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

August 4, 2011

Mr. Michael J. Pacilio
Senior Vice President, Exelon Generation Company, LLC
President and Chief Nuclear Officer (CNO), Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3, INTEGRATED INSPECTION REPORT 05000237/2011003; 05000249/2011003

Dear Mr. Pacilio:

On June 30, 2011, the U.S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your Dresden Nuclear Power Station, Units 2 and 3. The enclosed report documents the results of this inspection, which were discussed on July 14, 2011, with Mr. D. Czufin, and other members of your staff.

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

Based on the results of this inspection, one self-revealed and five NRC-identified findings of very low safety significance were identified. Each of these findings involved a violation of NRC requirements. However, because of their very low safety significance, and because the issues were entered into your corrective action program, the NRC is treating the issues as non-cited violations (NCVs) in accordance with Section 2.3.2 of the NRC Enforcement Policy. Additionally, a licensee-identified violation is listed in Section 4OA7 of this report.

If you contest the subject or severity of any of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Dresden Nuclear Power Station.
In addition, if you disagree with the cross-cutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region III, and the NRC Resident Inspector at the Dresden Nuclear Power Station.

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

Sincerely,

***/RA/***

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

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

Enclosure: Inspection Report 05000237/2011-003; 05000249/2011-003
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

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| Docket Nos: | 05000237; 05000249 |
| License Nos: | DPR-19; DPR-25 |
| Report No: | 05000237/2011-003; 05000249/2011-003 |
| Licensee: | Exelon Generation Company, LLC |
| Facility: | Dresden Nuclear Power Station, Units 2 and 3 |
| Location: | Morris, IL |
| Dates: | April 1 through June 30, 2011 |
| Inspectors: | C. Phillips, Senior Resident Inspector<br>D. Meléndez-Colón, Resident Inspector<br>J. Draper, Reactor Engineer<br>M. Bielby, Senior operations Engineer<br>C. Moore, Operations Engineer<br>C. Tilton, Senior Reactor Inspector<br>J. Corujo-Sandín, Reactor Engineer<br>M. Munir, Reactor Inspector<br>A. Dahbur, Senior Reactor Inspector<br>R. Winter, Reactor Inspector<br>T. Go, Health Physicist Inspector |
| Approved by: | M. Ring, Chief<br>Projects Branch 1<br>Division of Reactor Projects |

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## SUMMARY OF FINDINGS

IR 05000237/2011-003, 05000249/2011-003; 04/01/2011 – 06/30/2011; Dresden Nuclear Power Station, Units 2 & 3, Heat Sink Performance, Event Follow-Up, Licensee-Identified Violations.

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

### A. NRC-Identified and Self-Revealed Findings

#### Cornerstone: Initiating Events

- Green. A finding of very low safety significance and associated non-cited violation of Technical Specification 5.4.1 was self-revealed for a Nuclear Station Operator (NSO) failing to follow step G.14.a of procedure DOP 0600-06, "Feedwater Regulating Valve (FWRV) Operation," Revision 39. This resulted in a reduction in Unit 2 reactor water level. The licensee took the following immediate corrective actions. The NSO placed the 2B FWRV in manual and restored reactor water level. The NSO was relieved from duty.

  The finding was determined to be more than minor because the finding could be reasonably viewed as a precursor to a significant event. Specifically, the event could have led to a reactor scram. The inspectors concluded this finding was associated with the Initiating Events Cornerstone. The inspectors evaluated the finding using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," Table 4a, for the Initiating Events Cornerstone. Since the finding did not contribute to both the likelihood of a reactor scram and the likelihood that mitigation equipment or functions would not be available, the finding screened as Green. This finding had a cross-cutting aspect in the area of human performance, work practices, because the licensee did not ensure the proper use of human error prevention techniques. (H.4(a)) (Section 4OA3)

#### Cornerstone: Mitigating Systems

- Green. A finding of very low safety significance and associated non-cited violation (NCV) of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified by the inspectors for the licensee's failure to establish adequate instructions for inspecting bay 13 and portions of the intake structure surrounding the diesel generator cooling water pumps. Specifically, the procedure that provides guidance for inspecting these structures lacked specific instructions on how to detect and record degradation by erosion and corrosion. The licensee entered this issue into the corrective action program and initiated procedure revisions to provide further direction for capturing the degradation of these structures and related components.

