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

November 7, 2011

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, INTEGRATED INSPECTION REPORT 05000237/2011004; 05000249/2011004

Dear Mr. Pacilio:

On September 30, 2011, 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 October 12, 2011, with Mr. D. Czufin, 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, the NRC has also identified one NRC-identified and two self-revealed findings of very low safety significance. Two of the three 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 2.3.2 of the NRC Enforcement Policy.

If you contest the subject or severity of any of these non-cited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the 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 Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

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

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

Enclosure: Inspection Report 05000237/2011-004; 05000249/2011-004
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	05000237; 05000249
License Nos:	DPR-19; DPR-25
Report No:	05000237/2011-004; 05000249/2011-004
Licensee:	Exelon Generation Company, LLC
Facility:	Dresden Nuclear Power Station, Units 2 and 3
Location:	Morris, IL
Dates:	July 1 through September 30, 2011
Inspectors:	C. Phillips, Senior Resident Inspector D. Meléndez-Colón, Resident Inspector J. Draper, Reactor Engineer R. Orlikowski, Project Engineer J. Corujo-Sandín, Reactor Engineer L. Jones, Reactor Engineer
Approved by:	M. Ring, Chief, Branch 1 Division of Reactor Projects

Enclosure

TABLE OF CONTENTS

SUMMARY OF FINDINGS

IR 05000237/2011-004, 05000249/2011-004; 07/01/2011 – 09/30/2011; Dresden Nuclear Power Station, Units 2 & 3; Flooding, Plant Modifications, Post-Maintenance Testing.

This report covers a three-month period of inspection by resident inspectors. One Green finding was identified by the inspectors and two Green findings were self-revealed. Two of the 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: Mitigating Systems

- Green. A finding of very low safety significance and associated non-cited violation of 10 CFR 50, Appendix B, Criterion II, "Quality Assurance Program," was identified by the inspectors for the reclassification of the Unit 2 and 3 containment cooling service water (CCSW) pump vault drain check valves from a quality status of safety-related to non-safety-related. The licensee had not yet determined corrective actions for this violation by the end of the inspection period.

  The finding was determined to be more than minor because the finding, if left uncorrected, would become a more significant safety concern. Specifically, by removing the quality assurance requirements for this part, the licensee reduced the assurance that replacement parts are of sufficient quality to assure reliable service during and following design basis events. The inspectors concluded this finding was associated with the Mitigating Systems 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 for the Mitigating Systems Cornerstone. The finding screened as of very low safety significance (Green) because the finding was a qualification deficiency confirmed not to result in loss of operability or functionality. The inspectors did not identify a cross-cutting aspect associated with this finding, primarily because the reclassification occurred in 2004. (Section 1R06)

- Green. A finding of very low safety significance was self-revealed for the failure to follow the preventive maintenance program which resulted in the failure of the Unit 3 303241-52A GE HFA relay. This relay gives a start permissive signal for all three reactor feed pumps (RFPs). The licensee's corrective actions included restoring the correct preventive maintenance item (replace the relay), including adding a preventive maintenance item for the associated Unit 2 relay. The licensee also included a review of relays in multiple systems to ensure that the proper preventive maintenance items were identified and scheduled.

  The finding was determined to be more than minor because it was associated with the Mitigating Systems Cornerstone attribute of equipment performance and affected the cornerstone objective of ensuring the availability, reliability, and capability of a system that responds to an initiating event to prevent undesirable consequences. 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 for the reasons stated in the previous paragraph. The inspectors answered question 4 "YES." The finding represented an actual loss of safety function of one or more trains of equipment designated as risk-significant per 10 CFR 50.65 for >24 hours. The inspectors verified that Feedwater Level Control was a high safety significant function per the licensee’s Maintenance Rule database and that the inability to restart any of the Unit 3 RFP’s lasted longer than 24 hours. The Senior Reactor Analysts (SRAs) performed SDP phases 2 and 3 analyses of this finding. The exposure period was determined to be approximately 5 months, the time between the last known successful operation of the relay and the failure. For the phase 2 evaluation, the SRAs solved the transient (TRANS), small loss of coolant accident (SLOCA), and loss of direct current bus (LODC) worksheets in the “Risk-Informed Inspection Notebook for Dresden Nuclear Power Station Units 2 and 3 (Revision 2.1a)” assuming that the power conversion system (PCS) was unavailable for greater than 30 days. Using the counting rule for adding sequences described in IMC 0609 Appendix A, “Determining the Significance of Reactor Inspection Findings for At-Power Situations," the SDP result was a “6” or a finding of low to moderate safety significance. The SRAs determined that a phase 3 SDP was necessary because the phase 2 result assumed that the main feedwater (MFW) pumps would always be unavailable and because the exposure period was 5 months rather than 1 year assumed by the phase 2 SDP process. For the phase 3 evaluation, the SRA modified the Standardized Plant Analysis Risk Model (SPAR) for Dresden to add basic events modeling the potential for MFW to trip. The SRAs assumed MFW would trip in response to a reactor trip approximately 6 percent of the time and that MFW would not be recoverable. The estimated delta CDF over the exposure period was 9.0E-8/yr, which is a finding of low to moderate safety significance (Green). The dominant sequence was a manual shutdown followed by the trip of MFW and the inability to restart the pumps. Random failures of the isolation condenser, high pressure coolant injection and low pressure coolant injection were also part of the dominant sequence. There were no cross-cutting aspects to this finding. (Section 1R18)

- <u>Green</u>. A finding of very low safety significance and associated non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” was self-revealed for the failure to have a procedure adequate to ensure quality during the preventive maintenance (PM) performed on the high pressure coolant injection (HPCI) 2-2301-29, “Return to Condenser Valve,” in March 2011. The violation was entered into the licensee’s corrective action program as IR 1250901, “HPCI Return To Condenser Leak From Valve Body.” The licensee’s corrective actions included determining the acceptable internal and external inspection scope and revising procedure DMP 0040-06, “Copes-Vulcan Valve and Reverse Acting (Air to Open) Operator Maintenance,” as appropriate.

  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 availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to identify long term degradation during a preventive maintenance activity in March 2011 resulted in the HPCI system becoming inoperable in August 2011. 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 answered Question 2, (Does the finding represent a loss of system safety function?) "Yes" and went to Inspection Manual Chapter 0609, Appendix A. A Region III Senior Reactor Analyst performed an SDP phase 3 evaluation using the Standardized Plant Analysis Risk (SPAR) model for Dresden. The high pressure coolant injection system was modeled as unavailable for an exposure period of 6 days. The delta CDF estimate was 7.9E-8/yr, which represents a finding of very low safety significance (Green). The dominant core damage sequence was a loss of main feedwater followed by the failure or unavailability of high and low pressure injection sources. The inspectors did not identify a cross-cutting aspect associated with this finding. (Section 1R19)

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

No violations were identified.

REPORT DETAILS

Summary of Plant Status

Unit 2

On July 20, 2011, load was reduced to approximately 78 percent electrical due to low main condenser vacuum caused by prolonged high intake temperature due to seasonal variations. The unit returned to full power operation on July 24, 2011.

