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

May 7, 2013

EA-13-079

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 05000237/2013002, 05000249/2013002; PRELIMINARY WHITE FINDING

Dear Mr. Pacilio:

On March 31, 2013, 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 April 8, 2013, with Mr. D. Czufin, and other members of your staff. Additionally on April 19, 2013, the NRC discussed with Mr. S. Marik, of your staff, the preliminary White determination for the finding discussed below.

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.

This report discusses one NRC-identified finding, concerning the site's external flooding strategy, which has preliminarily been determined to be a White finding with low to moderate safety significance that may require additional NRC inspections. Specifically, Dresden Abnormal Operating Procedure, DOA 0010-04, "Floods," did not contain steps directing operators to maintain reactor vessel inventory during a probable maximum flood event when a reasonable simulation of this procedure was executed in August 2012.

The finding is not a current safety concern. A licensee procedure, TSG -3, Attachment T, establishing a pathway for adding make-up water to the reactor coolant system during external flooding events up to and including the probable maximum flood, was implemented in November 2012.

This finding with the supporting circumstances and details is documented in the enclosed inspection report. This finding was assessed based on the best available information, using the applicable Significance Determination Process. The basis for the NRC's preliminary significance determination is also described in the enclosed report. This finding is also an apparent violation of NRC requirements and is being considered for escalated enforcement action in accordance with the Enforcement Policy, which can be found on the NRC's Web site at http://www.nrc.gov/about-nrc/regulatory/enforcement/enforce-pol.html.

In accordance with NRC Inspection Manual Chapter 0609, we intend to complete our evaluation using the best available information and issue our final determination of safety significance within 90 days of the date of this letter. The significance determination process encourages an open dialogue between the NRC staff and the licensee; however, the dialogue should not impact the timeliness of the staff's final determination. Before we make a final decision on this matter, we are providing you with an opportunity (1) to attend a Regulatory Conference where you can present to the NRC your perspective on the facts and assumptions the NRC used to arrive at the finding and assess its significance, or (2) submit your position on the finding to the NRC in writing. If you request a Regulatory Conference, it should be held within 30 days of the receipt of this letter and we encourage you to submit supporting documentation at least one week prior to the conference in an effort to make the conference more efficient and effective. If a Regulatory Conference is held, it will be open for public observation. If you decide to submit only a written response, such submittal should be sent to the NRC within 30 days of your receipt of this letter. If you decline to request a Regulatory Conference or submit a written response, you relinquish your right to appeal the final Significance Determination Process determination, in that by not doing either, you fail to meet the appeal requirements stated in the Prerequisite and Limitation sections of Attachment 2 of Inspection Manual Chapter 0609.

Please contact Mr. Jamnes Cameron at 630-829-9833 and in writing within 10 days from the issue date of this letter to notify the NRC of your intentions. If we have not heard from you within 10 days, we will continue with our significance determination and enforcement decision. The final resolution of this matter will be conveyed in separate correspondence.

Because the NRC has not made a final determination in this matter, no Notice of Violation is being issued for the inspection finding at this time. In addition, please be advised that the number and characterization of the apparent violation described in the enclosed inspection report may change as a result of further NRC review.

This report also documents one additional NRC-identified finding of very low safety significance (Green). This additional finding was determined not to involve a violation of NRC requirements.

If you contest the subject or severity of this Green finding, 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 Agencywide Document Access and Management 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 by Kenneth G. O'Brien for/***

Steven A. Reynolds, Director
Division of Reactor Projects

Docket Nos. 50-237, 50-249
License Nos. DPR-19 and DPR-25

Enclosure: Inspection Report 05000237/2013002, 05000249/2013002
w/Attachment: Supplemental Information

cc w/encl: Distribution via ListServ

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

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|---|---|
| Docket Nos: | 05000237; 05000249 |
| License Nos: | DPR-19 and DPR-25 |
| Report No: | 05000237/2013002; 05000249/2013002 |
| Licensee: | Exelon Generation Company, LLC |
| Facility: | Dresden Nuclear Power Station, Units 2 and 3 |
| Location: | Morris, IL |
| Dates: | January 1 through March 31, 2013 |
| Inspectors: | G. Roach, Senior Resident Inspector<br>D. Meléndez-Colón, Resident Inspector<br>D. Jones, Reactor Inspector<br>J. Corujo-Sandín, Reactor Engineer<br>T. Go, Health Physicist |
| Approved by: | J. Cameron, Chief<br>Branch 6<br>Division of Reactor Projects |

Enclosure

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## SUMMARY OF FINDINGS

Inspection Report (IR) 05000237/2013002, 05000249/2013002; 01/01/2013 – 03/31/2013; Dresden Nuclear Power Station, Units 2 & 3; Adverse Weather Protection and Identification and Resolution of Problems.

This report covers a 3-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. Two findings were identified by the inspectors. One of these findings was considered an apparent violation of NRC regulations. The significance of inspection findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, “Significance Determination Process” (SDP), dated June 2, 2011. Cross-cutting aspects are determined using IMC 0310, “Components Within the Cross Cutting Areas,” dated October 28, 2011. Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. All violations of NRC requirements are dispositioned in accordance with the NRC’s Enforcement Policy dated January 28, 2013. 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. The inspectors identified a Finding having very low safety significance for the failure to include acceptance criteria in a surveillance test for equipment that is the sole source of make-up water to the isolation condenser and spent fuel pool for both units during a probable maximum flood (PMF) scenario postulated in the Updated Final Safety Analysis Report (UFSAR). As described in the Exelon Quality Assurance Manual, the licensee is committed to the requirements of ANSI/ANS 3.2-1988, which states that surveillance tests contain or reference acceptance criteria in appropriate design or other source documents.

The inspectors determined that the failure to include adequate acceptance criteria in a surveillance test was a performance deficiency warranting a significance evaluation. The inspectors determined that the finding was more than minor because if left uncorrected, it could lead to a more significant safety concern. Specifically, without any acceptance criteria in the surveillance test, the licensee cannot determine whether the flood pump was able to perform its function as described in the UFSAR and calculation DRE99-0035. The inspectors completed a Phase 1 significance determination of this finding and determined that the finding impacted the Mitigating Systems Cornerstone. The inspectors concluded that the diesel-driven make-up pump would be a mitigating system in the case of the probable maximum flood. The inspectors answered “No” to the question on Exhibit 2 - Mitigating Systems Screening Questions of Appendix A, “The Significance Determination Process for Findings At-Power,” of IMC 0609. As a result, the issue screened as of very low safety significance. Similar issues were identified previously by the inspectors involving inadequate surveillance test and operating procedures for the flood pump. Therefore, the inspectors determined that this finding has a cross-cutting aspect in the area of Problem Identification and Resolution, Corrective Action Program. (P.1(d)) (Section 1R01.3)

Preliminary White: The inspectors identified a finding and an associated Apparent Violation (AV) of Technical Specification (TS) Section 5.4.1. Technical Specification 5.4.1 requires, in part, that written procedures be established, implemented, and maintained covering the following activities: "the applicable procedures recommended in Regulatory Guide (RG) 1.33. Revision 2, Appendix A, February 1978." RG 1.33. Revision 2, Appendix A, Paragraph 6 addresses "Procedures for Combating Emergencies and Other Significant Events" and Item w addresses "Acts of Nature (e.g ., tornado, flood, dam failure, earthquakes)." From February 20, 1991, to November 21, 2012, the licensee failed to establish a procedure addressing all of the effects of an external flooding scenario on the plant. Specifically, DOA 0010-04, "Floods," did not account for reactor vessel inventory make up during an external flooding scenario up to and including the probable maximum flood event which could result in reactor vessel water level lowering below the top of active fuel. This finding does not represent an immediate safety concern in that the licensee now has procedures for providing reactor vessel make up water during an external flood scenario up to and including a PMF event.

The inspectors determined that the licensee's failure to consider reactor vessel inventory make up during an external flooding scenario up to and including the PMF was a performance deficiency warranting a significance evaluation. The finding was determined to be more than minor in accordance with Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," dated September 7, 2012, because it was associated with the Mitigating Systems Cornerstone attribute of procedure quality and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. A Significance and Enforcement Review Panel (SERP), using IMC 0609, Appendix M, "Significance Determination Process Using Qualitative Criteria," dated April 12, 2012, preliminarily determined the finding to be of low to moderate safety significance (White). The inspectors determined that this finding has a cross-cutting aspect in the area of Problem Identification and Resolution, Corrective Action Program, Self and Independent Assessments, since it involves the failure to identify the lack of procedural steps to address a critical function during a comprehensive self assessment of the flooding strategy. (P.3(a)) (Section 4OA2)

## B. Licensee-Identified Violations

None.

# REPORT DETAILS

## Summary of Plant Status

### Unit 2

With the exception of planned short duration reduction in power to support control rod pattern adjustments, Unit 2 remained at or near full power for the entirety of the inspection period.

### Unit 3

On January 28, operators reduced power to approximately 77 percent in an effort to control rising exciter bearing no.11 trends, which was due to improperly tensioned turbine shaft grounding brushes. Operators restored power to 100 percent on January 29, 2013.

On February 17, operators reduced power to approximately 96 percent for a planned insertion of control rod drive G-8 for scram solenoid pilot valve repairs. Operators restored power to 100 percent on February 17, 2013.

With the exception of planned short duration reduction in power to support control rod pattern adjustments, Unit 3 was maintained at or near full power for the remainder of the inspection period.

## 1. REACTOR SAFETY

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

1R01 Adverse Weather Protection (71111.01)

.1 Readiness for Impending Adverse Weather Condition – Extreme Cold Conditions

a. Inspection Scope

Since extreme cold conditions were forecast in the vicinity of the facility for January 22, 2013, the inspectors reviewed the licensee's overall preparations/protection for the expected weather conditions. The inspectors walked down the cribhouse and the 125 volts-direct current (Vdc) and 250 Vdc battery systems because their safety related functions could be affected or required as a result of the extreme cold conditions forecast for the facility. The inspectors observed insulation, heat trace circuits, space heater operation, and weatherized enclosures to ensure operability of affected systems. The inspectors reviewed licensee procedures and discussed potential compensatory measures with control room personnel. The inspectors focused on plant management's actions for implementing the station's procedures for ensuring adequate personnel for safe plant operation and emergency response would be available. Specific documents reviewed during this inspection are listed in the Attachment to this report.

This inspection constituted one readiness for impending adverse weather condition sample as defined in Inspection Procedure (IP) 71111.01-05.

b. Findings

No findings were identified.

.2 Readiness for Impending Adverse Weather Condition – High Wind Conditions

a. Inspection Scope

Since a strong winter storm with the potential for high winds was forecast in the vicinity of the facility for January 29, 2013, the inspectors reviewed the licensee's overall preparations/protection for the expected weather conditions. The inspectors walked down the high pressure coolant injection and isolation condenser systems, in addition to the licensee's emergency alternating current (AC) power systems, because their safety related functions could be required as a result of high-winds-generated missiles or the loss of offsite power. The inspectors evaluated the licensee staff's preparations against the site's procedures and determined that the staff's actions were adequate. During the inspection, the inspectors focused on plant-specific design features and the licensee's procedures used to respond to specified adverse weather conditions. The inspectors also toured the plant grounds to look for any loose debris that could become missiles from strong wind gusts. The inspectors evaluated operator staffing and accessibility of controls and indications for those systems required to control the plant. 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. The inspectors also reviewed a sample of corrective action program (CAP) items to verify that the licensee identified adverse weather issues at an appropriate threshold and dispositioned them through the CAP in accordance with station corrective action procedures. Specific documents reviewed during this inspection are listed in the Attachment to this report.

This inspection constituted one readiness for impending adverse weather condition sample as defined in IP 71111.01-05.

b. Findings

No findings were identified.