The performance deficiency was determined to be more than minor because, if left uncorrected, it would have the potential to lead a more significant safety concern. The finding screened as of very low safety significance because it was a qualification deficiency confirmed not to result in loss of operability or functionality. Specifically, a qualitative assessment of historic surveillance reports found the documented results acceptable. The inspectors determined the cause of this finding did not represent current licensee performance and no cross-cutting aspect was assigned. (Section 1R07.b(1))

- Green. A finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," was identified by the inspectors for the licensee's failure to establish adequate acceptance criteria for testing equipment relied upon to mitigate the consequences of a dam failure. Specifically, the acceptance criteria in DOS 0010-01, "Dresden Dam Failure Equipment Test," did not consider additional steps required to demonstrate the ability of the screen refuse pumps to deliver water to the enclosure for the safety-related pumps to support operability of the isolation condensers. The licensee entered this issue into the corrective action program and initiated procedure revisions to include these additional steps in the procedure's acceptance criteria.

  The performance deficiency was determined to be more than minor because it adversely affected the availability, reliability, and capability of mitigating systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). The finding screened as of very low safety significance because it was a qualification deficiency confirmed not to result in loss of operability or functionality. Specifically, the licensee estimated that the time it would take to perform the additional steps not included in the procedure's acceptance criteria was within the time required. The inspectors determined the cause of this finding did not represent current licensee performance and no cross-cutting aspect was assigned. (Section 1R07.b(2))

- Green. A finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," was identified by the inspectors for the licensee's failure to establish adequate instructions for coping with the consequences of a dam failure. Specifically, DOA 0010-01, "Dresden Lock and Dam Failure," lacked controls for the configuration of the associated foreign material exclusion (FME) screens and lacked specific instructions on how to shed load off the emergency diesel generators and restore power to the bus associated with equipment relied upon during this event. The licensee entered this issue into their corrective action program and initiated procedure revisions to provide adequate controls on the configuration of the FME screens and to provide further guidance on restoring power to the refuse screen pumps.

  The performance deficiency was determined to be more than minor because it adversely affected the availability, reliability, and capability of mitigating systems that respond to initiating events to prevent undesirable consequences. The finding screened as of very low safety significance because it was a qualification deficiency confirmed not to result in loss of operability or functionality. Specifically, the licensee estimated that the additional time required to install the inner screens and restore power to the screen refuse pumps was within the required 2-hours. The inspectors determined the cause of this finding did not represent current licensee performance and no cross-cutting aspect was assigned. (Section 1R07.b(3))

- Green. A finding of very low safety significance and associated NCV of 10 CFR Part 50.65(b)(2)(ii), "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," was identified by the inspectors for the licensee's failure to adequately scope a non-safety-related component relied upon to mitigate an accident. Specifically, the licensee failed to include the non-safety-related screen refuse pumps, which are credited as a support component during a dam failure for the safety-related isolation condensers, as part of their maintenance effectiveness program. As a corrective action, the licensee initiated IR 1221421, to evaluate the need to include the screen refuse pumps into their maintenance rule program.

  The performance deficiency was determined to be more than minor because it adversely affected the availability, reliability, and capability of mitigating systems that respond to initiating events to prevent undesirable consequences (i.e., core damage.) The finding screened as of very low safety significance because it was a qualification deficiency confirmed not to result in loss of operability or functionality. Specifically, the licensee performed further review of related maintenance and provided reasonable assurance the screen refuse pumps had not experienced a complete loss of function. The inspectors determined the cause of this finding did not represent current licensee performance and no cross-cutting aspect was assigned. (Section 1R07.b(4))

- Green. The inspectors identified a finding of very low safety significance and NCV of Technical Specification 3.3.2.1.1, Required Action B.1, for the licensee's failure to place one Rod Block Monitor (RBM) channel on Unit 3 in trip within 1 hour as required upon meeting Condition B, "Two RBM channels inoperable." The licensee performed troubleshooting on the issue and declared the RBM inoperable to repair the failed component.