On August 13, 2011, load was reduced to approximately 93 percent electrical because a circulation water pump had to be secured when the lift station lost power due to water shorting out the TR041 circuit switcher. The unit returned to full power operation on the same day.

On August 14, 2011, load was reduced to approximately 84 percent electrical for a control rod pattern adjustment. The unit returned to full power operation on the same day.

On September 1, 2011, load was reduced to approximately 81 percent electrical to maintain discharge canal effluent temperatures below 90 degrees Fahrenheit (˚F) to stay in compliance with Dresden's National Pollutant Discharge Elimination System (NPDES) permit. The unit returned to full power operation on September 4, 2011.

Unit 3

On July 12, 2011, load was reduced to approximately 91 percent electrical when the site went to 2 circulation water pump operation after the 2/3 A lift pump tripped due to neutral overcurrent. The unit returned to full power operation on July 13, 2011.

On July 18, 2011, load was reduced to approximately 78 percent electrical due to low main condenser vacuum caused by prolonged high intake temperature due to seasonal variations. The unit returned to full power operation on July 24, 2011.

On September 4, 2011, load was reduced to approximately 72 percent electrical for a control rod pattern adjustment. The unit returned to full power operation on the same day.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 Summer Seasonal Readiness Preparations

a. Inspection Scope

The inspectors performed a review of the licensee's preparations for summer weather for selected systems, including conditions that could lead to an extended drought.

During the inspection, the inspectors focused on plant specific design features and the licensee's procedures used to mitigate or respond to adverse weather conditions. Additionally, the inspectors reviewed the Updated Final Safety Analysis Report (UFSAR) and performance requirements for systems selected for inspection, and verified that operator actions were appropriate as specified by plant specific procedures. Specific documents reviewed during this inspection 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 the corrective action program in accordance with station corrective action procedures. The inspectors' reviews focused specifically on the following plant systems:

- Unit 2 drywell spray; and
- Units 2 and 3 station blackout diesel generators.

This inspection constituted one seasonal adverse weather sample as defined in Inspection Procedure (IP) 71111.01-05.

b. Findings

No findings were identified.

.2 (Discussed) Unresolved Item (URI) 05000237/2011003-01; 05000249/2011003-01,"Failure to Include Adequate Acceptance Criteria in a Surveillance Test"

Background: Dresden's UFSAR, Section 3.4.1.1, "External Flood Protection Measures," states, in part, that in the highly unlikely event that a probable maximum flood (PMF) is predicted (528 feet), the plant will shut down in advance of the time predicted for flood stage occurrence, i.e., grade level (517.5 feet). The PMF flood procedure will be implemented upon a forecast of river levels exceeding 506.5 feet.

When the water level reaches 509 feet both reactors will be shutdown, the drywells will be de-inerted, and both vessels will be flooded. The reactors will be cooled to the lowest legal temperature as quickly as possible.

If the water level reaches 513 feet at the plant site, cooling of the reactors will be transferred to the isolation condensers, which will thereafter maintain the primary system in a safe shutdown condition.

If forecasted flood levels exceed 517 feet, a diesel-driven emergency flood pump will be connected by hoses to a fire system header in each unit. Through these fire system headers, the emergency flood pump will be capable of providing at least 175 gallons per minute (gpm) of flow to each unit. This flow will be used for make up to the shell of the isolation condensers and the spent fuel pools.

Description: At the end of the previous inspection period the inspectors identified an unresolved item (URI 05000237/2011003-01; 05000249/2011003-01) regarding the failure to include adequate acceptance criteria in a surveillance test.

On April 8, 2011, the inspectors observed the performance of Work Order (WO) 872864, "D2/3 6Y PM Emergency Diesel Pump (Flood Pump) Operation." After the surveillance test was completed, the inspectors reviewed the completed work package and identified that the work instructions did not include acceptance criteria.

Task 1 of WO 872864, “MM D2/3 6Y PM Emergency Diesel Pump (Flood Pump) Operation,” stated that the surveillance was found and left within acceptance criteria. The comments section of Task 2 of WO 872864, “Ops Support Flood Emergency Makeup Pump Maintenance,” stated “there is no specific Acceptance Criteria in Task 01.” The licensee generated issue report (IR) 1209642, “NRC Identified URI with Flood Acceptance Criteria,” to address the inspectors concerns.

Calculation DRE99 0035, “Capacity and Discharge Head for Portable Isolation Condenser Make Up Pumps to be used during Flood Conditions,” Revision 4, determined that the most demanding hydraulic requirement for the flood pump is 350 gpm at 47 psig.

Dresden’s Updated Final Safety Analysis Report (UFSAR), Section 3.4.1.1, “External Flood Protection Measures,” requires, in part, that the emergency flood pump be capable of providing at least 175 gpm flow to each unit, should the flood levels exceed 517 feet.

10 CFR Part 50, Appendix B, Criterion XI, “Test Control,” requires that licensees establish a test program 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 acceptance limits contained in applicable design documents. Hence, the inspectors concluded that the test procedure for testing the emergency flood pump should have had acceptance criteria to demonstrate that the flood pump will perform satisfactorily in service.

Upon further discussions with the licensee, the inspectors noticed that in early 2007, the flood pump was reclassified as non-safety-related. Based on the definition of safety-related systems, structures and components, as described in Title 10 of the Code of Federal Regulations, Part 50.2, “Definitions,” and based on the fact that that the flood pump is utilized to mitigate the consequences of an event described in Section 3.4.1.1, “External Flood Protection Measures,” of the Dresden UFSAR, the inspectors were concerned that the flood pump had been misclassified as non-safety and it should have been classified as a safety-related piece of equipment.

The licensee was unable to produce documentation that explained the rationale behind the safety downgrade. However, licensee management personnel stated that the licensing basis definition for safety-related equipment for Dresden only included equipment used to mitigate design basis accidents described in Chapter 15 of the UFSAR. This definition was different than the definition of safety-related in 10 CFR 50.2, which states:

“Safety-related structures, systems, and components means those structures, systems, and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;
(2) The capability to shut down the reactor and maintain it in a safe shutdown condition; or
(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in § 50.34(a)(1) or § 100.11 of this chapter, as applicable.”

The licensee generated IR 1239579, “NRC Questions the Safety Classification of Diesel Flood Pump,” to address the inspector’s concerns. As part of this IR, the licensee generated an action to determine if the safety classification of the flood pump was appropriate based on Dresden’s design bases.

At the end of this inspection period the licensee was unable to produce any documentation that demonstrated that the NRC had accepted the definition that the term safety-related only referred to equipment used to mitigate UFSAR, Chapter 15 accidents. The licensee’s closure document for IR 1239579 stated that there was not a clear definition of a design basis event. The licensee concluded that the emergency diesel-driven flood pump should be classified as non-safety-related.

However, 10 CFR 50.49(b)(1)(ii), while intended for environmental qualification of electrical equipment, provides a clear definition of design basis events and states, “Design basis events are defined as conditions of normal operation, including anticipated operational occurrences, design basis accidents, external events, and natural phenomena for which the plant must be designed to ensure functions (b)(1)(i) (A) through (C) of this section.” Those functions in 10 CFR 50.49 are the same as those listed in 10 CFR 50.2.