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

a. Inspection Scope

The inspectors reviewed the unresolved item and reviewed the resolution of Unresolved Item (URI) 05000237/2011004-01; 05000249/2011004-01, "Classification of Emergency Diesel-Driven Flood Pump to Required Quality Standards," in Section 4OA5.1 of this report to determine whether a violation of any regulatory requirements existed.

b. Findings

Introduction: The inspectors identified a Finding having very low safety significance (Green) for the failure to include adequate acceptance criteria in a surveillance test for equipment that is the sole source of make-up water to the isolation condenser and spent

fuel pool for both units during a probable maximum flood (PMF) scenario as postulated in the UFSAR.

Description: 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 was completed, the inspectors reviewed the completed work package and identified that the work instructions did not include acceptance criteria for the surveillance.

Work Order 872864 instructed the licensee, in part, to:

- Throttle 2-inch brass valve until a discharge pressure of 50 psig (-0%, +2%) was reached;
- Record pump discharge pressure;
- Record engine speed;
- Record the number of gallons in the tank;
- Record the time required to fill the tank.

Revision 2 of the WO instructions stated: "Clarified work step no.19 to perform test or tests at the discretion of the test engineer. Test discharge pressure to be determined by test engineer." The test engineer determined that the 2-inch brass valve was to be throttled until discharge pressures of 50, 75 and 100 pounds-force per square inch gauge (psig) were reached.

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 gallons per minute (gpm) at 47 psig.

Dresden USFAR, Section 3.4.1.1, "External Flood Protection Measures," states, in part, that in the highly unlikely event that a PMF is predicted (528 feet (ft)) above mean sea level (MSL)), the plant will shutdown in advance of the time predicted for flood stage occurrence, i.e., grade level (517.5 ft). When the water level reaches 509 ft all reactors will be shut down, the drywells will be deinerted, and the vessels will be flooded.

If the water level reaches 513 ft MSL 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 forecast flood levels exceed 517 ft MSL, 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 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.

None of these requirements were referenced in the work order. 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."

As described in the Exelon Quality Assurance Manual, the licensee is committed to follow the requirements of American National Standard Administrative Control and Quality Assurance for the Operational Phase of Nuclear Power Plants (ANSI/AN 3.2-1988). This standard states, in part, under Section 5.3.14, "Test and Inspection Procedures," that tests, including surveillance tests, and inspection procedures contain or reference, as appropriate, acceptance criteria or limits contained in applicable design or other source documents, such as vendor's literature, engineering drawings or plant specification that will be used to evaluate the results.

Similar issues were identified previously by the inspectors, involving surveillance tests and operating procedures for the flood pump. Refer to non-cited violation (NCV) 05000237/2004010-02; 05000249/2004010-02, "Source of Make-up Water," URI 05000237/2006010-04; 05000249/2006010-04, "Full Flow Testing of the Diesel Driven Flood Pump at Design Conditions," and NCV 05000237/2007003-04;05000249/2007003-04, "Failure to Identify and Correct Issues with the Operation and Testing of the Diesel Driven Pump Used to Respond to External Flooding."

Analysis: The inspectors determined that the failure to include acceptance criteria in a surveillance test for equipment that is the sole source of make-up water to the isolation condenser for both units during a PMF scenario did not meet ANSI/ANS 3.2-1988, a performance deficiency warranting a significance evaluation in accordance with Inspection Manual Chapter (IMC) 0612, "Power Reactors Inspection Reports," Appendix B, "Issue Screening," issued on September 7, 2012. The inspectors determined that the finding was more than minor because if left uncorrected, it could lead to a more significant safety concern. Specifically, without any acceptance criteria in the surveillance test, the licensee cannot determine whether the flood pump was able to perform its function as described in the UFSAR and calculation DRE 99-0035.

The inspectors completed a Phase 1 significance determination of this finding using IMC 0609, "Significance Determination Process," Attachment 609.04, issued on June 19, 2012. The inspectors determined that the finding impacted the Mitigating Systems Cornerstone. The inspectors concluded that the diesel-driven make-up pump would be a mitigating system in the case of the probable maximum flood. The inspectors answered "No" to the questions on Exhibit 2 - Mitigating Systems Screening Questions of Appendix A, "The Significance Determination Process (SDP) For Findings At-Power," of IMC 0609. As a result, the issue screened as of very low safety significance (Green).

The inspectors determined that this finding has a cross-cutting aspect in the area of Problem Identification and Resolution, Corrective Action Program, since it involves the failure to ensure that issues potentially impacting nuclear safety are promptly identified, fully evaluated, and that actions are taken to address safety issues in a timely manner, commensurate with their significance. Specifically, the licensee failed to take appropriate corrective actions to include acceptance criteria in a surveillance test for equipment that is the sole source of make-up water to the isolation condenser and spent fuel pool for both units during a PMF scenario as postulated in the UFSAR. As discussed in the description section, similar issues were identified previously by the inspectors, involving surveillance test and operating procedures for the flood pump. [P.1(d)]

Enforcement: This finding does not involve enforcement action because, while a performance deficiency existed, no violation of a regulatory requirement was identified.

The licensee generated issue reports (IR) 1209642, "NRC Identified URI with Flood Pump Acceptance Criteria," and 1487554, "Follow-Up to IR to NRC Question on Diesel Driven Flood Pump," to document the inspector's concerns. Corrective action includes the development of acceptance criteria which ensure the pump meets licensing basis requirements for a PMF event.

Because this finding does not involve a violation and is of very low safety or security significance, it is identified as a FIN. (**05000237/2013002-01; 05000249/2013002-01**, **Failure to Include Acceptance Criteria in a Surveillance Test)**

This URI is closed. This activity does not represent a completed inspection sample.

1R04 Equipment Alignment (71111.04Q and S)

.1 Quarterly Partial System Walkdowns

a. Inspection Scope

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

- Unit 2/3 emergency diesel generator (EDG) during Unit 3 EDG maintenance;
- Unit 2 'A' standby liquid control (SBLC) train during 'B' SBLC train inoperable for relay repair;
- Unit 2 isolation condenser (IC) during Unit 2 high pressure coolant injection (HPCI) maintenance outage; and
- Unit 3 IC during Unit 3 HPCI maintenance outage.

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, 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 four partial system walkdown samples as defined in IP 71111.04-05.

b. Findings

No findings were identified.

.2 Semi Annual Complete System Walkdown

a. Inspection Scope

On March 12, 2013, the inspectors performed a complete system alignment inspection of the Unit 2 HPCI system to verify the functional capability of the system. This system was selected because it was considered both safety significant and risk significant in the licensee's probabilistic risk assessment. The inspectors walked down the system to review mechanical and electrical equipment lineups; electrical power availability; system pressure and temperature indications, as appropriate; component labeling; component lubrication; component and equipment cooling; hangers and supports; operability of support systems; and to ensure that ancillary equipment or debris did not interfere with equipment operation. A review of a sample of past and outstanding WOs was performed to determine whether any deficiencies significantly affected the system function. In addition, the inspectors reviewed the CAP database to ensure that system equipment alignment problems were being identified and appropriately resolved. Documents reviewed are listed in the Attachment to this report.

These activities constituted one complete system walkdown sample 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 8.2.5C, Unit 2 Lube Oil Room and Unit 2/3 Electro-hydraulic Control Reservoir Area, Elevation 517';
- Fire Zone 11.2.1, Unit 2 Southwest Corner Room, Elevation 476';
- Fire Zone 11.2.3, Unit 2 HPCI Pump Room, Elevation 476'; and
- Fire Zone 9.0C, Unit 2/3 Swing Diesel Generator Room, Elevation 517'.

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 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 (CCSW) pump vault with a focus on the floor drain system check valve.

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

No findings were identified.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Resident Inspector Quarterly Review of Licensed Operator Requalification (71111.11Q)

a. Inspection Scope

On February 8, 2013, the inspectors observed a crew of licensed operators in the plant's simulator during licensed operator requalification training 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 simulator sample as defined in IP 71111.11.

b. Findings

No findings were identified.

.2 Resident Inspector Quarterly Observation of Heightened Activity or Risk (71111.11Q)

a. Inspection Scope

On January 29, 2013, the inspectors observed power ascension to 100 percent rated thermal power (RTP) on Unit 3 and adverse condition monitoring of the main turbine bearing no. 11 which had previously exhibited high temperatures and erratic vibrations. The licensee previously lowered generator power output until temperature and vibrations stabilized. The licensee determined the bearing was being adversely affected by static voltages developing between the main turbine and the bearing metal which were the resultant of the main turbine shaft ground brushes not being properly fastened to the shaft at the conclusion of the previous refueling outage, D3R22, in December 2012. Once the licensee restored adequate brush tension on the shaft the stray voltages were alleviated and bearing conditions normalized. Prior to restoring plant conditions to full power, the licensee consulted with the vendor to ensure bearing no. 11 was capable of operating at rated conditions. This was an activity that required heightened awareness or was related to increased risk. 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 (if applicable);
- correct use and implementation of procedures;
- control board (or equipment) manipulations;
- oversight and direction from supervisors; and
- ability to identify and implement appropriate TS actions and Emergency Plan actions and notifications (if applicable).

The performance in these areas was compared to pre established operator action expectations, procedural compliance and task completion requirements. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one quarterly licensed operator heightened activity/risk sample as defined in IP 71111.11.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Routine Quarterly Evaluations (71111.12Q)

a. Inspection Scope

The inspectors evaluated degraded performance issues involving the following risk-significant system and evaluated the periodic assessment of the maintenance rule:

- Maintenance Rule Periodic Assessment no. 9 (10CFR50.65(a)(3) Assessment) Assessment Period 10/1/2010 – 9/30/2012; and
- 4 kv switchgear and circuit breakers.

The inspectors reviewed events such as where ineffective equipment maintenance had or could have 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)

.1 Maintenance Risk Assessments and Emergent Work Control

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 2 Yellow Risk during the performance of DIS 1500-05 (24 month low pressure coolant injection (LPCI) initiation circuitry testing);
- Unit 2 Yellow Risk during 2B SBLC train inoperable for relay repair;
- Unit 2 Yellow Risk during Division I CCSW work window;
- Unit 2 Yellow Risk for U2 HPCI maintenance outage; and
- Unit 3 Yellow Risk for U3 HPCI maintenance outage.

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. Specific documents reviewed during this inspection are listed in the Attachment to this report.

These maintenance risk assessments and emergent work control activities constituted five samples as defined in IP 71111.13-05.

b. Findings

No findings were identified.

1R15 Operability Determinations and Functional Assessments (71111.15)

.1 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following issues:

- IR 1453700, "Key Calculation Review Identifies Issues in DRE 98-0030;"
- IR 1455933, "U3 HPCI Valve Found in Alert Range During Valve Timing;"
- Engineering Change Evaluation 39168, "Unit 3 Drywell Equipment Drains Sump Cover Plate Bent," Revision 0;
- IR 1487125, "U2 Isolation Condenser Support Nut Engagement Deficiency;" and
- IR 1453610, "Installed Contactors Do Not Meet All Acceptance Criteria."

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 five samples as defined in IP 71111.15-05.

b. Findings

No findings were identified.

1R18 Plant Modifications (71111.18)

.1 Plant Modifications

a. Inspection Scope

The inspectors reviewed the following modification:

- IR 1481716, "3-0590-102D Relay Did Not Pick Up During Main Steam Line Isolation Valve Closure SCRAM Circuit Functional Test, DOS 0500-08"

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 temporary plant modification samples as defined in IP 71111.18-05.

b. Findings

No findings 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 1379083-03, "OP PMT EDG3 Fuel Oil Transfer Pump Motor 3-5203;"
- WO 1418376, "Dresden Unit 2 Two Year PM Standby Diesel Generator Inspection;"
- WO 1605861, "D2 Quarterly TS HPCI [high pressure coolant injection] Pump Operability Test and IST [in-service testing] Surveillance;"
- WO 1423633, "D3 2 Year TS HPCI Pump Comprehensive Operability Test and In-Service Test (IST) Surveillance;" and
- WO 1546493-02, "PM Air Start Regulator Valve on 2/3 EDG."