  The finding was determined to be more than minor because, if left uncorrected, if could lead to a more significant safety concern. Specifically, if a RBM channel had fewer than three real inputs, but was counting false inputs due to the shorted diode, it would be unable to fulfill its Inop function, and would not insert a rod block to prevent a rod withdrawal error. The inspectors performed a Phase 1 Significance Determination in accordance with IMC 0609, Attachment 4. The inspectors concluded that this finding affected the Mitigating Systems Cornerstone since it was a degradation of reactivity control. The finding screened as Green, or very low safety significance, in Table 4a of Attachment 4, since it is not a design or qualification deficiency and did not represent an actual loss of safety system function. This is because the RBM maintained a sufficient number of real inputs and because the licensee's Rod Withdrawal Error analysis does not assume the RBM function. The inspectors determined that this finding has a cross-cutting aspect in the area of Problem Identification and Resolution in the component of Corrective Action Program, since it involved the licensee's failure to thoroughly evaluate problems. (P.1(c)) (Section 4OA3)

## B. Licensee-Identified Violations

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

## REPORT DETAILS

**Summary of Plant Status**

**Unit 2**

On May 12, 2011, load was reduced to approximately 94 percent electrical due to low main condenser vacuum caused by prolonged high intake temperature due to seasonal variations. The unit returned to full power operation on May 13, 2011.

On May 21, 2011, load was reduced to approximately 61 percent electrical for a control rod pattern adjustment and planned maintenance for the feedwater regulating valves. The unit returned to full power operation on May 23, 2011.

On June 26, 2011, load was reduced to approximately 84 percent electrical for a control rod pattern adjustment. The unit returned to full power operation on the same day.

**Unit 3**

On May 12, 2011, load was reduced to approximately 94 percent electrical due to low main condenser vacuum caused by prolonged high intake temperature due to seasonal variations. The unit returned to full power operation on May 13, 2011.

On May 29, 2011, load was reduced to approximately 60 percent electrical for a control rod pattern adjustment. The unit returned to full power operation on the same day.

## 1. REACTOR SAFETY

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

1R01 Adverse Weather Protection (71111.01)

.1 External Flooding

a. Inspection Scope

The inspectors evaluated the design, material condition, and procedures for coping with the design basis probable maximum flood. The evaluation included a review to check for deviations from the descriptions provided in the Updated Final Safety Analysis Report (UFSAR) for features intended to mitigate the potential for flooding from external factors. As part of this evaluation, the inspectors checked for obstructions that could prevent draining, checked that the roofs did not contain obvious loose items that could clog drains in the event of heavy precipitation, and determined that barriers required to mitigate the flood were in place and operable. Additionally, the inspectors performed a walkdown of the protected area to identify any modification to the site which would inhibit site drainage during a probable maximum precipitation event or allow water ingress past a barrier. The inspectors also walked down underground bunkers/manholes subject to flooding that contained multiple train or multiple function risk-significant cables.
The inspectors also reviewed the abnormal operating procedure (AOP) for mitigating the design basis flood to ensure it could be implemented as written.

This inspection constituted one external flooding sample as defined in Inspection Procedure (IP) 71111.01-05.

b. Findings

Failure to Include Adequate Acceptance Criteria in a Surveillance Test

Introduction: The inspectors identified an unresolved item regarding the failure to include adequate acceptance criteria in a surveillance test for equipment that is the sole source of make-up water to the shell of the isolation condensers and the spent fuel pools for both units during a design basis flood.

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

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

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

If forecasted flood levels exceed 517 feet, a diesel-driven emergency flood pump will be connected by hoses to a fire system header in each unit. Through these fire system headers, the emergency flood pump will be capable of providing at least 175 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.” The licensee generated issue report (IR) 1209642, “NRC Identified URI with Flood Acceptance Criteria,” to address the inspectors concerns.

Upon further discussions with the licensee, the inspectors noticed that until early 2007, the flood pump was classified as an augmented quality piece of equipment. Therefore, the flood pump was within the scope of the licensee’s Quality Assurance Topical Report (QATR). Since 2007, the flood pump had been classified as non-safety-related. Based on the definition of safety-related systems, structures and components, as described in Title 10 of the Code of Federal Regulations, Part 50.2, “Definitions,” and based on the fact that that the flood pump is utilized to mitigate the consequences of an event described in Section 3.4.1.1, “External Flood Protection Measures,” of the Dresden UFSAR, the inspectors were concerned that the flood pump has been misclassified as non-safety and it should be classified as a safety-related piece of equipment.

At the end of the inspection period, the licensee was still working on identifying when and why the flood pump was downgraded from augmented quality to non-safety-related.

