG Wiring Deficiency”
- IR 00929002, “MR Function Z66-5 Not Tracked Per Performance Criteria”
- MSPI Derivation Reports U2/U3 April 2009 through March 2010
- PI Summary Reports U2/U3 April 2009 through March 2010
- IR 1081446, “MSPI Comments Not Entered per NEI 99-02”
- NEI 99-02, “Regulatory Assessment Performance Indicator Guideline,” Revision 6
- CY-DR-110-200; Plant System Sampling; Revision 20
- CY-AA-130-3010; Dose Equivalent Iodine Determination; Revision 2
- CY-AA-130-3010-F03; Dose Equivalent Iodine Determination; Calculations and Gamma Spectroscopy Results (selected periods between April 2009 and April 2010

4OA2 Identification and Resolution of Problems (71152)

- IR 01074245, “2B CRD PP Discharge Manual Valve Leakby”
- IR 01073925, “OWA – 2B CRD Pump Disch VLV Unable To Be Repaired On Line”
- OP-AA-102-103, “Operator Work-Around Program”
- DAN 902(3) – 5 B-2 Rev 08, “Rod Drive PP Trip”
- DOA 0300-01 Rev 21, “Control Rod Drive System Failure”
- DOP 0300-01 Rev 43, “Control Rod Drive System Start Up And Operation”
- IR 986078, “U3 125V Cell Voltage More Than .05V Below Average Voltage”
- IR 1059828, “U3 125VDC Battery Cell Voltage is Below the Avg Cell Voltage”

4OA3 Follow-Up of Events and Notices of Enforcement Discretion (71153)

- IR 1085681, “Historical Reportability Eval. for 2/3 EDG Wiring Errors”
- IR 994101, “2/3 Emergency Diesel Generator (EDG) Voltage Transient”
- DOS 6600-06, “Bus Undervoltage and ECCS Integrated Functional Test for Unit 2/3 Diesel Generator to Unit 2,” Revision 46 and Revision 47

4OA5 Inspection of Procedures and Processes for Managing Fatigue (TI 2515/180)

- LS-AA-119; Fatigue Management and Work Hour Limits; Revision 8
- LS-AA-119-1001; Fatigue Management; Revision 0
- LS-AA-119-1002; Scoping of Work Hour Limits; Revision 0
- LS-AA-119-1003; Calculating Work Hour; Revision 0
- LS-AA-119-1004; Reviews and Reporting; Revision 0
- LS-AA-119-1005; Contractor/Vendor Compliance with Fatigue Management and Work Hour Limits; Revision 0
- IR01029105, “Air void identified on U2 HPCI discharge piping above torus”
- IR01057049, “Air void U2 HPCI discharge piping above acceptance criteria”
- EC 379781, Evaluation of air identified in U2 HPCI discharge line, Revision 0
- FAI/08-70, Gas-voids pressure pulsations program, dated 09/03/08
- IR01081055, “NRC Question: GL 2008-01 HPCI piping”

4OA5.4 Other Activities

- Analysis No: SIA File No. 0901269.301; Dresden 2 Feedwater Nozzle Confirmatory Analysis – Design Inputs and Methodology for ASME Code Fatigue Usage Analysis of Feedwater Nozzle; Revision 0
- Analysis No: SIA File No. 0901269.302; Dresden 2 Feedwater Nozzle Confirmatory Analysis – Thermal / Mechanical Stress Analysis of Feedwater Nozzle; Revision 0
- Analysis No: SIA File No. 0901269.303; Dresden 2 Feedwater Nozzle Confirmatory Analysis – Environmentally Assisted Fatigue Analysis of Feedwater Nozzle; Revision 0
- Analysis No: SIA File No. EXLN-15Q-302; Dresden and Quad Cities – Environmental Assisted Fatigue Analysis – Updated Fatigue Analysis of Feedwater Nozzle; Revision 1

4OA5.5 Other Activities

- 72.48-421, “Implementation of Certificate of Compliance Amendments to Previously Loaded Spent Fuel Storage Casks,” Revision 0
- 72.212 Evaluation Change Request, “ East ISFSI Section 1.0, 1.3.9 (New) & 1.4,” January 15, 2010
- 72.212, “Evaluation Report East ISFSI,” Revision 4
- 72.212, “Evaluation Report West ISFSI,” Revision 2
- “Assist NSO and Common Unit Daily Surveillance Log”, Revision 71
- DAP 07-54, “Administrative Controls for the ISFSI,” Revision 4
- DM 5800-18, “Load Handling of Heavy Loads and Lifting Devices,” Revision 19
- DRE06-0036, “West ISFSI Fire Hazards Analysis,” Revision 0
- DRP 6021-22, “HI-STORM Radiation Survey,” June 4, 2010
- DRP 6021-23, “HI-STAR Radiation Survey,” Revision 1
- DRS 6021-33, “ISFSI Radiation Survey,” December 15, 2009
- DTP 67, “Fuel Selection and Documentation for Fuel Cask Loading,” Revision 8
- IR00825517, “NOS Identifies M&TE Not Entered into Passport,” April 2, 2008