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

No findings were identified.

## 1R22 Surveillance Testing (71111.22)

### .1 Surveillance Testing

a. Inspection Scope

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

- WO 01579759, “D2 Quarterly TS LPCI Pump Run and IST Surveillance,”
- WO 01407992, “Recirculation Pump Running Differential Pressure Switch Calibration” (routine);
- Unit 3 drywell floor drain sump and drywell equipment drain sump (RCS);
- WO 01234966, “Electrical Maintenance D3 4Y EQ Butyl Rubber Cable Surveillance MO3-1301-4” (routine);
- WO 01616773, “D2/3 1M TSTR/COM Diesel Fire Pump Operability Surveillance” (routine); and
- WO 01396238, “D2 Recirculation Flow Dual Limiter 262-26B” (routine).

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

- did preconditioning occur;
- the effects of the testing were adequately addressed by control room personnel or engineers prior to the commencement of the testing;
- acceptance criteria were 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, ASMEs 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, 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, one in-service testing sample, and one reactor coolant system leak detection sample as defined in IP 71111.22, Sections -02 and -05.

b. Findings

No findings were identified.

1EP6 Drill Evaluation (71114.06)

.1 Emergency Preparedness Drill Observation

a. Inspection Scope

The inspectors evaluated the conduct of a routine licensee emergency drill on February 7, 2013, to identify any weaknesses and deficiencies in classification, notification, and protective action recommendation development activities. The inspectors observed emergency response operations in the 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 corrective action program. 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 were identified.

## 2. RADIATION SAFETY

### CORNERSTONE: OCCUPATIONAL RADIATION SAFETY

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

This inspection constituted a partial sample as defined in IP 71124.01-05.

.1 Radiological Hazard Assessment (02.02)

a. Inspection Scope

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

b. Findings

No findings were identified.

.2 Instructions to Workers (02.03)

a. Inspection Scope

The inspectors selected various containers holding non-exempt licensed radioactive materials that may cause unplanned or inadvertent exposure of workers, and assessed whether the containers were labeled and controlled in accordance with 10 CFR 20.1904, “Labeling Containers,” or met the requirements of 10 CFR 20.1905(g), “Exemptions To Labeling Requirements”.

For work activities that could suddenly and severely increase radiological conditions, the inspectors assessed the licensee’s means to inform workers of changes that could significantly impact their occupational dose.

b. Findings

No findings were identified.

.3 Contamination and Radioactive Material Control (02.04)

a. Inspection Scope

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 has 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.

b. Findings

No findings were identified.

.4 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, radiation work permits, 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 electronic personal dosimeters in high noise areas as high radiation area monitoring devices.

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 examined the posting and physical controls for selected high radiation areas and very high radiation areas to verify conformance with the occupational performance indicator.

b. Findings

No findings were identified.

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

a. Inspection Scope

The inspectors discussed with the radiation protection manager the controls and procedures for high-risk high radiation areas and very high radiation areas. The inspectors discussed methods employed by the licensee to provide stricter control of very high radiation area access as specified in 10 CFR 20.1602, “Control of Access to Very High Radiation Areas,” and Regulatory Guide 8.38, “Control of Access to High and Very High Radiation Areas of Nuclear Plants.” The inspectors assessed whether any changes to licensee procedures substantially reduce the effectiveness and level of worker protection.

The inspectors discussed the controls in place for special areas that have the potential to become very high radiation areas during certain plant operations with first-line health physics supervisors (or equivalent positions having backshift health physics oversight authority). The inspectors assessed whether these plant operations require communication beforehand with the health physics group, so as to allow corresponding

timely actions to properly post, control, and monitor the radiation hazards including re-access authorization.

b. Findings

No findings were identified.

.6 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 radiological conditions in their workplace and the radiation work permit controls/limits in place, and whether their performance reflected the level of radiological hazards present.

b. Findings

No findings were identified.

2RS6 Radioactive Gaseous and Liquid Effluent Treatment (71124.06)

This inspection constituted one complete sample as defined in IP 71124.06-05.

.1 Inspection Planning and Program Reviews (02.01)

Event Report and Effluent Report Reviews

a. Inspection Scope

The inspectors reviewed the radiological effluent release reports issued since the last inspection to determine if the reports were submitted as required by the Offsite Dose Calculation Manual/TSs. The inspectors reviewed anomalous results, unexpected trends, or abnormal releases identified by the licensee for further inspection to determine if they were evaluated, were entered in the corrective action program, and were adequately resolved.

The inspectors identified radioactive effluent monitor operability issues reported by the licensee as provided in effluent release reports, to review these issues during the onsite inspection, as warranted, given their relative significance and determine if the issues were entered into the CAP and adequately resolved.

b. Findings

No findings were identified.

Offsite Dose Calculation Manual and Final Safety Analysis Report Review

a. Inspection Scope

The inspectors reviewed UFSAR descriptions of the radioactive effluent monitoring systems, treatment systems, and effluent flow paths so they could be evaluated during inspection walkdowns.

The inspectors reviewed changes to the Offsite Dose Calculation Manual made by the licensee since the last inspection against the guidance in NUREG-1301, 1302 and 0133, and Regulatory Guides 1.109, 1.21 and 4.1. When differences were identified, the inspectors reviewed the technical basis or evaluations of the change during the onsite inspection to determine whether they were technically justified and maintain effluent releases as-low-as-is-reasonably-achievable (ALARA).

The inspectors reviewed licensee documentation to determine if the licensee has identified any non-radioactive systems that have become contaminated as disclosed either through an event report or the Offsite Dose Calculation Manual since the last inspection. This review provided an intelligent sample list for the onsite inspection of any 10 CFR 50.59 evaluations and allowed a determination if any newly contaminated systems have an unmonitored effluent discharge path to the environment, whether any required Offsite Dose Calculation Manual revisions were made to incorporate these new pathways and whether the associated effluents were reported in accordance with Regulatory Guide 1.21.

b. Findings

No findings were identified.

Groundwater Protection Initiative Program

a. Inspection Scope

The inspectors reviewed reported groundwater monitoring results and changes to the licensee's written program for identifying and controlling contaminated spills/leaks to groundwater.

b. Findings

No findings were identified.

Procedures, Special Reports, and Other Documents

a. Inspection Scope

The inspectors reviewed Licensee Event Reports, event reports and/or special reports related to the effluent program issued since the previous inspection to identify any additional focus areas for the inspection based on the scope/breadth of problems described in these reports.

The inspectors reviewed Effluent Program implementing procedures, particularly those associated with effluent sampling, effluent monitor set-point determinations, and dose calculations.

The inspectors reviewed copies of licensee and third party (independent) evaluation reports of the Effluent Monitoring Program since the last inspection to gather insights into the licensee's program and aid in selecting areas for inspection review (smart sampling).

b. Findings

No findings were identified.

.2 Walkdowns and Observations (02.02)

a. Inspection Scope

The inspectors walked down selected components of the gaseous and liquid discharge systems to evaluate whether equipment configuration and flow paths aligned with the documents reviewed in Section 2RS6.1 02.01 above and to assess equipment material condition. Special attention was made to identify potential unmonitored release points (such as open roof vents in boiling water reactor turbine decks, temporary structures butted against turbine, auxiliary or containment buildings), building alterations which could impact airborne, or liquid effluent controls, and ventilation system leakage that communicates directly with the environment.

For equipment or areas associated with the systems selected for review that were not readily accessible due to radiological conditions, the inspectors reviewed the licensee's material condition surveillance records, as applicable.

The inspectors walked down filtered ventilation systems to assess for conditions such as degraded high-efficiency particulate air/charcoal banks, improper alignment, or system installation issues that would impact the performance or the effluent monitoring capability of the effluent system.

As available, the inspectors observed selected portions of the routine processing and discharge of radioactive gaseous effluent (including sample collection and analysis) to evaluate whether appropriate treatment equipment was used and the processing activities align with discharge permits.

The inspectors determined if the licensee has made significant changes to their effluent release points, (e.g., changes subject to a 10 CFR 50.59 review) or require NRC approval of alternate discharge points.

As available, the inspectors observed selected portions of the routine processing and discharge liquid waste (including sample collection and analysis) to determine if appropriate effluent treatment equipment is being used and that radioactive liquid waste is being processed and discharged in accordance with procedure requirements and aligns with discharge permits.

b. Findings

No findings were identified.

.3 Sampling and Analyses (02.03)

a. Inspection Scope

The inspectors selected effluent sampling activities, consistent with smart sampling, and assessed whether adequate controls have been implemented to ensure representative samples were obtained (e.g., provisions for sample line flushing, vessel recirculation, composite samplers, etc.)

The inspectors selected effluent discharges made with inoperable (declared out-of-service) effluent radiation monitors to assess whether controls were in place to ensure compensatory sampling was performed consistent with the radiological effluent TSs/Offsite Dose Calculation Manual and that those controls were adequate to prevent the release of unmonitored liquid and gaseous effluents.

The inspectors determined whether the facility was routinely relying on the use of compensatory sampling in lieu of adequate system maintenance, based on the frequency of compensatory sampling since the last inspection.

The inspectors reviewed the results of the Inter-Laboratory Comparison Program to evaluate the quality of the radioactive effluent sample analyses and assessed whether the Inter-Laboratory Comparison Program includes had-to-detect isotopes as appropriate.

b. Findings

No findings were identified.

.4 Instrumentation and Equipment (02.04)

Effluent Flow Measuring Instruments

a. Inspection Scope

The inspectors reviewed the methodology the licensee uses to determine the effluent stack and vent flow rates to determine if the flow rates were consistent with Radiological Effluent Technical Specifications/Offsite Dose Calculation Manual or UFSAR values, and that differences between assumed and actual stack and vent flow rates did not affect the results of the projected public doses.

b. Findings

No findings were identified.

Air Cleaning Systems

a. Inspection Scope

The inspectors assessed whether surveillance test results since the previous inspection for TSs required ventilation effluent discharge systems (high-efficiency particulate air and charcoal filtration), such as the Standby Gas Treatment System and the Containment/Auxiliary Building Ventilation System, met Technical Specifications acceptance criteria.

b. Findings

No findings were identified.

.5 Dose Calculations (02.05)

a. Inspection Scope

The inspectors reviewed all significant changes in reported dose values compared to the previous radiological effluent release report (e.g., a factor of 5, or increases that approach Appendix I criteria) to evaluate the factors which may have resulted in the change.

The inspectors reviewed radioactive liquid and gaseous waste discharge permits to assess whether the projected doses to members of the public were accurate and based on representative samples of the discharge path.

Inspectors evaluated the methods used to determine the isotopes that are included in the source term to ensure all applicable radionuclides are included within detectability standards. The review included the current Part 61 analyses to ensure hard-to-detect radionuclides are included in the source term.

The inspectors reviewed changes in the licensee's offsite dose calculations since the last inspection to evaluate whether changes were consistent with the Offsite Dose Calculation Manual and Regulatory Guide 1.109. Inspectors reviewed meteorological dispersion and deposition factors used in the Offsite Dose Calculation Manual and effluent dose calculations to evaluate whether appropriate factors were being used for public dose calculations.

The inspectors reviewed the latest Land Use Census to assess whether changes (e.g., significant increases or decreases to population in the plant environs, changes in critical exposure pathways, the location of nearest member of the public, or critical receptor, etc.) have been factored into the dose calculations.

For the releases reviewed above, the inspectors evaluated whether the calculated doses (monthly, quarterly, and annual dose) are within the 10 CFR Part 50, Appendix I and TSs dose criteria.