The licensee generated IR 1239579, “NRC Questions the Safety Classification of Diesel Flood Pump,” to address the inspector’s concerns. As part of this IR, the licensee generated an action to determine if the safety classification for the flood pump is appropriate based on Dresden’s design bases. The inspectors considered this issue an unresolved item (URI) pending review of the licensee’s evaluation (**URI 05000237/2011003-01; 05000249/2011003-01**, “Failure to Include Adequate Acceptance Criteria in a Surveillance Test”).

.2 Readiness of Offsite and Alternate AC Power Systems

a. Inspection Scope

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

- The coordination between the TSO and the plant during off-normal or emergency events;

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

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

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

Documents reviewed are listed in the Attachment to this report. The inspectors also reviewed corrective action program (CAP) items to verify that the licensee was identifying adverse weather issues at an appropriate threshold and entering them into their CAP in accordance with station corrective action procedures.

This inspection constituted one readiness of offsite and alternate AC power systems sample as defined in IP 71111.01-05.

b. Findings

No findings were identified.

1R04 Equipment Alignment (71111.04)

.1 Quarterly Partial System Walkdowns

a. Inspection Scope

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

- 2A standby liquid control (SBLC) train during planned maintenance of 2B SBLC pump;
- U3 high pressure coolant injection during U3 isolation condenser planned maintenance;
- 2B core spray during 2A core spray out-of-service for planned maintenance; and
- 3C low pressure coolant injection (LPCI) during 3D LPCI pump seal replacement.

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

These activities constituted 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 April 6, 2011, the inspectors performed a complete system alignment inspection of the emergency containment venting systems 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 line ups, 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 1.1.2.5.D, Unit 2 Standby Liquid Control Area, elevation 589';
- Fire Zone 1.1.1.5.D, Unit 3 Standby Liquid Control Area, elevation 589';
- Fire Zone 1.1.1.5.A, Unit 3 Isolation Condenser Area, elevation 589'; and
- Fire Zone 8.2.5.A, Unit 2 Reactor Feed Pumps, 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.

.2 Annual Fire Protection Drill Observation (71111.05A)

a. Inspection Scope

On May 14, 2011, the inspectors observed a fire brigade activation for a fire drill scenario on Unit 3 reactor building motor control center 38-4 cube A4. Based on this observation, the inspectors evaluated the readiness of the plant fire brigade to fight fires. The inspectors verified that the licensee staff identified deficiencies; openly discussed them in a self-critical manner at the drill debrief, and took appropriate corrective actions. Specific attributes evaluated were:

- proper wearing of turnout gear and self-contained breathing apparatus;
- proper use and layout of fire hoses;
- employment of appropriate fire fighting techniques;
- sufficient firefighting equipment brought to the scene;
- effectiveness of fire brigade leader communications, command, and control;
- search for victims and propagation of the fire into other plant areas;
- smoke removal operations;
- utilization of pre-planned strategies;
- adherence to the pre-planned drill scenario; and
- drill objectives.

Documents reviewed are listed in the Attachment to this report.

These activities constituted one annual fire protection inspection sample as defined in IP 71111.05-05.

b. Findings

No findings were identified.

1R06 Flooding (71111.06)

.1 Underground Vaults

a. Inspection Scope

The inspectors selected underground bunkers/manholes subject to flooding that contained cables whose failure could disable risk-significant equipment. The inspectors determined that the cables were not submerged, that splices were intact, and that appropriate cable support structures were in place. In those areas where dewatering devices were used, such as a sump pump, the device was operable and level alarm circuits were set appropriately to ensure that the cables would not be submerged. In those areas without dewatering devices, the inspectors verified that drainage of the area was available, or that the cables were qualified for submergence conditions. The inspectors also reviewed the licensee's corrective action documents with respect to past submerged cable issues identified in the corrective action program to verify the adequacy of the corrective actions. The inspectors performed a walkdown of the following underground bunkers/manholes subject to flooding:

- Manhole Security MH#1;
- Manhole Station Blackout MH#2; and
- Manhole Station Blackout MH#3.

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

b. Findings

No findings were identified.

1R07 Triennial Heat Sink Performance (71111.07T)

.1 Triennial Review of Heat Sink Performance

a. Inspection Scope

The inspectors reviewed operability determinations, completed surveillances, vendor manual information, associated calculations, performance test results and cooler inspection results associated with the 2A LPCI and the 2A emergency diesel generator (EDG). These heat exchangers/coolers were chosen based on their risk-significance in the licensee's probabilistic safety analysis, their important safety-related mitigating system support functions, their operating history, and their relatively low margin.