- IR00859514, “HI-STORM Fuel Cask Vents Blocked with Ice,” December 22, 2008
- IR00859843, “HI-STORM Cask Air Ducts Blocked,” December 23, 2008
- IR00888276, “RBOC Main Hoist Wire Rope is Worn,” March 3, 2009
- IR00982205, “Control of Loaded Spent Fuel Casks Under Different CoC Amendments,” November 20, 2009
- IR01042170, “NOS Identifies Gasoline Storage Tank Needs Paint, Signage, Possible Berm,” August 17, 2009
- IR01068467, “Byron NRC Issues with 72.212 Initial Issuance,” May 12, 2010
- “Fuel Move Sheet MPC-68-309” May 10, 2010
- “Fuel Selection Package No: DRE-0043,” March 8, 2010
- NO-AA-10, “Quality Assurance Topical Report,” Revision 84
- Work Order 01320434, “Reactor Building Overhead Crane Inspection,” April 5, 2010

4OA5.6 Other Activities

- DRE00-0100; Simplified Stability Analysis for HI-TRAC and HI-STORM Components; Revision 002B
- DRE10-0006; Structural Evaluations Associated with the Cask Stack-Up Restraint System; Revision 0
- DRE-009 Structural Evaluation of Seismic Restraint Forces on Dresden HI-TRAC & HI-STORM
- DRE10-0010; Evaluation of RB Trackway at Elev. 517’6” for the Loads Associated with Dry Casks; Revision 0
- AR 1070952; NRC Questions Regarding DRE10-0006; dated May 7, 2010
- AR 1076634; NRC Questions Regarding DRE10-0009; dated May 14, 2010

LIST OF ACRONYMS USED

AC	Alternating Current
ACE	Apparent Cause Evaluation
ADAMS	Agencywide Document Access Management System
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
AVR	Automatic Voltage Regulator
BWROG	BWR Owners Group
BWR	Boiling-Water Reactor
CAP	Corrective Action Program
CCSW	Containment Cooling Service Water
CFR	Code of Federal Regulations
CoC	Certificate of Compliance
CRD	Control Rod Drive
DC	Direct Current
DNMS	Division of Nuclear Materials Safety
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
EACE	Equipment Apparent Cause Evaluation
DW	Drywell
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EPD	Electronic Personal Dosimeter
EPG	Emergency Procedure Guidelines
EPU	Extended Power Uprate
EQ	Environmental Qualification
°F	Degrees Fahrenheit
FW	Feedwater
HEPA	High-Efficiency Particulate Air
HP	Health Physics
HPCI	High Pressure Coolant Injection
HRA	High Radiation Areas
ISFSI	Independent Spent Fuel Storage Installation
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IR	Inspection Report
IR	Issue Report
ISI	Inservice Inspection
IST	In-Service Testing
JPM	Job Performance Measure
KVAR	KiloVolt-Ampere-Reactance
LAR	License Amendment Request
LER	Licensee Event Report
LHRA	Locked High Radiation Area
LOCA	Loss of Coolant Accident
LORT	Licensed Operator Requalification Training
LPCI	Low Pressure Coolant Injection
MCID	Materials Control ISFSI and Decommissioning
MSPI	Mitigating Systems Performance Index
MOV	Motor-Operated Valve

MPC	Multi-Purpose Canister
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulations
NSO	Nuclear Station Operator
OWA	Operator Workaround
PARS	Publicly Available Records System
PI	Performance Indicator
P&ID	Piping and Instrumentation Diagram
PI&R	Problem Identification and Resolution
PM	Planned, Preventative Maintenance, or Post-Maintenance
PMT	Post-Maintenance Testing
PWR	Pressurized-Water Reactor
RCA	Radiologically Controlled Area
RCIC	Reactor Core Isolation Cooling
RCS	Reactor Coolant System
RIS	Regulatory Issue Summary
RP	Radiation Protection
RPM	Radiation Protection Manager
RPV	Reactor Pressure Vessel
RTD	Resistance Temperature Detector
RWP	Radiation Work Permit
SAT	Systems Approach to Training
SBLC	Standby Liquid Control
SDC	Shutdown Cooling
SDP	Significance Determination Process
SSC	Structures, Systems, and Components
TC	Thermocouple
TCS	Terms, Conditions, and Specifications
TI	Temporary Instruction
TS	Technical Specification
TSO	Transmission System Operator
U3	Unit 3
UFSAR	Updated Final Safety Analysis Report
USAR	Updated Safety Analysis Report
URI	Unresolved Item
UT	Ultrasonic Examination
VHRA	Very High Radiation Areas
VOM	Volt-Ohm Meter
WO	Work Order
WRGM	Wide Range Gas Monitor

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Sincerely,

/RA/

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

Docket Nos. 50-237; 50-249; 72-037
License Nos. DPR-19; DPR-25

Enclosure: Inspection Report 05000237/2010-003; 05000249/2010-003; 07200037/2010-001 w/Attachment: Supplemental Information

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Letter to M. Pacilio from M. Ring dated July 30, 2010

SUBJECT: DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3 AND INDEPENDENT SPENT FUEL STORAGE INSTALLATION INTEGRATED INSPECTION REPORT 05000237/2010-003; 05000249/2010-003; 07200037/2010-001

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