The inspectors reviewed, as available, records of any abnormal gaseous or liquid tank discharges (e.g., discharges resulting from misaligned valves, valve leak-by, etc) to ensure the abnormal discharge was monitored by the discharge point effluent monitor. Discharges made with inoperable effluent radiation monitors, or unmonitored leakages were reviewed to ensure that an evaluation was made of the discharge to satisfy 10 CFR 20.1501 so as to account for the source term and projected doses to the public.

b. Findings

No findings were identified.

.6 Groundwater Protection Initiative Implementation (02.06)

a. Inspection Scope

The inspectors reviewed monitoring results of the Groundwater Protection Initiative to determine if the licensee had implemented its program as intended and to identify any anomalous results. For anomalous results or missed samples, the inspectors assessed whether the licensee had identified and addressed deficiencies through its CAP.

The inspectors reviewed identified leakage or spill events and entries made into 10 CFR 50.75 (g) records. The inspectors reviewed evaluations of leaks or spills and reviewed any remediation actions taken for effectiveness. The inspectors reviewed onsite contamination events involving contamination of ground water and assessed whether the source of the leak or spill was identified and mitigated.

For unmonitored spills, leaks, or unexpected liquid or gaseous discharges, the inspectors assessed whether an evaluation was performed to determine the type and amount of radioactive material that was discharged by:

Assessing whether sufficient radiological surveys were performed to evaluate the extent of the contamination and the radiological source term and assessing whether a survey/evaluation had been performed to include consideration of hard-to-detect radionuclides.

Determining whether the licensee completed offsite notifications, as provided in its Groundwater Protection Initiative implementing procedures.

The inspectors reviewed the evaluation of discharges from onsite surface water bodies that contain or potentially contain radioactivity, and the potential for ground water leakage from these onsite surface water bodies. The inspectors assessed whether the licensee was properly accounting for discharges from these surface water bodies as part of their effluent release reports.

The inspectors assessed whether on-site ground water sample results and a description of any significant on-site leaks/spills into ground water for each calendar year were documented in the Annual Radiological Environmental Operating Report for the Radiological Environmental Monitoring Program or the Annual Radiological Effluent Release Report for the Radiological Effluent TSs.

For significant, new effluent discharge points (such as significant or continuing leakage to ground water that continues to impact the environment if not remediated), the inspectors evaluated whether the offsite dose calculation manual was updated to include the new release point.

b. Findings

No findings were identified.

.7 Problem Identification and Resolution (02.07)

a. Inspection Scope

Inspectors assessed whether problems associated with the Effluent Monitoring and Control Program were being identified by the licensee at an appropriate threshold and were properly addressed for resolution in the licensee CAP. In addition, they evaluated the appropriateness of the Corrective Actions for a selected sample of problems documented by the licensee involving radiation monitoring and exposure controls.

b. Findings

No findings were identified.

## 4. OTHER ACTIVITIES

**Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness, Occupational Radiation Safety, Public Radiation Safety, and Security**

4OA1 Performance Indicator Verification (71151)

.1 Unplanned Scrams per 7000 Critical Hours

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Scrams per 7000 Critical Hours (IE01) performance indicator (PI) for Dresden Nuclear Power Station Units 2 and 3 covering the period from the first through fourth quarter 2012. 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, dated October 2009, were used. The inspectors reviewed the licensee’s operator narrative logs, issue reports, event reports and NRC Integrated Inspection Reports for the period of first through the fourth quarter 2012 to validate the accuracy of the submittals. The inspectors also reviewed the licensee’s issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

This inspection constituted two unplanned scrams per 7000 critical hours samples as defined in IP 71151-05.

b. Findings

No findings were identified.

.2 Unplanned Scrams with Complications

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Scrams with Complications (IE02) performance indicator for Dresden Nuclear Power Station Units 2 and 3 covering the period from the first through the fourth quarter 2012. 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, event reports and NRC Integrated Inspection Reports for the period of first through the fourth quarter 2012 to validate the accuracy of the submittals. The inspectors also reviewed the licensee’s issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

This inspection constituted two unplanned scrams with complications samples as defined in IP 71151-05.

b. Findings

No findings were identified.

.3 Unplanned Transients per 7000 Critical Hours

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Transients per 7000 Critical Hours (IE03) performance indicator Dresden Nuclear Power Station Units 2 and 3 covering the period from the first through the fourth quarter 2012. 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, maintenance rule records, event reports, and NRC Integrated Inspection Reports for the period of first through the fourth quarter 2012 to validate the accuracy of the submittals. The inspectors also reviewed the licensee’s issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

This inspection constituted two unplanned transients per 7000 critical hours 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 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: Review of the Site's Procedural and Physical Modifications for the Response to a Probable Maximum Flood Event

a. Inspection Scope

In August 2012, as required by a letter from the NRC to licensees entitled, "Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident," dated March 12, 2012 (ADAMS Accession No. ML12053A340), the licensee performed external flooding vulnerability walk downs of the site and a reasonable simulation of the flood response Abnormal Operating Procedure DOA 0010-04, "Floods," under the observation of the site's NRC resident inspectors and NRC staff from the Japan Lessons Learned Directorate (JLD). The results of the simulation and walk downs indicated that the licensee could perform the procedure as written, except potentially for several non-critical steps, within the appropriate timeline and that the procedure could achieve its goal of placing both Units 2 and 3 in a Hot Shutdown condition (Mode 3) and maintaining each unit's respective

spent fuel pool filled with make-up water. Many areas for improvement with the procedure were identified, but no specific issue was identified that by itself could be shown as preventing the licensee from achieving the desired end state within the time line identified by the 1982, Hydrological Considerations Technical Evaluation Report which defines the PMF scenario at the Dresden Nuclear Power Station. The PMF for Dresden assumes a severe precipitation event covering northern Illinois and Indiana which results in flood still water levels reaching 525 ft above mean sea level (MSL) with wave run up reaching 528 ft MSL. Ground elevation at Dresden is 517.5 ft MSL with Illinois River level maintained at approximately 505 ft MSL by the downstream U. S. Army Corps of Engineers controlled Dresden Island Lock and Dam.

The site's flooding response requires opening safety related structures including the reactor building to the environment once flood levels reach site grade (517.5 ft MSL) allowing flood waters to enter, as the structures are not designed to resist the static force of the flood water on their outer walls. This results in a station blackout as offsite and onsite AC electrical power would be unavailable and emergency core cooling systems (ECCS) and other sources of cooling and injection would not be available as they are submerged by the flood waters or are without electrical power to operate. To cope with this condition, the licensee would operate a diesel-powered pump which will be connected to the fire protection system and provide water from the flooded reactor building to the shell side of each unit's isolation condenser to remove decay heat from the reactors and provide make-up water to both spent fuel pools. The flood pump was originally to be supported above the flood waters by a chain fall mounted to a jib crane in the reactor building track way, but can now be mounted on a floating dock which would be staged in the reactor building trackway in the lead up to the flood waters reaching site grade.

The inspectors performed a historical review of DOA 0010-04, "Floods," from its origin until the current revision (Revision 38); observed the licensee's performance during numerous simulations and actual demonstrations of portions of the flood strategy; ensured the availability of various instruments, gauges and indications relied upon by the licensee to implement the flood strategy; reviewed the licensee's implementation plans for the Aqua-Dam; assisted Headquarters and Regional management in developing and reviewing follow-up questions for the licensee; and reviewed the licensee's future plans for structurally modifying the reactor building in order to ensure adequate strength to resist the static forces of the flood waters and to ensure its water-tight integrity. This structural modification is intended to maintain the reactor building free from flooding and as such ensure the availability of onsite emergency AC power and numerous safety related systems installed in the plant to keep the reactors safely shut down.

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

b. <u>Findings and Observations</u>

Following the walk downs and simulation, the licensee along with a contracted engineering firm who observed these activities, developed a number of mitigating strategies for external flood events impacting the site up to and including the PMF. As previously noted, the site becomes vulnerable to flooding as soon as flood waters reach site grade. As a result, the licensee has purchased equipment and revised procedures to mitigate the impact on the site from an external flood. In addition, the NRC has raised

a number of questions and concerns regarding the outcome of the walk downs and reasonable simulation. These questions were submitted in a 30 day response letter to the licensee on November 1, 2012, (ML12306A393) with the licensee response to these questions received on December 1, 2012 (ML12348A012). The initial round of NRC questions along with licensee identified weaknesses in the site's response strategy resulted in the following enhancements and mitigating strategies being developed:

1) Enhancements

- Attachments were added to DOA 0010-04 which specifically list electrical buses to be de-energized and underground fuel tanks to be filled prior to flood waters reaching site grade.
- The purchase of four motor boats maintained in the owner controlled area (OCA) for use of moving personnel around the site instead of having to rely on offsite assets.
- The addition of the floating dock to mount the diesel-driven flood pump and six diesel fuel barrels provides a more stable platform that will naturally adjust to changing flood height levels as compared to suspending the pump from a chain fall strung from the reactor building jib crane.
- Purchasing back up floating gas powered drafting pumps which can be used in the event the diesel-driven flood pump needs to be taken out of service.
- Flood height markers added in the reactor building trackway and the cribhouse will assist the site in trending flood waters.
- Licensee follow up to the reasonable simulation identified that during the early stages of the flooding event, when level in the reactor building was still relatively shallow, it is possible for air in addition to water to be drawn into the suction of the flood pump. The licensee enhanced the flood procedure to direct operators to put the flood pump suction line into the CRD pull-put channel to increase its submergence and minimize the possibility of air binding the pump.

2) Mitigating Strategies

- The acquisition of 4000 feet of Aqua-Dam to provide protection for the power block for floods up to 5.5 feet above ground elevation. The berm is stored on two 53 foot long flatbeds parked in the OCA for deployment. The licensee also had a custom bridge/dock made to be able to transport personnel over the Aqua-Dam to boats/land on either side.
- Installation of flood barriers at the isolation condenser make-up pump house to reduce the amount of time and personnel required to provide flood protection to this structure early in the event. At the time of the reasonable simulation, the licensee provided two feet of flood protection by building a sand bag berm which required approximately 8.5 hours to construct. The addition of the barriers provides four feet of flood mitigation and only requires maintenance technicians two hours to deploy. Providing this protection allows the site to rely on plant installed equipment to provide make up to the isolation condensers removing reactor decay heat for lesser flooding events and delaying the transition to the dock mounted diesel flood pump which would still be providing make-up to the reactors and spent fuel pools.
- The licensee developed Technical Support Guideline (TSG) – 3, Attachment T, which created a proceduralized pathway to add make-up water to the reactors utilizing the pressurized fire header, through test connections in the SBLC,

directly to the reactor for both units if make-up via a traditional means is not able to be restored. The licensee also modified the flooding procedure to instruct operators to close the reactor recirculation loop isolation valves isolating recirculation pump seals from the reactor and removing a potential source of reactor coolant inventory loss.

The NRC submitted a second round of questions to the licensee on January 3, 2013, with a focus on several of the enhancements and mitigating strategies recently implemented and follow-up on the licensee's original responses (ML13003A226). The licensee responded on January 31, 2013, to the NRC's inquiries (ML13037A045).