For the 2A LPCI and 2A EDG heat exchangers, the inspectors verified that testing, inspection, maintenance, and monitoring of biotic fouling and macrofouling programs were adequate to ensure proper heat transfer. This was accomplished by verifying: (1) the test method used was consistent with accepted industry practices, or equivalent; (2) the test conditions were consistent with the selected methodology; (3) the test acceptance criteria were consistent with the design basis values; and (4) results of heat exchanger performance testing. The inspectors also verified that the test results appropriately considered differences between testing conditions and design conditions, the frequency of testing based on trending of test results was sufficient to detect degradation prior to loss of heat removal capabilities below design basis values and test results considered test instrument inaccuracies and differences.

For the 2A EDG heat exchanger, the inspectors reviewed the methods and results of heat exchanger performance inspections. The inspectors verified the methods used to inspect and clean heat exchangers were consistent with industry standards, and the as-found results were recorded, evaluated, and appropriately dispositioned such that the as-left condition was acceptable.

In addition, the inspectors verified the condition and operation of the 2A LPCI and 2A EDG heat exchangers were consistent with design assumptions in heat transfer calculations and as described in the UFSAR. This included verification that the number of plugged tubes was within pre-established limits based on capacity and heat transfer assumptions. The inspectors verified the licensee evaluated the potential for water hammer and established adequate controls and operational limits to prevent heat exchanger degradation due to excessive flow-induced vibration during operation. In addition, eddy current test reports and visual inspection records were reviewed to determine the structural integrity of the heat exchanger.

The inspectors verified the performance of ultimate heat sink (UHS) and safety-related service water systems and their subcomponents such as piping, intake screens, pumps, valves, etc., by tests or other equivalent methods, to ensure availability and accessibility to the in-plant cooling water systems.

The inspectors verified that the licensee's inspection of the UHS was comprehensive and of significant depth to ensure sufficient reservoir capacity. This included review of the licensee's periodic monitoring and trending of sediment build-up and bay inspections. In addition, the inspectors reviewed the licensee's periodic performance monitoring of the UHS structural integrity and verified that adjacent non-seismic or non-safety-related structures cannot degrade or block safety-related flow paths, during a

severe weather or seismic event. The inspectors also performed walkdowns of accessible portions of the ultimate heat sink supply and return piping to look for possible settlement or movement and piping conditions that would indicate loss of structural integrity.

The inspectors reviewed the licensee's operation of the service water system and UHS. This included review of the licensee's procedures for a loss of the service water system or UHS and the verification that instrumentation, which is relied upon for decision making, was available and functional. In addition, the inspectors verified that macrofouling was adequately monitored, trended, and controlled by the licensee to prevent clogging. The inspectors verified that the licensee's biocide treatments for biotic control were adequately conducted and the results monitored, trended, and evaluated.

The inspectors also verified that the licensee ensured adequate isolation during design basis events, consistency between testing methodologies and design basis leakage rate assumptions, and proper performance of risk significant non-safety-related functions.

In addition, the inspectors reviewed condition reports related to the heat exchangers and heat sink performance issues to verify that the licensee had an appropriate threshold for identifying issues and to evaluate the effectiveness of the corrective actions. The documents that were reviewed are included in the Attachment to this report.

These inspection activities constituted three heat sink inspection samples as defined in IP 71111.07 05.

b. Findings

(1) Inadequate Instructions for the Inspection of Safety-Related Portions of the Intake Structure

Introduction: A finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," was identified by the inspectors for the licensee's failure to establish adequate instructions for inspecting bay 13 - the source of water for containment cooling service water (CCSW) pumps and the diesel fire pumps (DFPs) - and portions of the intake structure surrounding the diesel generator cooling water (DGCW) pumps. Specifically, the procedure that provides guidance for inspecting these structures lacked specific instructions on how to detect and record degradation by erosion and corrosion.

Description: On February 11, 2011, the inspectors identified the licensee failed to establish adequate instructions in surveillance procedures used to inspect bay 13, the source of water for the CCSW pumps and the DFP, and portions of the intake structure surrounding the DGCW pumps.

In response to GL 89-13, the licensee committed to inspect these safety-related structures to monitor, trend and evaluate any degradation resulting from erosion, corrosion, silt build-up and biological fouling.