“(i) This equipment is that relied upon to remain functional during and following design basis events to ensure--

(A) The integrity of the reactor coolant pressure boundary;

(B) The capability to shut down the reactor and maintain it in a safe shutdown condition; or

(C) The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guidelines in § 50.34(a)(1), § 50.67(b)(2), or § 100.11 of this chapter, as applicable.”

In addition, UFSAR Section 3.2.7, “Identification of Safety-Related Components of Systems or Structures,” stated that Generic Letter 83-28, "Required Actions Based on Generic Implications of Salem ATWS Events," defines safety-related systems and components. That definition is the same as 10 CFR 50.2.

UFSAR Section 3.2.7 also stated that subsequent to Generic Letter 83-28 a reclassification of mechanical and electrical systems and components was under taken utilizing the “Guideline for safety classification of systems, components, and parts used in Nuclear Power Plant Applications (NCIG-17) NP-6895 Research Project Q101-20 Final Report, February 1991.” Flooding was included in the definition of a Design Basis Event in NCIG-17 on page 4-3.

Finally, 10 CFR Part 50, Appendix A, Criterion 2—Design bases for protection against natural phenomena states, in part: Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions.

Structures, systems, and components important to safety at Dresden are not designed to withstand the effects of a maximum probable flood. Dresden was designed and constructed before the design criteria in 10 CFR Part 50, Appendix A, were required and therefore, both Dresden units were licensed before the requirements to protect against flooding were developed.

However, Dresden Unit 2 was reviewed by the NRC as part of the Systematic Evaluation Program (SEP). The purpose of SEP was to determine if an adequate level of safety existed at the plants that were licensed before the 10 CFR Part 50, Appendix A, Design Criteria were developed. Flooding was one of the areas reviewed under SEP. The NRC stated in NUREG-0823, "Integrated Plant Safety Assessment," dated February 1983, that the licensee's flood emergency plan in its existing form (at the time) was inadequate. One reason was that the plan did not adequately address post-flood conditions such as sources of emergency cooling water. The NRC recommended that the licensee have the capability to install and operate an emergency pump above the probable maximum flood level capable of providing water to the isolation condenser and other cooling needs for the duration of the flood.

Since the emergency diesel-driven flood pump described earlier is the only component capable of providing condensate water to the isolation condensers and make up water to the fuel pools on both units during and after a maximum probable flood, the inspectors concluded that the emergency diesel-driven flood pump was necessary to maintain both reactors in a safe shutdown condition and prevent a potential offsite exposure due to a loss of inventory in both unit fuel pools. Therefore, the inspectors concluded that the emergency diesel-driven flood pump was required to be classified as safety-related in order to ensure that the safety function was met.

Subsequent to the end of the inspection period, and after the initial exit meeting, on October 28, 2011, the licensee presented the inspectors with substantial additional documentation regarding the safety classification status of the emergency diesel-driven flood pump. The licensee contended that this documentation demonstrated NRC approval of the flood pump classification as other than safety-related. Since the inspectors have not completed their review of this additional documentation, the emergency diesel-driven flood pump safety classification is considered an Unresolved Item (**URI 05000237/2011004-01; 05000249/2011004-1, Classification of Emergency Diesel-Driven Flood Pump to Required Quality Standards)** pending further inspector review.

Reaching an enforcement conclusion on the failure to include adequate acceptance criteria in the emergency flood pump surveillance is dependent on the safety classification of the flood pump. Therefore, URI 05000237/2011003-01; 05000249/2011003-01 remains open pending the above review.

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 2 and Unit 3 emergency diesel generators (EDGs) when Unit 2/3 EDG tripped on high engine temperature;

- Unit 2 ‘A’ and ‘D’ containment cooling service water (CCSW) pumps during ‘B’ and ‘C’ pumps inoperable due to vault drain line check valve replacement; and
- U2 isolation condenser with high pressure coolant injection (HPCI) out-of-service for emergent maintenance.

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, 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.

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

.1 Routine Resident Inspector Tours (71111.05Q)

a. Inspection Scope

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

- Fire Zone 8.2.2B, Unit 3 Containment Cooling Service Water pumps, elevation 495’;
- Fire Zone 1.1.2.5.A, Unit 2 Isolation Condenser Area, elevation 589’;
- Fire Zone 11.1.1, Unit 3 Southwest Corner Room, elevation 476’; and
- Fire Zone 11.1.2 Unit 3 Southeast Corner Room Elevation 476’.

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 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(s) 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.

Specific documents reviewed during this inspection are listed in the Attachment to this report. This inspection constituted one internal flooding sample as defined in IP 71111.06-05.

b. Findings

Improperly Classifying the Unit 2 and 3 Containment Cooling Service Water (CCSW) Pump Vault Drain Check-Valves as Non-Safety-Related

Introduction: A finding of very low safety significance and associated non-cited violation of 10 CFR 50, Appendix B, Criterion II, “Quality Assurance Program,” was identified by the inspectors for the reclassification of the Unit 2 and 3 CCSW pump vault drain check valves from a quality status of safety-related to non-safety-related.

Description: The inspectors identified that the licensee reclassified the Unit 2 and 3 CCSW pump vault drain check valves from safety-related to non-safety-related on September 30, 2004. The inspectors determined that the CCSW vault drains are required to be safety-related. The Dresden UFSAR Section 3.4.1.2.1, “Protection of the Condensate Pump Room and Containment Cooling Service Water Pump Room,” stated, in part, that the CCSW pumps were enclosed in a water tight vault that will not result in a loss of all four pumps due to flooding. Licensee procedure CC-AA-304, “Component Classification,” Revision 2, provided the criteria and methodology used in determining the safety classification of components. Procedure CC-AA-304, Step 4.1.1.13, states, in part, that if the component performs any safety-related function with consideration given to whether the component is a panel, cabinet, enclosure, or structure required for the protection of safety-related equipment, then classify the component as safety-related. The CCSW vault drains are an extension of the water-tight enclosure that protects the safety-related CCSW system in the event of internal flooding.

Analysis: The inspectors determined that declassifying the Units 2 and 3 CCSW vault drain check valves from safety-related to non-safety-related was contrary to CC-AA-304, “Component Classification,” Revision 2 and was a performance deficiency.

The finding was determined to be more than minor because the finding, if left uncorrected, would become a more significant safety concern. Specifically, by removing the quality assurance requirements for this part, the licensee reduced the assurance that replacement parts are of sufficient quality to assure reliable service during and following design basis events. The inspectors concluded this finding was associated with the Mitigating Systems 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 for the Mitigating Systems Cornerstone. The finding screened as of very low safety significance (Green) because the finding was a qualification deficiency confirmed not to result in loss of operability or functionality.

The inspectors did not identify a cross-cutting aspect associated with this finding, primarily because the reclassification occurred in 2004.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion II, “Quality Assurance Program,” requires, in part, that the licensee identify the structures, systems, and components to be covered by the quality assurance program. Licensee procedure CC-AA-304, “Component Classification,” requires that structures required for the protection of safety-related equipment be classified as safety-related and hence covered by the Quality Assurance Program. The CCSW vault drain check valves are an extension of the CCSW vault and required for the protection of safety-related equipment.