Through review of the licensee's responses to the staff's concerns in addition to observing the reasonable simulation, the inspectors noted that the licensee's procedure for monitoring canal level as the flood waters rose above 509 ft. MSL required the use of canal level instrumentation. In particular, a level transmitter that could be read remotely from the control room as Plant Process Computer point E354 was called out as the primary means of determining flood level until flood waters reached site grade at 517.5 ft MSL. The flood procedure also allowed operators to manually identify flood level in the cribhouse with a tape measure or other level indicator if the canal level indicator was not available. During the reasonable simulation and prior, the E354 computer point was not functioning and Dresden operators would have always had to rely on local, manual actions to identify flood levels between 509 ft MSL and 517.5 ft MSL. It should be noted that following the NRC's Systematic Evaluation Program review of Unit 2 in 1982, the licensee committed in a letter from Thomas J. Rausch to Paul O'Connor titled, "Dresden 2 SEP Topic: II-3.B, Flooding Potential and Protective Requirements; II-3.B.1, Capability of Operating Plants to Cope with Design Basis Flood Conditions; and II-3.C, Safety Related Water Supply (Ultimate Heat Sink)," dated November 17, 1982, to install a level gauge in the intake canal. The inspectors noted that this level gauge was not operational for several years or may have never been operational requiring the operators to perform manual actions to identify flood height. In addition to performing several actions directed by DOA 0010-04 based on the flood water height prior to reaching site grade as measured by operators, the Dresden Emergency Plan requires the site to declare an ALERT under Emergency Action Level (EAL) HA4 when intake canal level reaches 513 ft MSL. The inspectors determined that not meeting a NRC commitment in that an operational canal level indicator was not installed following the Systematic Evaluation Program review of Unit 2 in 1982 was a performance deficiency. This performance deficiency was not more than minor in that the licensee had procedurally driven compensatory measures in place and possessed the equipment necessary to accurately measure flood water height between 509 ft MSL and 517.5 ft MSL and as a result would have been capable of performing the flooding response strategy and carry out the site's Emergency Plan.

The inspectors identified a second performance deficiency associated with no licensee procedure for providing make-up water to the reactor coolant system while flood waters are above site grade level.

<u>Introduction</u>: A finding preliminarily determined to be of low to moderate safety significance (White) and an associated Apparent Violation (AV) of TS Section 5.4.1 was identified by the inspectors, in that, prior to November 2012, the licensee's procedures and specifically Abnormal Operating Procedure (AOP) DOA 0010-04, "Floods," did not account for reactor inventory make up during an external flooding scenario up to and

including the PMF event which could result in reactor vessel water level lowering below top of active fuel (TAF) leading to core damage.

Description: In August 2012, the licensee performed a reasonable simulation of its external flooding strategy for coping with a PMF event. The inspectors subsequently noted that the flood strategy did not contain steps accounting for losses in reactor vessel inventory. Under normal plant conditions there are both unidentified and identified leakage paths from the reactor coolant system. During the PMF event, systems which would provide normal and emergency make up capacity to the reactors would be inundated by the flood waters and would not be available.

The NRC questioned the licensee regarding this concern in a 30 day response letter to the licensee on November 1, 2012 (ML12306A393). In response to this concern, the licensee developed TSG-3, Attachment T, effective November 21, 2012, proceduralizing the use of the fire protection header which would be pressurized and supplied by the diesel flood pump to supply through mechanical adapters, which are labeled and stored above projected PMF flood levels, the reactor by connecting to test connections in the SBLC system. The licensee also revised DOA 0010-04, “Floods,” directing operators to shut reactor recirculation loop isolation valves to reduce potential reactor leakage sources to those governed by the unidentified leakage Technical Specification.

The licensee determined that with reactor unidentified leakage at 5 gpm, the maximum permitted by TS, it would take approximately 130 hours to reach a reactor water level at the TAF. The Dresden PMF hydrograph indicates that flood waters would exist at site grade for approximately 57 hours. After the flood waters recede, the licensee identified that TSG-3 Attachment H, “Reactor Pressure Vessel Injection Using Portable Diesel Driven Pump,” could be used to provide injection by tying a dedicated diesel driven “FLEX” pump (three are maintained onsite) into the fire protection ring header and injecting to each unit’s reactor vessel through the low pressure coolant injection system. In addition, after the flood waters recede, mechanical level instruments for the entire fuel zone would once again be available to the operators.

The inspectors challenged the licensee’s leakage assertion in that TS permit a total leakage rate of up to 25 gpm which would significantly reduce the amount of time until TAF is reached under the worst case permitted leakage conditions in the reactor coolant system. The inspectors based this concern on the fact that the licensee did not originally employ the strategy for isolating reactor recirculation loops which would make them susceptible to losses due to both unidentified and identified pathways. During the site flood event operators would be limited in their ability to monitor reactor water level as the mechanical level indicator called out by licensee procedures has an indication band between +60 inches and – 60 inches of water. Top of active fuel for Dresden is considered at –143 inches of water. As a result, operators would have to provide make up to the reactor vessel much sooner than 130 hours in order for level in the reactor to not become indeterminate or possibly reach TAF during the 57 hours the flood waters inundate the site.

The licensee reviewed its inventory of procedures prior to November 2012 up to and including the Severe Accident Mitigation Guidelines and was not able to identify written instructions for operators attempting to control reactor vessel level under station blackout conditions with high pressure coolant injection and all installed diesel driven systems

(fire protection) unavailable to provide injection capacity, which would be the situation on site while flood waters were present.

Analysis: The inspectors determined that the licensee's failure to consider reactor vessel inventory make up during an external flooding scenario up to and including the PMF was a performance deficiency warranting a significance evaluation. The finding was determined to be more than minor in accordance with Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," dated September 7, 2012, because it was associated with the Mitigating Systems Cornerstone attribute of procedure quality and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events 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, "Initial Characterization of Findings," dated June 19, 2012, and Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," Exhibit 2, "Mitigating Systems Screening Questions," dated June 19, 2012. The inspectors answered YES to the External Event Mitigation question which directed them to Exhibit 4, "External Events Screening Questions." The inspectors determined the statement that the finding "would degrade one or more trains of a system that supports a risk significant system or function" was TRUE and as a result a detailed risk evaluation was required.

A Significance and Enforcement Review Panel (SERP) determined that IMC 0609, Appendix M, "Significance Determination Process Using Qualitative Criteria," dated April 12, 2012, was appropriate to use due to the lack of existing quantitative SDP tools for evaluating external flooding risk. As part of that process, the Region III Senior Reactor Analyst (SRA) developed a simple event tree model to perform a bounding quantitative evaluation. The model represents an external flood event that exceeds grade level elevation (517.5') and requires implementation of the flood procedure, DOA 0010-04, "Floods," Revision 32.

The input assumptions were highly uncertain and were varied to calculate a range of risk estimates. The values for flood frequency, the probability of reactor pressure vessel leakage requiring makeup, and the likelihood of successful makeup to the vessel during and after the flood recedes were key inputs to the evaluation. The change in core damage frequency ($\Delta$CDF) estimates ranged from Green, a finding of very low safety significance, to Yellow, a finding of substantial safety significance. For the quantitative evaluation, the flood frequency was varied from 1E-4/yr to 1E-6/yr. The probability of reactor pressure vessel leakage requiring inventory makeup was varied from 1.0, makeup would always be required, to .02, makeup would be required 2 percent of the time. To represent the change in risk due to the performance deficiency, the SRA assumed that reactor pressure vessel (RPV) makeup during the flood would not be successful and that makeup after the flood would have a failure probability ranging from 0.1 to 0.5. For the base case, absent the performance deficiency, the SRA assumed that a makeup strategy would be available both during and after the flood and that the nominal failure probability would be much lower than in the performance deficiency case.

For the evaluation of the qualitative decision-making attributes, the NRC determined that defense-in-depth, safety margin, and the period of time the performance deficiency existed were the most important factors.

The flood procedure did not provide for defense-in-depth for reactor inventory control during a flood event. A flood greater than elevation 517.5 ft will fail sources of reactor inventory makeup including feedwater, condensate, control rod drive injection, fire pumps, service water and all ECCS. Recovery of plant systems after the flood is not likely to be successful and the use of alternate temporary systems after the flood recedes was not specified in the flood procedure. Plant shutdown under flooding conditions would require the implementation of many diverse plant procedures which are not well integrated into the flood procedure. The lack of integrated procedures combined with the lack of flood level predictive capabilities could result in variable plant conditions at the onset of significant flood impacts. As a result, the need for inventory makeup during the flood is possible. Also, the defense-in-depth barriers of secondary containment and primary containment are degraded during the implementation of the flood procedure.

The licensee did not have an administrative limit for reactor operation with reactor vessel leakage, only TS limits. The lack of a conservative limit allowed no safety margin between operational practices and TS limits.

The performance deficiency represents a longstanding issue. In 1982 the PMF scenario was initially identified. The licensee had the potential to identify that, during an extended site inundation flooding event, there would be a need to have planned actions to maintain RPV inventory but did not identify that need until the lack of a proceduralized method was questioned by the NRC.

A SERP held on April 18, 2013, made a preliminary determination that the finding was of low to moderate safety significance (White) based on the quantitative and qualitative evaluations. Considerations involved in that determination included the minimal defense in depth for addressing reactor vessel makeup, lack of administrative limits for reactor vessel leakage, that during a flooding event the operators would be required to implement numerous procedures which did not appear to be well integrated into the flooding response procedure, that the licensee had numerous opportunities recently and in the past to identify and then address the deficiency of not addressing a reactor makeup method, and the length of time the performance deficiency existed.

The inspectors determined that this finding has a cross-cutting aspect in the area of Problem Identification and Resolution, CAP, Self and Independent Assessments, since it involves the failure to identify the lack of procedural steps to address a critical function during a comprehensive self assessment of the flooding strategy. Specifically, the licensee failed to conduct a self assessment with sufficient depth when reviewing the site's flooding strategy during a reasonable simulation and comprehensive flooding strategy site walk down in August 2012. (P.3(a))

<u>Enforcement</u>: Technical Specification 5.4.1 requires in part, that written procedures be established, implemented, and maintained covering the following activities: "the applicable procedures recommended in Regulatory Guide (RG) 1.33. Revision 2, Appendix A, February 1978." RG 1.33. Revision 2, Appendix A, Paragraph 6, addresses "Procedures for Combating Emergencies and Other Significant Events" and Item w addresses "Acts of Nature (e.g ., tornado, flood, dam failure, earthquakes)." An AV of TS 5.4.1 has been identified in that, from February 20, 1991, to November 21, 2012, the licensee failed to ensure procedures existed which ensured reactor vessel inventory could be maintained during external floods. Specifically, DOA 0010-04, "Floods," did not

account for reactor vessel inventory make up during an external flooding scenario up to and including the probable maximum flood event which could result in reactor vessel water level lowering below the top of active fuel. (**AV 05000237/2013002-02; 05000249/2013002-02, Deficiency In Abnormal Operating Procedures for Site Response to External Flooding Events**)

This finding does not represent an immediate safety concern. The licensee entered this issue into the corrective action program as IR 1485203, “NRC Question Regarding External Flooding.” Corrective actions completed include, implementation of a TSG-3, Attachment T, as of November 21, 2012, for reactor vessel inventory make up with the diesel flood pump and revising DOA 0010-04 requiring operators to isolate the reactor recirculation loops in order to minimize reactor coolant system leakage.

The inspectors intend on observing the licensees additional modifications and simulations of the flood strategy and the maintenance and control of flood strategy equipment through the annual external flooding inspection sample as governed by Inspection Procedure 711111.01, Adverse Weather Protection.

.4 Selected Issue Follow-Up Inspection: Corrective Actions Following Identification of a Potential Non-conservative Technical Specification

a. Inspection Scope

The inspectors reviewed plant design analysis and licensee actions to correct a potential non-conservative technical specification. Technical Specification 3.6.2.5, “Drywell to Suppression Chamber Differential Pressure,” potentially did not address design basis accident impacts on the containment. Following a Safety Communication from General Electric in July 2002, the licensee determined that TS 3.6.2.5 was potentially non-conservative in that the limiting condition for operations (LCO) action did not establish plant conditions which addressed the effects on containment of a design basis loss of coolant accident (LOCA). Specifically, in August 2002, the licensee implemented the administrative controls of NRC Administrative Letter 98-10, “Dispositioning of TSs that are Insufficient to Ensure Plant Safety,” for maintaining adequate differential pressure between the drywell and the suppression pool (torus). Subsequent to establishing administrative controls, the licensee has not submitted a license amendment to address the technical specification.