However, the inspectors noted that procedure DTS 3900-07, "Crib House/Intake Structure Inspections" did not include specific guidance on how to detect, monitor or evaluate degradation resulting from erosion and corrosion. For example, the inspectors

identified the procedure did not provide instruction for what entails degradation of concrete structures or components by erosion, corrosion and biological fouling mechanisms. In addition, the procedure did not provide instructions on when or how to document these types of degradations. The inspectors were concerned that the lack of written instructions to identify and record the as-found condition of the intake structure could result in the licensee's inability to properly assess the effect of erosion, corrosion, and biological fouling on these structures and related components.

As a corrective action, the licensee initiated IR 1173579 to revise procedure DTS 3900-07 and provide specific instructions on how to accomplish the purpose of the procedure. These instructions include criteria for what to look for and how and what to document during these inspections related to component and structural degradation by erosion and corrosion.

Analysis: The inspectors determined that the licensee's failure to establish adequate instructions in surveillance procedures used to inspect bay 13 and portions of the intake structure surrounding the DGCW pumps was contrary to the requirements of 10 CFR Part 50, Appendix B, Criterion V, and was a performance deficiency. The performance deficiency was determined to be more than minor because, if left uncorrected, it would have the potential to lead to a more significant safety concern. Specifically, since the licensee's procedures did not contain instructions to properly inspect bay 13 and portions of the intake structure surrounding the DGCW pumps, the potential exists for an unacceptable degradation of these structures or related components to go undetected, affecting operability of these safety-related pumps.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," Table 4a for the Mitigating System Cornerstone. The finding screened as of very low safety significance (Green) because the finding was a qualification deficiency confirmed not to result in loss of operability or functionality. Specifically, the licensee performed a historical review of the surveillance reports and found the documented results acceptable. A qualitative assessment of the inspections established reasonable assurance that the results did not represent loss of operability. The inspectors did not have further concerns.

The inspectors determined the cause of this finding did not represent current licensee performance and no cross-cutting aspect was assigned.

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

Contrary to the above, as of February 11, 2011, the licensee did not establish surveillance procedures related to bay 13 and portions of the intake structure surrounding the DGCW pumps appropriate to the circumstances. Specifically, procedure DTS 3900-07, the procedure used to inspect these structures, lacked instructions on detecting or recording degradation by erosion and corrosion. Without these instructions, the effect of any degradation on system operability could not be evaluated. Because this violation was of very low safety significance and it was entered into the licensee's CAP as IR 1173579, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy

(**NCV 05000237/2011003-02; 05000249/2011003-02**, “Inadequate Instructions for the Inspection of Safety-Related Portions of the Intake Structure”).

(2) Inadequate Acceptance Criteria for Testing Equipment Relied Upon to Mitigate the Consequences of a Dam Failure

Introduction: A finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures and Drawings,” was identified by the inspectors for the licensee’s failure to establish adequate acceptance criteria for testing equipment relied upon to mitigate the consequences of a dam failure. Specifically, the acceptance criteria does not consider additional steps required to demonstrate the ability of the screen refuse pumps to deliver water to the containment cooling service water (CCSW)/fire pump enclosure (bay 13) to support operability of the isolation condensers.

Description: On February 9, 2011, the inspectors identified that the licensee failed to establish an adequate acceptance criteria for testing equipment relied upon to mitigate the consequences of a dam failure.

Section 9.2.5 of Dresden UFSAR titled “Ultimate Heat Sink,” describes the station’s ability to safely shutdown both units after a catastrophic failure of Dresden’s lock and dam. Following a dam failure, the level in the intake structure would drop drastically and both reactors would automatically shutdown on condenser low vacuum. Relief valves from the primary system to the suppression chamber would open to prevent over-pressurizing the reactor vessels. Each of the reactors would be depressurized at a controlled rate using the isolation condenser for each unit. During this event, the credited source of water for the isolation condensers is river water pumped by the diesel-driven fire pumps (DFP). The DFPs take their suction from bay 13, the fire pump enclosure. To operate the DFPs, water level needs to be restored and maintained at 505 feet by reflooding bay 13 due to the decrease in level as a result of failure of the dam. This evolution needs to take place within two hours to support make-up to the isolation condensers.