Contrary to the above, between September 30, 2004, and September 30, 2011, the licensee failed to assure that Units 2 and 3 CCSW vault drain check valves were identified as components to be covered by the quality assurance program. Specifically, the licensee inappropriately classified the Units 2 and 3 CCSW vault drain check valves as non-safety-related. The licensee had not yet determined corrective actions for this violation by the end of the inspection period. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program as IR 1269945, "NRC Issue With Classification of the CCSW Check Valve," this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. (**NCV 05000237/2011004-02; 05000249/2011004-02**: Improperly Classifying the Unit 2 and 3 CCSW Pump Vault Drain Check-Valves as Non-Safety-Related).

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope

On August 29, 2011, 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 crews' performances in these areas were 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 were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

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

- Unit 2 emergency diesel generator; and
- Unit 3 emergency diesel generator.

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

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

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

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

b. Findings

No findings 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 station black out ventilation;
- Unit 2 and Unit 3 EDGs protected when Unit 2/3 EDG tripped on high engine temperature;
- Unit 2 HPCI inoperable; and
- 3B core spray inoperable.

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 were identified.

1R15 Operability Evaluations and Functional Assessments (71111.15)

a. Inspection Scope

The inspectors reviewed the following issues:

- Unit 2 EDG Cooling Water Operability Evaluation 11-002;
- IR 1245549, “U3 HPCI Room Temperature";
- IR 1229574, “New U3 DW Thermocouple (Recently Connected) is Reading High";
- OE-10-05, Revision 2, “Busses 23, 24, 33, 34 4kv Breakers have degraded shock absorbers”;
- OE OPEV 11-05, Revision 0,”Seismic effects on BWR Control Rod Scram at Low Reactor Pressures"; and
- IR 1263710,”NRC Concern – Impact of Unavailable Screen Refuse PPS.”

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

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

b. Findings

No findings were identified.

1R18 Plant Modifications (71111.18)

a. Inspection Scope

The inspectors reviewed the following modification(s):

- WO 1456191-01, "No Standby Light For 3A RFP When In Standby."

The inspectors reviewed the configuration changes and associated 10 CFR 50.59 safety evaluation screening against the design basis, the UFSAR, and the TS, as applicable, to verify that the modification did not affect the operability or availability of the affected system. The inspectors, as applicable, observed ongoing and completed work activities to ensure that the modifications were installed as directed and consistent with the design control documents; the modifications operated as expected; post-modification testing adequately demonstrated continued system operability, availability, and reliability; and that operation of the modifications did not impact the operability of any interfacing systems. As applicable, the inspectors verified that relevant procedure, design, and licensing documents were properly updated. Lastly, the inspectors discussed the plant modification with operations, engineering, and training personnel to ensure that the individuals were aware of how the operation with the plant modification in place could impact overall plant performance. Documents reviewed in the course of this inspection are listed in the Attachment to this report.

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

b. Findings

(1) Inadequate Relay Preventative Maintenance

Introduction: A finding of very low safety significance was self-revealed for the failure to follow the preventive maintenance program which resulted in the failure of the 303241-52A GE HFA relay. The relay gives a start permissive signal for all three reactor feed pumps (RFPs).

Description: On July 20, 2011, the 3A Reactor Feed Pump (RFP) was secured due to a plant down power. The indicating light for the 3A RFP being in standby did not illuminate as expected. The licensee performed troubleshooting and determined that the 303241-52A GE HFA relay had an acrid odor and heat marks were in the general vicinity of the relay's coil, indicating relay failure.

The reactor feed pumps are required to have forced ventilation in order to run.
The 303241-52A GE HFA relay provides a permissive signal that ventilation is running

so that any of the three feed pumps may be started. To clarify, with this relay failed, if any or all of the reactor feed pumps were stopped or tripped for any reason, the RFPs could not be restarted.

The licensee was committed to ANS 3.2-1988, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants." Paragraph 5.2.7.1 of ANS 3.2-1988, required a preventative maintenance (PM) program. The PM program was the same for safety-related and non-safety-related components. The PM program was implemented by procedure MA-AA-716-210, "Performance Centered Maintenance (PCM) Process," Revision 11. The licensee's preventive maintenance program was based on PM templates depending on the component (MA-AA-716-210, Step 4.7). The PM template can be deviated from, but a justification why the deviation was put in place is required (MA-AA-716-210, Step 4.7). Once a new PM is determined to be appropriate, due dates for completion are required to be established (MA-AA-716-210, Step 4.9.2).

The licensee determined that the most probable apparent cause for the relay failure was insufficient preventive maintenance (PM). The licensee determined that the reason for the insufficient PM was that the relay had been misclassified within the preventive maintenance program as a component that did not require preventative maintenance. The classification of the component determines what maintenance is required per the PM template. The licensee identified that the relay was subsequently reclassified as a component that required PM. However, the licensee was unable to determine when the relay was reclassified. When the component was reclassified no PM items required by the new preventive maintenance classification were generated. The reclassification of the relay required a replacement preventive maintenance activity every 10 years, which was not planned or scheduled.

<u>Analysis</u>: The inspectors determined that the failure to have a planned and scheduled PM activity to replace the 303241-52A GE HFA relay every 10 years was contrary to the preventive maintenance program described in MA-AA-716-210, "Performance Centered Maintenance (PCM) Process," Revision 11, and was a performance deficiency.

The finding was determined to be more than minor because it was associated with the Mitigating Systems Cornerstone attribute of equipment performance and affected the cornerstone objective of ensuring the availability, reliability, and capability of a system that responds to an initiating event to prevent undesirable consequences. Specifically, the feedwater system is one of the systems described in DEOP 100, "RPV [reactor pressure vessel] Control," Revision 10, as a system relied upon to maintain reactor water level between 8 and 48 inches in the reactor vessel after a transient. With the 303241-52A GE HFA relay failed, the RFPs would not restart once they tripped or were secured. Therefore, the lack of preventive maintenance affected the availability and reliability of all three Unit 3 RFPs.

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 for the reasons stated in the previous paragraph. The inspectors answered question 4 "YES." The finding represented an actual loss of safety function of one or more trains of equipment designated as risk-significant per 10 CFR 50.65 for >24 hours. The inspectors verified that Feedwater Level Control was a high safety-significant function per the licensee's Maintenance Rule database and that the inability to restart any of the Unit 3 RFP's lasted longer than 24 hours.

The Senior Reactor Analysts (SRAs) performed an SDP phase 2 and 3 analysis of this finding. The exposure period was determined to be approximately 5 months, the time between the last known successful operation of the relay and the failure. For the phase 2 evaluation, the SRAs solved the transient (TRANS), small loss of coolant accident (SLOCA), and loss of direct current bus (LODC) worksheets in the "Risk-Informed Inspection Notebook for Dresden Nuclear Power Station Units 2 and 3 (Revision 2.1a)" assuming that the power conversion system (PCS) was unavailable for greater than 30 days. Using the counting rule for adding sequences described in IMC 0609 Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," the SDP result was a "6" or a finding of low to moderate safety significance.