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

b. Observations

On July 26, 2002, General Electric published Safety Information Communication SC02-10, “Drywell-to-Wetwell Differential Pressure Control TS for Some Mark I Containments,” in which it was discussed that BWR with Mark I containments requiring a differential pressure of at least 1 psid between the drywell and the torus could be at risk due to the effects of pool swell loads during a design basis LOCA. Specifically, if the 1 psid differential pressure was not maintained, excessive water columns would form in the downcomer lines in the torus increasing the severity of pool swell loading on the containment during a design basis LOCA event. The Safety Communication also noted that the governing TS, TS 3.6.2.5, for plants with Improved Standard TSs allowed for continued operation for 24 hours with differential pressure conditions less than 1 psid

and then required the licensee to lower reactor power below 15 percent rated thermal power (RTP) within 8 hours. This subsequent step of reducing power below 15 percent RTP does not address the concern of a pressure surge in the drywell, resulting from a LOCA, driving water in the downcomer vents into the torus and creating excessive swell loading. Reactor pressure and temperature would remain constant when power was reduced below 15 percent RTP and as a result the drywell pressure conditions driving the water columns would be unaffected by the power change. In order to remove the driving force of the LOCA event on containment when differential pressure could not be maintained greater than 1 psid, the reactor coolant system would need to be cooled and depressurized.

Following this safety communication, the licensee took administrative actions establishing an Operations Standing Order, Unit 2/3 Standing Order 02-05, on August 14, 2002, which required placing the affected unit in Mode 3 (hot shutdown) within 55 hours and Mode 4 (cold shutdown) within 79 hours of entering Condition B of TS 3.6.2.5 which itself required reducing power below 15 percent RTP within 8 hours of entry. In October 2002, the license performed a bounding analysis to determine an acceptable allowed outage time based on the frequency of a seismically induced LOCA. This analysis was used in establishing Technical Requirements Manual (TRM) 3.6.c, “Drywell-to-Suppression Chamber Differential Pressure” which replaced Operations Standing Order 02-05 in October 2003. Section 3.6.c of the TRM required in part, that if drywell to torus differential pressure cannot be restored to greater than 1 psid within 67 hours, the affected unit must placed in Mode 3 in 12 hours and Mode 4 within 36 hours. This procedure remains in effect at the time of writing this report. Based on the incorrect assumption that the Boiling Water Reactor Owners Group (BWROG) was going to review this generic issue and recommend industry wide actions, the licensee failed to submit a license amendment request regarding the potentially non-conservative TS 3.6.2.5 and has operated with administrative controls in place since August 2002.

The inspectors noted the licensee performed a Plant Unique Analysis Report (PUAR) in May 1983 in response to NUREG 0661, Mark I Containment Long-Term Program. In this analysis the licensee calculated stresses on the torus and piping and components within the torus during a design basis LOCA under differential pressure conditions of greater than 1 psid and zero psid. In both instances, the stress and pressure values determined were well within the structural capacity of the torus and the components within it. This PUAR was reviewed and accepted by the NRC as documented by “Safety Evaluation By the Office of Nuclear Reactor Regulation Related to Mark I Containment Long-Term Program Pool Dynamic Loads Review Commonwealth Edison Company Docket Numbers 50-237/249,” dated September 18, 1985. With this approved analysis on record, the inspectors determined that TS 3.6.2.5 was not non-conservative as the site maintains an approved design analysis showing that even under zero differential pressure conditions in the containment at the initiation of a design basis accident, the containment would not be adversely affected. The inspectors determined that the actions of TS 3.6.2.5 were ineffective in addressing the design basis behind the existence of the requirement to maintain a differential pressure between the drywell and the torus and was potentially in conflict with licensee’s TRM 3.6.c actions. The licensee has entered this condition into its corrective action program as IR 117545 and is considering submitting a License Amendment Request to address this specification.

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

### .1 Unit 3 Downpower Due to Bearing Number 11 High Temperature

#### a. Inspection Scope

The inspectors reviewed the licensee's response to degrading conditions on Unit 3 main turbine bearing no. 11 first identified on January 27, 2013. At approximately 0530, main control room operators noted the start of an increasing trend in bearing no. 11 lubricating oil temperature, metal temperature and vibrations. Bearing no. 11 is located between the alterex exciter and the main generator supporting the Unit 3 main turbine shaft.

As conditions worsened in the early morning hours of January 28, 2013, operators reduced power a total of 220 MWe over the course of several hours in order to maintain bearing temperatures below upper limits. The potential to develop AC or DC voltages in the main turbine shaft in the vicinity of the main generator and exciter is not uncommon and is normally mitigated by grounding brushes which are tensioned in contact with the main turbine shaft. If the grounding brushes do not direct current flows to ground, then arcing between the shaft metal and the bearing metal will begin to degrade the bearing. The Operations personnel initial attempt to determine the status of the grounding brushes by a visual inspection could not identify the tension with which they were making contact with the shaft. This attempt to determine the status of the grounding brushes delayed the licensee's eventual recovery from the condition as management believed initially that the brushes were performing their function. On the evening of January 28, 2013, licensee electrical maintenance personnel performed troubleshooting under Work Order 01610877-09 and discovered that the shaft grounding brushes were not properly tensioned and corrected the condition. Bearing vibrations immediately returned to a stable, normal value. Bearing metal and oil temperatures stabilized at a slightly higher than normal value.

The licensee performed bearing analysis measurements with the bearing vendor and identified that the bearing had sustained minor damage, but would be able to continue operations at full power conditions for the remainder of the planned operating cycle. The licensee developed and implemented a detailed adverse condition monitoring plan for control room and field operators to perform in order to ensure bearing parameters remain stable during the operating cycle. In addition, a root cause analysis performed by the licensee identified that a miscommunication between licensee mechanical and electrical maintenance staff during plant restoration coming out of refueling outage D3R22 in December 2012 resulted in the shaft grounding brushes being installed on their brackets without being tensioned in place. Licensee corrective actions include revising model work orders for turbine shaft grounding equipment to include vendor manual steps to verify correct adjustment with completion signature required.

On January 29, 2013, the licensee restored Unit 3 to full power conditions. The inspectors observed main control room operations throughout this event, reviewed licensee troubleshooting plans, bearing analysis results, adverse condition monitoring, and the eventual root cause analysis.

Documents reviewed in this inspection are listed in the Attachment to this report.

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

b. Findings

No findings were identified.

4OA5 Other Activities

.1 (Closed) Unresolved Item 05000237/2011004-01; 05000249/2011004-01, "Classification of Emergency Diesel-Driven Flood Pump to Required Quality Standards"

a. Inspection Scope

The inspectors reviewed the unresolved item and additional documentation provided by the licensee regarding the safety classification status of the emergency diesel-driven flood pump to determine the proper safety classification of the pump.

b. Findings

Introduction: The inspectors identified an unresolved item regarding the safety classification of the emergency diesel-driven flood pump.

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

Title 10 of the Code of Federal Regulations, Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," establishes quality assurance requirements for the design, manufacture, construction, and operation of structures, systems and components that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. The pertinent requirements of this appendix apply to all activities affecting the safety-related functions of those structures, systems, and components.

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 questioned whether the test procedure for testing the emergency diesel-driven flood pump should have had acceptance criteria to demonstrate that the flood pump would 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 10 CFR 50.2, "Definitions," and based on the fact 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.

Upon further evaluation, the inspectors determined that flooding cannot be considered a Design Basis Event because Dresden's original license was issued describing the plant as a dry site. Even though the full term license for Unit 2 was issued incorporating the Systematic Evaluation Program (SEP) requirement for a Flooding Emergency Plan, the SEP did not require any backfits. The Emergency Flood Plan is a license requirement because it was incorporated into the Full Term Operating License for Unit 2. Therefore, since flooding is not a Design Basis Event, the emergency diesel-driven flood pump would not be required to be safety-related.

This Unresolved Item is closed.

## 4OA6 Management Meetings

### .1 Exit Meeting Summary

On April 8, 2013, 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 verified that no proprietary information was retained by the inspectors or documented in this report.

### .2 Interim Exit Meetings

Interim exits were conducted for:

- The inspection results for the areas of radiological hazard assessment and exposure controls; and radioactive gaseous and liquid effluent treatment with Mr. S. Marik, Plant Manager, on February 1, 2013.
- The preliminary White determination for the finding associated with the plant's flooding response procedure with Mr. S. Marik on April 19, 2013.

The inspectors confirmed that none of the potential report inputs discussed were considered proprietary.

ATTACHMENT: SUPPLEMENTAL INFORMATION

## SUPPLEMENTAL INFORMATION

## KEY POINTS OF CONTACT

Licensee

D. Czufin, Site Vice President
S. Marik, Station Plant Manager
J. Biegelson, Engineering
H. Bush, Radiation Protection Manager
J. Cady, Radiation Protection Manager
P. Chambers, Dresden Licensed Operator Requalification Training Lead
P. DiSalvo, GL 89-13 Program Owner
H. Do, Corporate ISI Manager
D. Doggett, Emergency Preparedness Manager
H. Dodd, Regulatory Assurance Manager
J. Fox, Design Engineer
D. Glick, Radioactive Material Shipping Specialist
G. Graff, Nuclear Oversight Manager
M. Hosain, Site EQ Engineer
R. Johnson, Chemist RETS/ODCM
B. Kapellas, Operations Director
D. Ketchledge, Engineering
J. Knight, Director, Site Engineering
M. Knott, Instrument Maintenance Manager
J. Kish, Site ISI
S. Kvasnicka, NDE Level III
D. Leggett, Chemistry Manager
T. Mohr, Supervisor, Engineering Programs
P. Mankoo, Radiation Protection
G. Morrow, Shift Operations Superintendent
M. McDonald, Maintenance Director
T. Mohr, Programs Engineering Manager
P. O'Brien, Regulatory Assurance – NRC Coordinator
D. O'Flanagan, Security Manager
M. Otten, Operations Training Manager
R. Ruffin, Licensing Engineer
J. Sipek, Work Control Director
R. Sisk, Buried Pipe Program Owner
L. Torres, Engineering

Nuclear Regulatory Commission

J. Cameron, Chief, Division of Reactor Projects, Branch 6
L. Kozak, Senior Risk Analyst

IEMA

R. Zuffa, Illinois Emergency Management Agency

## ITEMS OPENED, CLOSED AND DISCUSSED

Opened

| | | |
|---|---|---|
| 05000237/2013002-01<br>05000249/2013002-01 | FIN | Failure to Include Adequate Acceptance Criteria in a Surveillance Test (1R01.3) |
| 05000237/2013002-02<br>05000249/2013002-02 | AV | Deficiency In Abnormal Operating Procedures for Site Response to External Flooding Events (Section 4OA2.3) |

Closed

| | | |
|---|---|---|
| 05000237/2013002-01<br>05000249/2013002-01 | FIN | Failure to Include Acceptance Criteria in a Surveillance Test. (1R01.3) |
| 05000237/2011003-01<br>05000249/2011003-01 | URI | Failure to Include Adequate Acceptance Criteria in a Surveillance Test (1R01.3) |
| 05000237/2011004-01<br>05000249/2011004-01 | URI | Classification of Emergency Diesel-Driven Flood Pump to Required Quality Standards (4OA5) |

Discussed

| | | |
|---|---|---|
| 05000237/2004010-02<br>05000249/2004010-02 | NCV | Source of Make-up Water (1R01.3) |
| 05000237/2006010-04<br>05000249/2006010-04 | URI | Full Flow Testing of the Diesel Driven Flood Pump at Design Conditions (1R01.3) URI was closed in IR 05000237/2007003, 05000249/2007003 |
| 05000237/2007003-04<br>05000249/2007003-04 | NCV | Failure to Identify and Correct Issues with the Operation and Testing of the Diesel Driven Pump Used to Respond to External Flooding (1R01.3) |

## 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)

- OP-AA-108-107-1001, "Station Response to Grid Capacity Conditions," Revision 4
- OP-AA-102-102, "General Areas Checks and Operator Field Rounds," Revision 12
- IR 1465907, "NRC Senior Resident Question"