To accomplish filling of bay 13, foreign material exclusion (FME) screens (used to prevent debris from entering the bay) would need to be replaced by stop logs. Dewatering valves would need to be opened to permit river water to flow from the circulating water bay to the refuse pit. Electrical connections would need to take place to power the refuse pumps off the emergency diesel generators (EDG). The screen refuse pumps would need to be aligned and operated to supply water from the refuse pit to bay 13. Finally, water level in bay 13 would need to rise and be maintained at a level of 505 feet.

Section 9.2.5.4 of the UFSAR requires the licensee to perform a surveillance test every third refueling outage to verify that the required manual operations described above can be performed. The licensee developed procedure DOS 0010-01, “Dresden Dam Failure Equipment Test,” as the means to verify the equipment required to support primary containment cooling in the event of a dam failure is available and to demonstrate the ability of the screen refuse pumps to deliver water to the containment cooling service water (CCSW)/fire pump enclosure (bay 13). This procedure is considered a quality procedure, subject to 10 CFR Part 50, Appendix B requirements.

Section H.2 of DOS 0010-01, states the procedure's acceptance criteria is to complete Steps I.2 through I.14 in less than two hours. Steps I.2 through I.14 guide the operators on how to align valves to allow water from the circulating water bay to the refuse pump pit, align the refuse pumps for operation and run the refuse pumps to fill bay 13.

The inspectors identified four examples of inadequate acceptance criteria in this procedure.

1. When time is recorded per procedure in Step I.2, the system has been flushed, the needed tools are available and accessible, and the operators are already staged. The acceptance criterion of 2 hours is not adjusted for operators to assess the event, recognize the need to be in this procedure, or to arrive at the equipment location to perform the steps in the procedure. The licensee acknowledged the inspectors' concern and estimated that it would take about 15 minutes to complete these additional actions.

2. The procedure, DOS 0010-01, does not require installation of the stop logs and does not account for the time to perform this action in the acceptance criteria. The licensee stated that during the October 7, 2000, performance of DOS 0010-01, the FME screens were removed per procedure in order to install the stop logs. This evolution allowed foreign material to enter bay 13 and caused the two CCSW pumps to be inoperable. The licensee took adequate corrective actions by installing inner screens that do not need to be removed for the installation of the stop logs. However, the licensee also removed the steps that install the stop logs from the surveillance procedure and did not account for the timing of this evolution in the acceptance criteria. The inspectors verified that the stop logs are properly pre-staged and all required tools and equipment are present. The licensee acknowledged the inspectors concerns and based on previous validations, estimated it would take approximately one hour to remove the FME screens and install the stop logs.

3. The acceptance criteria in procedure DOS 0010-01, does not include the steps necessary to perform electrical connections to power the refuse pumps from the EDGs nor does it include a time estimate to perform these actions. As a result of the inspectors' concern, the licensee estimated it would take approximately 60 minutes to perform this evolution, which could be done concurrent with the rest of the procedure.

4. The procedure DOS 0010-01 does not account for the time needed for the water level in bay 13 to reach a level in which the DFP can safely operate (505 feet as measured in bay 13.) The time it takes for level to be restored to this point depends on the number of screen refuse pumps operating (one or two). As a result of the inspectors' inquiry, the licensee performed informal calculations and estimated that with one screen refuse pump running at the design flow rate of 2,400 gallons per minute (gpm), it would take 4.5 minutes and for two pumps it would take 2.25 minutes, to flood bay 13 to a level of 505 feet.

The inspectors concluded that the current acceptance criteria, defined as completing Steps I.2 through I.14 in less than two hours, was inadequate. This acceptance criteria did not account for additional steps, described above, needed to take place in order to

meet the intent of the procedure and demonstrate the ability of the screen refuse pumps to deliver water to bay 13 to support operability of the isolation condensers.

As a corrective action, the licensee initiated an IR 1221390, to revise procedure DOS 0010-01, and include the additional steps described above in the acceptance criteria.

Analysis: The inspectors determined that the licensee's failure to establish adequate acceptance criteria for testing equipment relied upon to mitigate the consequences of a dam failure was contrary to the requirements of 10 CFR Part 50, Appendix B, Criterion V, and was a performance deficiency. The performance deficiency was determined to be more than minor because it adversely affected the availability, reliability, and capability of mitigating systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, failure to ensure the screen refuse pumps would be capable of delivering water to bay 13 during a seismic event, assuming a loss of offsite power, affected the reliability of the isolation condensers to perform their safety function.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," Table 4a for the Mitigating System Cornerstone. The finding screened as of very low safety significance (Green) because the finding was a qualification deficiency confirmed not to result in a loss of operability or functionality. Specifically, the licensee estimated that the time it would take to perform the additional steps not included in the procedure's acceptance criteria was within the required two hours. As a result, the licensee provided reasonable assurance the isolation condensers did not experience a complete loss of function during this event.