The SRAs determined that a phase 3 SDP was necessary because the phase 2 result assumed that the MFW pumps would always be unavailable and because the exposure period was 5 months rather than 1 year assumed by the phase 2 SDP process. For the phase 3 evaluation, the SRA modified the Standardized Plant Analysis Risk Model (SPAR) for Dresden to add basic events modeling the potential for main feedwater to trip. The SRAs assumed MFW would trip in response to a reactor trip approximately 6 percent of the time and that MFW would not be recoverable. The estimated delta CDF over the exposure period was 9.0E-8/yr which is a finding of low to moderate safety significance (Green). The dominant sequence was a manual shutdown followed by the trip of MFW and the inability to restart the pumps. Random failures of the isolation condenser, high pressure coolant injection and low pressure coolant injection were also part of the dominant sequence.

The inspectors did not identify a cross-cutting aspect associated with this finding.

Enforcement: No violation of regulatory requirements occurred. The licensee's corrective actions included restoring the correct preventive maintenance item (replace the relay) including adding a preventive maintenance item for the associated Unit 2 relay. The licensee also included a review of relays in multiple systems to ensure that the proper preventive maintenance items were identified and scheduled. (**FIN 05000237/2011004-03; 05000249/2011004-03**, Inadequate Relay Preventive Maintenance)

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

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

- Total Unit 2 CCSW Vault Leakage WO 1403807-02, "Op Perform PMT As Left DOS 4400-01";
- WO 1458933, "U2/3 Diesel Generator Trip";
- WO 1424257-02, "D2 12Y PM INSP/OH HPCI Steam Line Valve 2-2301-29";

- U2 EDG ASR and Three-Way Valves (DGCW) Replacement, WO 1272974-03, “OP PMT 2-3930-525 IAW” and WO 00485003-05, “OP U2 EDG PMT ECCS Start"; and
- 2/3 ‘B’ SBGT “B” Train PM, WO 01272688-01, “D2/3 2Y TS B SBGT Charcoal Freon R-11 Leak Test” and WO 01272866-01, “2/3 SBGT HEPA Filter Leak Test.”

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 TSs, 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

(1) Inadequate Preventive Maintenance Procedure For Valve 2-2301-29

Introduction: A finding of very low safety significance and associated non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” was self-revealed for the failure to have a procedure adequate to ensure quality during the preventive maintenance (PM) performed on the high pressure coolant injection (HPCI) 2-2301-29, “Return to Condenser Valve,” in March 2011.

Description: On August 12, 2011, the 2-2301-29 valve was determined to have a through-wall leak on the body. This valve directed flow from the upstream steam drain pot to the condenser. The valve was normally open and was required to close automatically on a HPCI system actuation. The valve failure resulted in declaring the HPCI system inoperable from August 12 through August 18, 2011. When performing the equipment apparent cause evaluation (EACE), the licensee determined that the 2-2301-29 failed due to steam impingement due to two-phase flow.

The licensee also determined that this valve was disassembled and inspected for PM purposes in March 2011 under WO 779273, “MM D2 12Y PM Insp/OH HPCI Stm Line Drn Isolation Valve.” The licensee’s EACE Attachment 6, Step 4 stated, “However, the component monitoring program attributes performed by the PM inspections are supposed to identify normal or abnormal equipment degradation. Since the impingement erosion has been occurring long term, the degradation should have been noticed during the last internal inspection."

The last internal inspection was the one performed in March 2011 under WO 779273. The licensee's EACE Attachment 6, Step 3 stated: "The valve was examined only from the bonnet area with the trim set removed per inspection procedure DMP 0040-06, 'Copes-Vulcan Valve and Reverse Acting (Air to Open) Operator Maintenance,' Revision 10. Therefore, the inlet and outlet areas of the valve could not be fully inspected. This was not described in depth in inspection procedure DMP 0040-06." The licensee's EACE also stated: "The valve inspection under WO 779273 did not involve verifying that the web of the valve would be degraded. However, the procedure used for inspection, DMP 0040-06, contains a simple checklist for parts inspection." There is no specific guidance or requirement to intrusively inspect the valve from the inlet or outlet in order to determine additional degradation that is not visible from the bonnet. The licensee determined that this inspection deficiency and the lack of subsequent corrective action to repair or replace the valve was the apparent cause of the failure.

Analysis: The inspectors determined that failure to have adequate procedural guidance to perform the preventive maintenance activity was contrary to 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," 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 availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to identify long term degradation during a preventive maintenance activity in March 2011 resulted in the HPCI system becoming inoperable in August 2011.

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 answered Question 2, (Does the finding represent a loss of system safety function?) "Yes" and went to Inspection Manual Chapter 0609, Appendix A. A Region III Senior Reactor Analyst performed an SDP phase 3 evaluation using the Standardized Plant Analysis Risk (SPAR) model for Dresden. The high pressure coolant injection system was modeled as unavailable for an exposure period of 6 days. The delta CDF estimate was 7.9E-8/yr, which represents a finding of very low safety significance (Green). The dominant core damage sequence was a loss of main feedwater followed by the failure or unavailability of high and low pressure injection sources.

The inspectors did not identify a cross-cutting aspect associated with this finding.

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

Contrary to the above, on March 15, 2011, the licensee failed to have a procedure that affected quality that was appropriate to the circumstances for which it was used. Specifically, the licensee performed a preventive maintenance internal inspection of the 2-2301-29, HPCI Return to Condenser Valve, using procedure DMP 0040-06, "Copes-Vulcan Valve and Reverse Acting (Air to Open) Operator Maintenance," Revision 10. Procedure DMP 0040-06 failed to have instructions appropriate to the circumstances in that the steps that governed the internal inspection of the valve were not in sufficient detail to identify degradation of valve internals. The licensee's corrective actions included determining the acceptable internal and external inspection scope and revising procedure DMP 0040-06, as appropriate. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program as IR 1250901, "HPCI Return To Condenser Leak From Valve Body," this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. (**NCV 05000237/2011004-04; 05000249/2011004-04**, Inadequate Preventive Maintenance Procedure For Valve 2-2301-29)

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:

- WO 1434074, "D3 Qtr TS CS Pmp Test with Torus Avail for IST Data Surv"; (IST)
- WO 1403807-03, "Op Perform As Found DOS 4400-01";
- WO 1450094-01, "D3 45D TS 3C CCSW Alert Range Testing";
- WO 1433192, "D3 Qtr TS Core Spray MO Valve Operability and Timing Surv"; and
- WO 1264817-01, "D2 24M TS Isolation Condenser Auto-Actuation."

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 was in accordance with TSs, the UFSAR, 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 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 four routine surveillance testing samples, and one inservice testing sample, as defined in IP 71111.22, Sections -02 and -05.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, and Emergency Preparedness

4OA1 Performance Indicator Verification (71151)

.1 Mitigating Systems Performance Index (MSPI) - Heat Removal System

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index - Heat Removal System performance indicator for Unit 2 and Unit 3 for the period from the second quarter 2010 through the first quarter 2011. To determine the accuracy of the Performance Indicator (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, dated October 2009, were used. The inspectors reviewed the licensee's operator narrative logs, MSPI derivation reports, issue reports, event reports and NRC Integrated Inspection Reports for the period of May 2010 through May 2011 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 heat removal system samples as defined in IP 71151-05.

b. Findings

No findings were identified.