1R04 Equipment Alignment (71111.04)

- DOP 1300-M1/E1, "Unit 2 Isolation Condenser System," Revision 17
- Drawing M-28, "Diagram of Isolation Condenser Piping"
- NRC Inspection Report 05000219/2012005, 10/01/2012 – 12/31/2012; Exelon Energy Company, LLC, Oyster Creek Generating Station
- Exelon Oyster Creek Procedure 2400-GMM-3900.52
- IR 1487125, "U2 Isolation Condenser Support Nut Engagement Deficiency"
- DOP 1300-M1/E1, "Unit 3 Isolation Condenser," Revision 23
- WO 1571975-13, Weld Map for HPCI Drain Line Replacement of Chrome-Moly Piping with Stainless Steel Pipe
- DOP 2300-M1/E1, "U2 HPCI System Checklist," Revision 38
- Drawing M-51, "Diagram of High Pressure Coolant Injection Piping"
- SA-AA-141, "Management & Control of Hexavalent Chromium During Welding, Cutting, and Grinding Activities," Revision 1

1R05 Fire Protection (71111.05)

- CALCULATION: DRE97-0105, Determination of Combustible Loading, Revision 8
- Dresden Station Units 2 and 3, Commonwealth Edison Company, Fire Protection Reports, Volume 4, Interim Measures/Exemption Requests, Section 3.5, Justification for Lack of Complete Detection and Suppression in Fire Area TB-II.
- Dresden Station Units 2 and 3, Commonwealth Edison Company, Fire Protection Reports, Volume 1, Updated Fire Hazards Analysis
- Dresden Station Units 2 and 3, Commonwealth Edison Company, Fire Protection Reports, Volume 4, Interim Measures/Exemption Requests, Section 3.5, Justification for Lack of Complete Detection and Suppression in Fire Area RB2-II
- IR 1485984, "NRC Question on Fire Protection"
- IR 1485977, "Fire Protection – Pre-Fire Plans"

1R06 Flooding (71111.06)

- WO 01505382, "Need WR: U3 CCSW VLT Drain CHK VLV Replace with SR Comp."
- WO 01297399-01, D3 8Y PM Perform Check Valve Inspection 3-4999-75"
- IR 1477386, "Threaded Elbow Degraded and in Need of Replacement"
- IR 1477499, "Slow Draining During Performance of DOS 4400-01"

1R11 Licensed Operator Requalification Program (71111.11)

- IR 1467190, “Apparent Cause Report – Dresden High Exam Failure Rate During LORT Cycle Exam”

1R12 Maintenance Effectiveness (71111.12)

- IR 1448857, “Request Design Engineering to Review HELB Barriers for MRule”
- IR 1486451, “NRC Identified Editorial Error in MRULE A3 Assessment”
- Maintenance Rule Periodic Assessment no.9 (10CFR50.65(a)(3) Assessment) Assessment Period 10/1/2010 – 9/30/2012
- DES 6700-09, Revision 23, “Inspection and maintenance of General Electric MC-4.76 Horizontal Draw-out Metal Clad Switchgear”
- MA-DR-067-002, “Circuit Breaker Control” Revision 0
- MA-DR-725-113, “Inspection and Maintenance of General Electric 4 KV Magne-Blast Circuit Breakers Types AMH4.76-250 (Horizontal Drawout), Revision 04
- MA-DR-773-302, “Dresden Standby Diesel Generator 2 and 4 KV ACB 2422 Control Circuit Checks” Revision 09
- IR 1374783, “Maintenance Rule Function Z67-3 At Risk”
- IR 1364609, “Result of 4KV BKR Delayed Closing Failure Analysis Report”
- IR 1303972, “4KV Breaker Cubicle MOC Switch Parts Issues”
- IR 1282685, “Breaker Did Not Reclose in Test Position During Surveillance”
- IR 1281382, “Potentially Deficient Prop Spring Bracket hardware 4KV BKR”
- IR 1280299, “4KV Cubicle MOC Switch Rubber Bumper Degrading (New)”
- IR 1251072, “1-6712-8, Main Feed to Bus 15 from Bus 12 BRKR Will Not Close”
- IR 1216097, “HCCT no.3 Pump Motor Breaker Will Not Rack In”
- IR 1191551, “4KV Merlin Gerin Circuit Breaker “B” Phase Bottle Cracked”
- IR 1472026, “Over Voltage Relay Flag Is Up”
- IR 1437824, “4KV Breaker UTC 2861736 Does Not Charge Consistently”
- IR 1361972, “Results of Troubleshooting, 4KV Breaker Charging Motor”
- IR 1399081, “NRC Concern – Historical Operability for 2A CCSW PP Breaker”
- IR 13339668, “Product Advisory Letter Not Incorporated Into Breaker Insp.”
- IR 1232355, “Streamline 4160V Breaker Trip/Close Fuse Replacements”
- IR 1280668, “Bus 23 Undervoltage Load Shed”
- WO 1064044, “16Y PM Overhaul 4KV Breaker UTC 997129”
- IR 1482952, “Engineering Requests ACE – 125VDC PCM Template”
- IR 1437844, “TS Required Paperwork Not Submitted for Record Retetention”
- IR 1443849, “Feed Breaker Not Operating Properly D3R22SU”
- IR 1471251, “2D LPCI Pump Breaker Will Not Charge Springs”
- IR 1477941, “MRULE: 4KV Breaker Performance Criteria is Non-Conservative”
- IR 1443228, “MRULE: System 67, MPFF is a RMPFF”
- IR 1441171, “Bus 33 CUB 10, Bus 33-1 Feed Breaker Won’t Close”

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

- OP-AA-108-117 Rev.3
- IR 1478889, “NRC Inspector Questioned Shift on PPW for SBLC”
- Ops Policy 02, Attachment B – Protected Equipment List

1R15 Operability Determinations and Functional Assessments (71111.15)

- Calculation DRE98-0030, Rev. 0, “Determination of Setpoint of CST Low-Low Level Switches to Prevent Potential Air Entrainment form Vortexing During HPCI Operation.”
- IR 1453700, “Key Calc Review Identifies Issues in DRES98-0030”
- EC 391829, “Evaluation of Issues Identified in Calculation DRE98-0030,” Revision 0
- EC 360021, “Reroute of the Buried HPCI Cross-Tie Piping,” Revision 1
- Drawing M-197, Sheet 11 “Piping Isometric HPCI Cross-Tie 2/3-3327A-16”
- Drawing M-197, Sheet 1 “Outdoor Piping”
- WO 01575845, “Dresden 3 Quarterly Technical Specification HPCI Motor Operated Valve Operability Surveillance,” Revision 1
- IR 1455933, “U3 HPCI Valve Found in Alert Range During Valve Timing”
- Drawing M-347. “Diagram of Reactor Feed Pump”
- Drawing M-374, “Diagram of High Pressure Coolant Injection Piping”
- ER-AA-321, “IST Valve Evaluation Form,” Revision 12
- Dresden Updated Final Safety Analysis Report (UFSAR), Section 5.2.5, “Detection of Leakage Through Reactor Coolant Pressure Boundary”
- IR 128973, “NRC Drywell closeout inspection items”
- OP-DR-108-111-1003, “Drywell Leakage Troubleshooting,” Revision 03
- DOS 1600-29, “Unit 2 and 3 Drywell Temperature Surveillance,” Revision 5
- DAN 902(3)-5 F-3, “Rod Drive Temp Hi,” Revision 15
- DAN 902(3)-4 C-17,”Drywell Equip Sump Temp Hi,” Revision 8
- IR 1487125, “U2 Isolation Condenser Support Nut Engagement Deficiency”
- IR 1487131, “U3 Isolation Condenser Support Nut Engagement Deficiency”
- MA-MW-736-600, “Torquing and Tightening of Bolted Connections”, Revision 5
- Drawing ISI-201, “Inservice Inspection Class II Isolation Condenser Piping”
- Drawing ISI-206, “Inservice Inspection Class II Isolation Condenser Piping”
- Drawing ISI-307, “Inservice Inspection Class III Isolation Condenser and Vent Piping”
- Drawing ISI-305, “Inservice Inspection Class III Isolation Condenser and Vent Piping”
- Drawing Hanger M-1163D-553
- Drawing Hanger M-1199D-1022
- Drawing Hanger M-1199D-1023
- Drawing Hanger M-1199D-1024
- Drawing Hanger M-1163D-554
- Drawing Hanger M-1163D-555
- EC 388891, “2013: Unit 2 SR 480V Bucket Replacement Project,” Revisions 0 and 1

1R18 Plant Modifications (71111.18)

- 50.59 Screening No. 2013-0064, “Unit 3 Main Steam Line Isolation Valve Closure Scram Circuit Functional Test”
- Drawing 12E-3466, Schematic Diagram Reactor Protection System Channel “B” Scram & Auxiliary Trip Relays, Sheet No. 1
- Drawing 12E-3464, Schematic Diagram Reactor Protection System Channel “B” Trip Aux. Relays, Sheet No. 2
- IR 1484861, “U3 ‘C’ Main Steam Line Limit Switch Failure”
- DOS 0500-27, “Unit 3 main Steam Line Isolation Valve Closure Scram Circuit Functional Test,” Revision 1

### 1R19 Post-Maintenance Testing (71111.19)

- IR 1465026, “Broken Motor Cooling Blades”
- WO 1418376, “Dresden Unit 2 Two Year PM Standby Diesel Generator Inspection”
- IR 1475119, “Re-occurring Unexpected Unit 2 EDG Alarm”, February 14, 2013
- DOS 6600-01, Diesel Generator Surveillance Tests, Revision 122
- WO 1605861, “D2 Quarterly TS HPCI Pump Operability Test and IST Surveillance”
- IR 1488336, “NRC Identified Housekeeping Issues in the U2 HPCI Room”
- WO 1423633, “D3 2Y TS HPCI Pump Comprehensive Oper Test and IST Surv”
- IR 1490995, “Unexpected Alarm HPCI Room Sump Level High”
- IR 1492807, “Newly Replaced Valve Leaks Past Seat for D2/3 EDG”
- WO 1424316-03, “D2/3 2 year PM Diesel Generator Engine Temperature Instrument CAL – Check for Leaks”
- WO 681423, “D2/3 4year Inspect Cubicle 3 at Bus 40 (Bus Tie to Bus 23-1) – Perform PMT on Bus 40C”
- DOS 6600-01, “Diesel Generator Surveillance Tests,” Revision 122

### 1R22 Surveillance Testing (71111.22)

- DOS 1500-10, “LPCI System Pump Operability and Quarterly Test with Torus Available and Inservice Test,” Revision 67
- DIS 1500-09, Revision 19, “LPCI Reactor Recirculating Pump A and B Differential Pressure Indicating Switch Calibration and Channel Functional Test”
- IR 1462734, “Data Transfer Error Discovered During Review of DIS 1500-09”
- IR 1461469, “Old Word Perfect Symbol Caused < or = to become > or =,” January 11, 2013
- Drawing M-357, Sheet 2, “Diagram of Nuclear Boiler & Reactor Recirculating Piping”
- Drawing 12E-3437A, “Schematic Diagram LPCI Containment Cooling System”
- Drawing 12E-3438A, “Schematic Diagram LPCI Containment Cooling System”
- DOP 2000-180, “Drywell Sump Operation With Unit On-Line,” Revision 04
- Appendix A, “Unit Daily Surveillance Log, Attachment A, Eight Hour Shifts,” Revision 129
- DES 0040-02, “600 Volt Butyl Cable EQ Surveillance,” Revision 10
- DFPS 4123-05, “2/3 Diesel Fire Pump Operability,” Revision 50
- Electrical Drawing 12E-2750A, Sheet 1, “Wiring Diagram Feedwater and Recirculation Panel 9”
- Electrical Drawing 12E-2424, Sheet 1, “Schematic Diagram Recirculating Pump Speed Control”
- WO 1396238, “D2 2Y PM&C Recirc Flow Dual Limiter 262-26B”
- WO 1579759, “D2 QTR TS LPCI System Pump Run and IST Surveillance”