.2 Mitigating Systems Performance Index - Residual Heat Removal System

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index - Residual Heat Removal System performance indicator for Unit 2 and Unit 3 for the period from the second quarter 2010 through the first quarter 2011. 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, MSPI derivation reports, issue reports, event reports and NRC Integrated Inspection Reports for the period of May 2010 through May 2011 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 residual heat removal system samples as defined in IP 71151-05.

b. Findings

No findings were identified.

.3 Mitigating Systems Performance Index - Cooling Water Systems

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index - Cooling Water Systems performance indicator for Unit 2 and Unit 3 for the period from the second quarter 2010 through the first quarter 2011. 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, MSPI derivation reports, issue reports, event reports and NRC Integrated Inspection Reports for the period of May 2010 through May 2011 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 cooling water system samples as defined in IP 71151-05.

b. Findings

No findings 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: identification of the problem was complete and accurate; timeliness was commensurate with the safety significance; 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 Attachment to this report.

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 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 were identified.

.3 Selected Issue Follow-Up Inspection Associated with IR 1240793, "U3 Battery RM Temp High"

a. Inspection Scope

The inspectors reviewed the corrective actions associated with an adverse trend in the temperature of the Unit 3 battery room that was identified by the licensee and documented in issue report (IR) 1240793. The inspectors chose this issue for an in-depth review due to the safety and risk significance of the batteries. The inspectors reviewed the surveillance test procedure and the troubleshooting activities to verify that the licensee was appropriately addressing the adverse trend in their corrective action program.

b. Findings and Observations

During daily operator rounds on July 18, 2011, a non-licensed operator (NLO) identified that the Unit 3 battery room temperature was above the maximum allowed. Room temperature at that time was 96 degrees Fahrenheit (˚F) when compared to a maximum allowed temperature of 85˚F. The licensee documented this adverse trend as IR 1240793, "U3 Battery RM Temp High." The licensee determined that the air conditioner unit tripped, causing the elevated room temperatures. The ventilation unit was still operating; therefore, hydrogen accumulation was not of concern. The licensee assigned the issue report a Significance Level 4 based on the issue not having any actual impact on nuclear safety and an Investigation Class D because no formal investigation into the issue was required in accordance with the licensee's corrective action program (CAP).

As described in engineering change (EC) 350673, "The effects of elevated temperatures on the Unit 3 Station Batteries," elevated battery room temperatures result in decreased battery life and individual cell voltages. Therefore, the licensee determined that the Unit 3 batteries were still able to perform their required function. Although the cause of the air conditioner unit trip was not known, the licensee continued collecting data and troubleshooting.

The inspectors determined that the licensee's documentation of the adverse battery room temperature trend in the CAP was complete and accurate. The inspectors also determined that the classification and prioritization of the resolution of the issue was appropriate commensurate with its safety significance.

No findings were identified.

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

.4 Annual Sample: Review of Operator Workarounds

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 operator workarounds (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 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 the 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 were identified.

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

a. Inspection Scope

The inspectors were made aware of plant events either through direct contact with the licensee or through the review of issue reports. In each case, the inspectors interviewed plant personnel, and reviewed procedures, work orders, and issue reports applicable to the appropriate event.

b. Findings and Observations

.1 Alert Declared Due to Chemical Spill

Introduction: The inspectors identified an unresolved item regarding 10 CFR 50.59 "Changes, Tests, and Experiments." The licensee installed tanks containing sodium hypochlorite and Hydroxyethylidenediphosphonic acid (HEDP, a strong acid) and may not have accounted for conditions for an accident of a different type than any previously evaluated in the Updated Final Safety Analysis Report (UFSAR).

Description: On July, 15, 2011, the licensee restricted access to the crib house due to a combined spill of sodium hypochlorite and HEPD. The mixture of the two chemicals produced chlorine gas. Based on initial assessment of the event and meeting the Emergency Action Level (EAL) threshold criteria an Alert, HA7, was declared at 10:16 a.m. The Alert was terminated at 3:20 p.m. on July 15, 2011, when actions to ventilate the crib house were completed and all areas were verified to be clear.

The inspectors were in the process of reviewing two modifications in regard to 10 CFR 50.59 which may not have accounted for conditions for an accident of a different type than any previously evaluated in the UFSAR. The first was the addition of the chemical storage tanks and the second was the removal of the control room ventilation Toxic Gas Analyzer from service. The inspectors considered this issue an unresolved item (URI) pending further evaluation efforts. (**URI 05000237/2011004-05; 05000249/2011004-05**)

.2 Closed Licensee Event Report (LER) 237/2009-002-01, "Unit 2 High Pressure Coolant Injection Suction Valve Fails to Close"

As a follow-up to LER 237/2009-002-00, the licensee completed a Root Cause Evaluation and documented the findings in LER 237/2009-002-01. The inspectors reviewed the results, findings and subsequent program adjustments to prevent similar failures. At this time, there were no performance deficiencies or violations of regulatory requirements. This LER is closed.

This event follow-up review constituted two samples as defined in IP 71153-05.

4OA6 Management Meetings

.1 Exit Meeting Summary

On October 12, 2011, the inspectors presented the inspection results to Mr. D. Czufin, 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. Proprietary material received during the inspection was returned to the licensee.

On November 4, 2011, the inspectors met with Messrs. D. Czufin and S. Marik regarding the Unresolved Items associated with the emergency diesel-driven flood pump.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

D. Czufin, Site Vice President
S. Marik, Station Plant Manager
H. Bush, Radiation Protection Manager
P. DiSalvo, GL 89-13 Program Owner
J. Fox, Design Engineer
G. Gates, Operations
G. Graff, Nuclear Oversight Manager
D. Gronek, Operations Director
M. Hosain, Site EQ Engineer
G. Ice, Regulatory Assurance – NRC Coordinator
L. Jordan, Training Director
B. Kapellas, Work Control Director
J. Knight, Chemistry Manager
M. Knott, Instrument Maintenance Manager
D. Leggett, Regulatory Assurance Manager
T. Loch, Design Engineering Manager
G. Morrow, Operations
M. McDonald, Maintenance Director
T. Mohr, Engineering Program Manager
P. O'Brien, Regulatory Assurance – NRC Coordinator
D. O'Flanagan, Security Manager
P. Quealy, Emergency Preparedness Manager
R. Ruffin, Licensing Engineer
J. Sipek, Engineering Director

Nuclear Regulatory Commission

S. West, Director, Division of Reactor Projects
M. Ring, Chief, Division of Reactor Projects, Branch 1

IEMA

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

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000237/2011004-01 05000249/2011004-01	URI	Classification of the Emergency Diesel-Driven Flood Pump to Required Quality Standards (1R01)
05000237/2011004-02 05000249/2011004-02	NCV	Improperly Classifying the Unit 2 and 3 Containment Cooling Service Water (CCSW) Pump Vault Drain Check-Valves as Non-Safety-Related (1R06)
05000237/2011004-03 05000249/2011004-03	FIN	Inadequate Relay Preventative Maintenance (1R18)
05000237/2011004-04 05000249/2011004-04	NCV	Inadequate Preventive Maintenance Procedure For Valve 2-2301-29 (1R19)
05000237/2011004-05 05000249/2011004-05	URI	Alert Declared Due to Chemical Spill (4OA3)

Closed

05000237/2011004-02 05000249/2011004-02	NCV	Improperly Classifying the Unit 2 and 3 Containment Cooling Service Water (CCSW) Pump Vault Drain Check-Valves as Non-Safety-Related
05000237/2011004-03 05000249/2011004-03	FIN	Inadequate Relay Preventative Maintenance (1R18)
05000237/2011004-04 05000249/2011004-04	NCV	Inadequate Preventive Maintenance Procedure For Valve 2-2301-29
237/2009-002-01	LER	Unit 2 High Pressure Coolant Injection Suction Valve Fails to Close

Discussed

05000237/2011003-01 05000249/2011003-01	URI	Failure to Include Adequate Acceptance Criteria in a Surveillance Test

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)

- WC-AA-107, “Seasonal Readiness,” Revision 9
- OP-AA-108-115, “Operability Determinations (CM-1),” Revision 10
- EC 375686, “Dresden Station Blackout (SBO) Building Ventilation Air Requirement Calculation Number DR-27D-M-002, Revision 3,” Revision 0
- IR 1240151, “SBO Availability Based on Ambient Temperature”
- IR 1241651, “Operation Outside of Engineering Analysis EC 375686”
- IR 1109567, “Dresden AR for Site Summer Readiness Actions 2011”
- IR 1091608, “Potential Freon Leak on U3 Battery AC System”
- IR 1144219, “TR-22 Local Temp Gauge Not Reading Correctly”
- IR 1242841, “NRC Senior Resident Concern Identified”
- IR 784713, “NRC Questions Concurrent Operability of 2/3 EDG and U2 CCSW”

1R04 Equipment Alignment (71111.04Q)

- DOP 1300-M1/E1; Unit 2 Isolation Condenser System; Revision 17
- DOP 0900-E1; Unit 2 (3) Control Room Panels; Revision 20
- Dres207LN001; Dresden Operations Training: Isolation Condenser; Revision May 06, 2011

1R05 Fire Protection (71111.05)

- Pre-Fire Plan for Fire Zone 11.1.1
- DSSP 0220-T6, “Temporary 4KV Feed Connections – SDC, LPCI, RBCCW, CCSW,” Revision 8
- IR 1257536, “2/3 DFP Capacity Test Data”
- IR 1258212, “2/3 DFP Engine Speed High”
- IR 1259060,”DFPS 4123-05 Diesel Fire Pump Engine Speed Out of Spec”
- ECR 401684 – Review of Diesel Fire Pump Engine Speed Requirements
- Pre-Fire Plan for Fire Zone 11.1.2
- CC-AA-211, “Fire Protection Program," Rev 4
- IR 1269198, NOS ID: Transient Combustibles Not Controlled Per Procedure, 09/28/2011

1R06 Flooding (71111.06)

- UFSAR Section 3.4.1.2.1.2, “Isolation of the Containment Cooling Service Water Pumps from Flood Water”
- WO 1278420, “D3 18M TSTR CCSW Pump Vault Penetration Seal Testing”
- IR 1202238, “Degraded Unit 3 CCSW Vault Penetration Identified”
- WO 1266345, “D3 18M TSTR CCSW Pump Vault Water Tight Door Leak Test”
- IR 1184652, “U3 CCSW Vault Door Failed LLRT, B & C CCSW Pumps Inop.”
- WO 1168289, “D3 18M TSTR CCSW PMP Vault Drain Line CHK VLV Leak Test”
- WO 99150018, “D3 8Y PM Repl Solenoid on CCSW Vault Drain AO – 3-4999-74”
- WO 692858, “D3 8Y PM CCSW PMP Vault Flood Prot Level Switch Funct Test”

- WO 505508, “D3 8Y Actuator Assembly Overhaul 3-4999-74”
- WO 1253126, “D3 24M TS CCSW Vault Drain Valve Test”
- EC 350183, “Component Classification for CCSW Vault Drain Line Valves and Instrumentation”
- EC 358324, “Seismic Qualification of Fisher-Bauman Control Valve with Air Operator P/N 32-24688”
- Dresden Special Report No. 33, “Final Flood Protection Measures, Dresden Units 2 and 3, (Permanent Flood Protection of the Containment Cooling Service Water Pumps and Diesel Generator Cooling Water Pumps)”
- NUREG-1796, “Safety Evaluation Report Related to the License Renewal of the Dresden Nuclear Power Station, Units 2 and 3 and Quad Cities Nuclear Power Station , Units 1 and 2”
- IR 1250805, “NRC: Question Submitted by the NRC”
- CC-AA-304, “Component Classification,” Revision 5
- IR 1256757, “Preemptive Replacement of U3 CCSW Vault Floor Drain Piping”

1R11 Licensed Operator Requalification Program (71111.11)

- OPEX-07-09-9B (Loss of RPS MG/Loss of vacuum; failure to scram; loss of feedwater), Revision 1, 7/11
- IR 1255935, “Licensed Operator Failed Re-Exam”

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

- -OP-AA-108-117; Protected Equipment Program; Revision 2
- -WC-AA-104; Integrated Risk Management; Revision 18
- -ER-AA-600; Risk Management; Revision 6
- -ER-AA-600-1042; On-Line Risk Management; Revision 7
- -Paragon Risk Software Remain in Service Results for High Pressure Core Injection Out-of-Service

1R15 Operability Evaluations (71111.15)

- IR 1229652, “2-3999-634 Appears To Be Passing Flow In Reverse Direction”
- M22, “Diagram of Service Water Piping,” Revision ED
- M29, “Diagram of L.P. Coolant Injection Piping,” Revision BD
- IR 1242868, “U3 HPCI Room Temperature Trend”
- IR 1233845, “Unit 3 HPCI Room Cooler Leak”
- IR 1255364, “NRC Concern With U3 HPCI Component Temperatures”
- EC 380152; Evaluation of Impact of Elevated Drywell Temperature on 3-1301-1
- WO 1356275; Install EC 380858
- AR 1070338; Failure to Monitor EQ Equipment in the DW for Adverse Temperatures
- AR 1229574; New U3 DW Thermocouple (Recently Connected) is Reading High
- AR 1064681; NRC Resident – Question on U3 DW Ambient Temperature
- AR 1069168; MOV 3-1301-01 EQ Life Affected by High DW Temperature
- DTS 1600-38; Drywell EQ Temperature Monitoring (W-1); Revision 1
- DRE 98-0077, “Dresden HPCI Room Thermal Response Reduced Room Cooler Capability”
- CHRON #200474, “19930425, EQ EVAL XMTL EQER 12-93-007/Selected Components EQ Zones 4, 5 +6/Affects of Increased Temp Due to Equipment Heat Load”
- EC 380272, Rev 2, “ Revision Needed to License Renewal Requirement for DW Equipment Drain Sump DISCH Line Inspection”

1R19 Post-Maintenance Testing (71111.19)