### 1EP6 Drill Evaluation (71114.06)

- IR 1472861, “EP DEP Failure During TSC Focus Area Drill,” February 8, 2013
- IR 1474238, “NOS ID: DEP Failures Identified as Level 4 Issue Reports,” February 12, 2013

### 2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

- RWP 10014505; 2013 Radwaste Concentrated Waste Vault Sludge Removal; Revision 0
- Unit 2/3 Radwaste EL 517’ Reboiler Area Survey Map; January 8, 2013
- Unit 2/3 Radwaste EL 517’ Reboiler Area Survey Map; January 16, 2013
- Unit 3 reactor 570’ Fuel Pool Cleanup Pump/Heat Exchanger Area; January 8, 2013

- RWP 10014505; 2013 Radwaste Concentrated Waste Vault Sludge Removal; WO 1221622; Vacuum Vessel Full Sludge from the Concentrated Waste Vault; ALARA Plan; January 29, 2013
- RWP 10014505; 2013 Radwaste Concentrated Waste Vault Sludge Removal; Barnhart to Remove Floor Plugs in the “B” Concentrator Vault for Prep of the CW Vault; January 24, 2013
- RWP 10014505; 2013 Radwaste Concentrated Waste Vault: Decon Area Remove Misc Bags and Parts from Area at 517’ Reboiler; ALARA Briefing Checklist RP-AA-401

<u>2RS6 Radioactive Gaseous and Liquid Effluent Treatment (71124.06)</u>

- CY-DR-170-220; Revision 5; Unit 2 and 3 Reactor Building Vent Noble Gas Sampling; January 31, 2013
- DIS-1700-14; Unit2/3 Reactor Building Vent Stack SPING Calibration; November 2, 2012
- RP-AA-605; Waste Stream Results Review Part 61; May 11, 2012
- R-01398192; Document Request of the Licensee’s Process Monitor Setpoint Bases; August 7, 2012
- DIS-1700-14; Unit2/3 Main Chimney SPING Calibration; February 2, 2012
- DIS-1700-14; Unit2/3 Main Chimney SPING Calibration; February 19, 2012
- WC-AA-104;Quarterly Tech Spec Reactor Building Vent Radiation Monitor Calibration and Functional Tests; January 24, 2013
- L53233; Teledyne Brown; Report of Analysis/Certificate of Conformance; Waste Surge Tanks Sampling Wells; January 15, 2013
- L533074; Teledyne Brown; Report of Analysis/Certificate of Conformance; Steam Dryer Mausoleum Sampling Wells; January 30, 2013,
- AR-01322619; Unable to Complete Sample Line Inspection for Chimney SPING; February 3, 2012
- AR-01322619; Engineering Evaluation for Visual Inspection of Sample Lines from the SPING to Chimney; March 23, 2012
- WO 1337345; 24 Month Tech Spec.(TS) Reactor Building Vent Sampler Flowmeter Calibration; May 5, 2012
- WO 1575796; D3 Quarterly TS Reactor Vent Radiation Monitor Calibration and Functional; December 20, 2012
- WO 1236618; D3 TS Reactor Building Vent Sampler Flowmeter Calibration; April 15, 2011
- AR-01305481; D2/3 Main Chimney SPING not Responding During Calibration; February 4, 2012
- AR-01329443; Unexpected Alarm on the Liquid Process Rad Monitor; February 12, 2012
- AR-01330334; Liquid Process Rad Monitor Downscale; February 22, 2012
- AR-01397142; Liquid Process Rad Monitor Alarms on Unit-2 and Unit-3; August 4, 2012
- AR-01392741; HPGE Detector Failed Multiple Performance Checks; July 27, 2012
- AR-01392819; Out of Calibration RP Instrument Found in Plant; July 25, 2012

<u>4OA1 Performance Indicators</u> (71151)

- NEI 99-02, “Regulatory Assessment Performance Indicator Guideline”, Revision 6
- Unit 2 and 3 Performance Indicator data for First through Fourth Quarter 2012

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

- EP-AA-1004, “Radiological Emergency Plan Annex For Dresden Station,” Revision 29
- IR 1453073, “NRC Question on TSG 3 Strategies”
- IR 1469502, “CDBI-2013 IC Diesel Makeup Pump Building Foundation Buoyancy”

- IR 1485203, “NRC Question Regarding External Flooding”
- IR 1482141, “EC 391644 Approved Prior to Affected Calculation Approval”
- EP-AA-1004, “Exelon Nuclear Radiological Emergency Plan Annex For Dresden Station”, Revision 29
- DOA 0010-04, “Floods”, Revision 32
- DOA 0010-04, “Floods”, Revision 37
- Calculation DRE01-030, “Probable Maximum Flood Effects at the ISFSI Pad,” Revision 0
- Drawing M-23 Sheet 3, Diagram of Fire Protection Piping
- Drawing M-33, Diagram of Standby Liquid Control Piping
- TSG-3, “Operational Contingency Action Guidelines,” Revision 10
- TSG-3, Attachment H, “RPV Injection Using Portable Diesel Driven Pump”
- TSG-3, Attachment T, “Provide RPV Make Up From Fire Protection System Via SBLC”
- COM-02-041-2, Plant Unique Analysis Report Volume 2, “Suppression Chamber Analysis,” Revision 0, May 1983
- SC02-10, GE Nuclear Energy Safety Communication, “Drywell-to-Wetwell Differential Pressure Control Technical Specification for Some Mark I Containments,” July 26, 2002
- SA-1115, Significance of Seismic-Induced LOCAs at Dresden,” Revision 1
- OP-AA-102-104, U2/3 Standing Order, “Tech Spec LCO 3.6.2.5 Supplemental Administrative Actions,” Revision 0
- Commitment Letter from Commonwealth Edison to Mr. Paul O’Connor, “SEP Topic II-3.B, Flooding Potential and Protective Requirements; II-3.B.1, Capability of Operating Plants to Cope with Design Basis Flood Conditions; and II-3.C, Safety Related Water Supply (Ultimate Heat Sink), “ Dated November 17, 1982
- Letter from NRC to Licensee, “Request for a Written Response to NRC Observations and Concerns Regarding Dresden Station Response Plan for External Flooding Events,” Dated November 1, 2012
- Letter from Mr. David Czufin to NRC, “Response to NRC Request for a Written Response to NRC Observations and Concerns Regarding Dresden Station Response Plan for External Flooding Events,” Dated December 1, 2012
- Letter from NRC to Licensee, “Acknowledgement of Response to NRC Request for a Written Response to NRC Observations and Concerns Regarding Dresden Response Plan for External Flooding Events,” Dated January 3, 2013
- Letter from Mr. David Czufin to NRC, “Response to Acknowledgement of Response to NRC Request for a Written Response to NRC Observations and Concerns Regarding Dresden Response Plan for External Flooding Events,” Dated January 31, 2013
- GE Safety Information Communication, “Drywell to Wetwell Differential Pressure Control Technical Specification for Some Mark I Containments”, July 26, 2002
- Letter from J. Henry, “Evaluation of Entry Into Technical Specification Limiting Condition of Operation 3.6.25 for Drywell to Suppression Chamber Differential Pressure,” October 16, 2002
- IR 117545, “DW to Torus DP Control Tech Spec for a Mark I Containment”
- 50.59 Screening Form, TRM 3.6 C, 2003-0349, Revision 0
- IR 1490293, “De-energized Relay Picked Up”
- Letter from NRC to Licensee, “Mark I Containment Long Term Program,” dated September 18, 1985

<u>4OA3 Follow Up of Events and Notices of Enforcement Discretion</u> (71153)

- IR 1467631, “Unexpected Alarm; 903-8 E-12 U3 Gen/Exc Ground”
- IR 1468057, “U3 Turbine Shaft Grounding Brushes Found Not Fully Tensioned”
- IR 1468569, “Follow-Up Actions for Unit 3 Bearing no. 11 Issue”

- IR 1468765, “U3 Exciter BRG no. 11 Lube Oil Flow Measurement Not Accurate”
- Root Cause Report 1468057-04, “Unit 3 Bearing 11 Rising Vibration, Oil & Metal Temperatures due to Work Order Referenced Vendor Manual Steps without Requiring Step by Step Worker Sign Offs”
- OP-AA-108-111, “Unit 3 Bearing 11 Metal Temperature and Vibration,” Revision 1

## LIST OF ACRONYMS USED

| | |
|---|---|
| AC | alternating current |
| ADAMS | Agencywide Document Access Management System |
| ALARA | As-Low-As-Reasonably-Achievable |
| AOP | Abnormal Operating Procedure |
| ASME | American Society of Mechanical Engineers |
| AV | Apparent Violation |
| BWR | Boiling Water Reactor |
| CAP | Corrective Action Program |
| CCSW | Containment Cooling Service Water |
| CDF | Core Damage Frequency |
| CFR | Code of Federal Regulations |
| CRD | Control Rod Drive |
| DC | direct currnet |
| DOA | Dresden Abnormal Operating Procedure |
| DRP | Division of Reactor Projects |
| EAL | Emergency Action Level |
| ECCS | Emergency Core Cooling System |
| EDG | Emergency Diesel Generator |
| FIN | Finding |
| FLEX | Diverse and FLEXIBLE equipment availability program |
| HPCI | High Pressure Coolant Injection |
| IC | Isolation Condenser |
| IMC | Inspection Manual Chapter |
| INPO | Institute of Nuclear Power Operations |
| IP | Inspection Procedure |
| IR | Inspection Report |
| IR | Issue Report |
| ISI | Inservice Inspection |
| JLD | Japan Lessons Learned |
| LCO | Limiting Condition for Operation |
| LOCA | Loss of Coolant Accident |
| LLC | Limited Liability Corporation |
| LORT | Licensed Operator Requalification Training |
| LPCI | Low Pressure Coolant Injection |
| MSL | Mean Sea Level |
| NCV | Non-Cited Violation |
| NEI | Nuclear Energy Institute |
| NRC | U.S. Nuclear Regulatory Commission |
| OCA | Owner Controlled Area |
| PARS | Publicly Available Records System |
| PI | Performance Indicator |
| PM | Planned or Preventative Maintenance |
| PMF | Probable Maximum Flood |
| PMT | Post-Maintenance Testing |
| psid | pounds per square inch differential |
| psig | pounds per square inch gauge |
| PUAR | Plant Unique Analysis Report |
| RP | Radiation Protection |

| | |
|---|---|
| RPV | Reactor Pressure Vessel |
| RTP | Rated Thermal Power |
| SBLC | Standby Liquid Control |
| SDP | Significance Determination Process |
| SEP | Systematic Evaluation Program |
| SERP | Significance and Enforcement Review Panel |
| SSC | Systems, Structures, and Components |
| SRA | Senior Risk Analyst |
| TAF | Top of Active Fuel |
| TS | Technical Specification |
| TSG | Technical Support Guidance |
| UFSAR | Updated Final Safety Analysis Report |
| URI | Unresolved Item |
| Vdc | Volts direct current |
| WO | Work Order |

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Sincerely,

***/RA by Kenneth G. O'Brien for/***

Steven A. Reynolds, Director
Division of Reactor Projects

Docket Nos. 50-237, 50-249
License Nos. DPR-19 and DPR-25

Enclosure: Inspection Report 05000237/2013002, 05000249/2013002
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*OE and NRR concurrence on 4/29 via e-mail

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Letter to M. Pacilio from J. Cameron dated May 7, 2013.

SUBJECT: DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3, INTEGRATED INSPECTION REPORT 05000237/2013002, 05000249/2013002

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