- DOS 4400-01, “Containment Cooling Service Water Vault Floor Drain,” Revision 12
- WO 1229061, “D2 24M TS D/G Test/Endur & Margin/Full Load Rej/ECCS”
- IR 1246218, “2/3 EDG High Engine Temp Alarms Early”
- IR 1246223, “2/3 EDG Tripped During Endurance Run”
- IR 1246225, “U2/3 Diesel Generator Trip”
- IR 1247466, “NRC Question”
- WO 1424257-02, “D2 12Y PM INSP/OH HPCI Steam Line Valve 2-2301-29”
- WO 1424257-03, “OP Verify No Leakage At System Pressure”
- EACE 1250901, “HPCI 2-2301-29 Return to Condenser Valve Body Leaks”
- DOS 2300-01; High Pressure Coolant Injection Valve Operability and Timing; Revision 48
- DOS 0040-07; Verification of Remote Position Indication for Valves Included in Inservice Testing (IST) Program; Revision 42
- Drawing M-51; Diagram of High Pressure Coolant Injection Piping; Revision CM
- WO 1424057; D2 12Y PM INSP/OH HPCI Steam Line 2-2301-29
- M-22, “Diagram of Service water Piping," Rev ED
- DOS 6600-02, “Reversal of Emergency Diesel Generator Cooling Water Flow," Rev 19
- WO 1272974-03, OP PMT 2-3930-525 IAW DOS 6600-02 (Flow Reversal), 08/03/11
- WO 00485003-05, OP U2 EDG PMT ECCS Start DOS 6600-12 / DOS 6600-01, 09/13/11
- IR 1263529, “U2 EDG PMT Delayed Return to Service," 09/14/2011
- CY-DR-120-413, “Cooling and Service Water Chemical Injection System," Rev 16
- 12E-2550A, “Schematic Diagram Engine Control & Gen. Excitation Standby Diesel Generator-2," Rev AR
- DOS 6600-12, “Diesel Generator Tests Endurance and Margin / Full Load Rejection / ECCS / Hot Start," Rev 54
- M-49, Diagram of Standby Gas Treatment, Revision QY
- WO 01272688-01, D2/3 2Y TS B SBGT Charcoal Freon R-11 Leak Test, 07/09/11
- WO 01272866-01, 2/3 SBGT HEPA Filter Leak Test, 07/09/11
- CO 00094872, First Hang – 2/3B Standby Gas Treatment Fan, Checklist 001
- CO 00094872, Final Hang – 2/3B Standby Gas Treatment Fan, Checklist 002
- IR 1265829, DTS 7500-11 Requires Revision, 09/21/2011
- IR 1266202, “B” SBGT Valve 2/3 7504B Did Not Close, 09/21/2011
- DTS 7500-13, SBGT System Visual Inspection, Revision 3

1R22 Surveillance Testing (71111.22)

- DOS 1400-05, “Core Spray System Pump Operability and Quarterly IST Test with Torus Available,” Revision 44
- DOS 4400-01, “Containment Cooling Service Water Vault Floor Drain,” Revision 12
- DOS 1500-02, “Containment Cooling Water Pump Test and Inservice Test (IST),” Revision 78
- DOS 1400-02, “Core Spray System Valve Operability and Timing,” Revision 29
- M-29 P&ID L.P. Coolant Injection Piping
- M-3121 P&ID Control Room HVAC

4OA1 Performance Indicator Verification (71151)

- DR-MSPI-01, “Reactor Oversight Program MSPI Bases Document Dresden Nuclear Station,” Revision 8
- NEI 99-02, Regulatory Assessment Performance Indicator Guideline, Revision 6
- IR 1162402, Total U2 CCSW Vault Leakage Above Admin Limit, 01/13/2011
- IR 1184652, U3 CCSW Vault Door Failed LLRT, B & C CCSW Pumps Inop, 03/08/2011

4OA2 Identification and Resolution of Problems (71152)

- IR 1240793, “U3 Battery Rm Temp High”
- IR 1241125, “U3 Battery Room Exhaust Fan and Lighting Not Working”
- IR 1241431, “Found A/C Unit Tripped”
- IR 1244491, “Issues with U3 Battery Ventilation”
- IR 1244713, “South 2/3 Chimney Lights Cause Breaker Trip”
- RP-AA-700-1221, “Calibration and Operation of TMX 412 Multi Gas Meter,” Revision 0
- RP-AA-901, “Explosive Gas Monitoring Program”
- IR 1257463, “Barrier on TB Roof will Impede Air to Ventilation Systems”
- IR 1257545, “U3 24/48 Battery Cell Temperatures Over Procedural Limit”
- DOS 8300-07, “Unit 2 (3) Weekly Station Battery Inspection,” Revision 09
- Engineering Change (EC) 350673, “The Effects of Elevated Temperatures on the Unit 3 Station Batteries”
- IR 1263037, “NRC Concern Regarding Restarting Fuel Pool Cooling”
- OP-AA-102-103, “Operator Work Around

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

- M-29, Diagram of L.P. Coolant Injection Piping Sh. 2, EDSF (no date)
- M-360, Diagram of L.P. Coolant Injection Piping Sh. 2, EDSF (no date)
- M-3121, Piping & Instrument Diagram Control Room HVAC, EDSF (no date)

LIST OF ACRONYMS USED

ADAMS	Agencywide Document Access Management System
AOP	Abnormal Operating Procedure
CAP	Corrective Action Program
CCSW	Component Cooling Service Water
CFR	Code of Federal Regulations
DFP	Diesel Fire Pump
DGCW	Diesel Generator Cooling Water
DRP	Division of Reactor Projects
EACE	Equipment Apparent Cause Evaluation
EAL	Emergency Action Level
EC	Engineering Change
EDG	Emergency Diesel Generator
˚F	Degrees Fahrenheit
gpm	Gallons Per Minute
HPCI	High Pressure Coolant Injection
HVAC	Heating, Ventilation, Air Conditioning
IMC	Inspection Manual Chapter
INPO	Institute of Nuclear Power Operations
IP	Inspection Procedure
IR	Inspection Report
IR	Issue Report
IST	In-Service Testing
LER	Licensee Event Report
LLRT	Local Leak Rate Testing
LODC	Loss of Direct Current
LPCI	Low Pressure Coolant Injection
MFW	Main Feedwater
MSPI	Mitigating Systems Performance Index
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NLO	Non-Licensed Operator
NOV	Notice of Violation
NPDES	National Pollutant Discharge Elimination System
NRC	U.S. Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
OWA	Operator Workaround
PARS	Publicly Available Records System
PCS	Power Conversion System
PI	Performance Indicator
PM	Planned, Post or Preventative Maintenance
PCM	Performance Centered Maintenance
PMF	Probable Maximum Flood
psig	Pounds Per Square Inch Gauge
RFP	Reactor Feed Pump
RPV	Reactor Pressure Vessel
SDP	Significance Determination Process
SEP	Systematic Evaluation Program
SLOCA	Small Loss of Coolant Accident
SPAR	Standardized Plant Analysis Risk Model

SRA	Senior Reactor Analyst (NRC)
SSC	Structure, System, and Component
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
WO	Work Order

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Sincerely,

/RA/

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

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

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Letter to M. Pacilio from M. Ring dated November 7, 2011

SUBJECT: DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3, INTEGRATED INSPECTION REPORT 05000237/2011004; 05000249/2011004

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