The inspectors determined the cause of this finding did not represent current licensee performance and no cross-cutting aspect was assigned.

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

Contrary to the above, as of February 11, 2011, procedure DOS 0010-01, "Dresden Dam Failure Equipment Test," a quality procedure, was not appropriate for the circumstances. Specifically, the procedure's acceptance criteria did not account for all the necessary steps needed to perform the intent of the procedure of demonstrating the ability of the screen refuse pumps to deliver water to bay 13. Without adequate acceptance criteria, the 2-hour requirement to perform this procedure to support isolation condenser operability could not be evaluated. Because this violation was of very low safety significance and it was entered into the licensee's CAP as AR1221390, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy (**NCV 05000237/2011003-02; 05000249/2011003-02**, Inadequate Acceptance Criteria for Testing Equipment Relied Upon to Mitigate the Consequences of a Lock and Dam Failure).

(3) Inadequate Instructions for Coping with a Dresden Lock and Dam Failure

Introduction: A finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," was identified by the inspectors for the licensee's failure to establish adequate instructions for coping with a Dresden lock and dam failure. Specifically, the procedure that provides guidance for the safe shutdown of both units following a dam failure lacked controls for the configuration of the associated FME screens and lacked specific instructions on how to shed load off the EDGs and restore power to the bus associated with equipment relied upon during this event.

Description: On February 9, 2011, the inspectors identified that the licensee failed to establish adequate instructions for coping with a dam failure.

As described above in Section 1R07.b.(2), a seismic event is a credible event that would cause failure of the dam and a loss of offsite power (LOOP). In order to mitigate the effect of this event and to safely shutdown both reactors, bay 13 needs to be reflooded by aligning and operating the screen refuse pumps. Once the level in bay 13 reaches and is maintained at 505 feet, the DFPs can operate and provide makeup to the isolation condensers. This evolution needs to take place within two hours in order to support isolation condenser operability.

Procedure DOA 0010-01, "Dresden Lock and Dam Failure," describes the steps needed to mitigate the consequences of a dam failure and safely shutdown both reactors. Steps in this procedure need to be completed within two hours in order to provide make-up to the isolation condenser and support its operability.

The inspectors identified two examples of inadequate instructions in procedure DOA 0010-01 as follows:

1. As described above in Section 1R07.b.(2), the licensee installed additional FME screens (inner) in bay 13 that do not need to be removed for the installation of the stop logs. As a result of adding these inner screens, the licensee updated procedure DOA 0010-01, and added two additional steps to reflect the new configuration. The first step added, D.8.h(4), states "IF the downstream (inner) screens are installed, THEN continue at Step D.8.h.(7) to install stop logs in upstream (outer) track." The second step added, D.8.h(5), states "IF the upstream (outer) screens are installed, THEN install downstream (inner) screens" and explains how to perform this evolution. Step D.8.h(5) allowed for a configuration where the outer screens could be installed and not the inner screens. In this specific case, the operators would have to install the inner screens, remove the outer screens, and put the stop logs in place. The inspectors were concerned this new step allowed a configuration which was not time-validated. After inquired upon, the licensee estimated it would take an additional 20 minutes to install the inner screens.

2. The inspectors noted procedure DOA 0010-01 lacked specific instructions on how to shed loads off the EDGs and restore power to the bus associated with equipment relied upon during this event. In addition, it did not provide information that would aid the operators on how many kilowatts (kw) needed to be shed off the EDGs in order to run the screen refuse pumps. The inspectors were concerned that operators performing these tasks during this event would not have adequate guidance to

properly execute this electrical switching. This could potentially lead to overloading the EDGs or not being able to perform the procedure within the required time.

In conclusion, the inspectors had reasonable doubt that procedure DOA 0100-01 could perform its function within the required timeframe.

As a corrective action, the licensee initiated IR 1221386 and IR 1202741, to revise procedure DOA 0010-01, to provide adequate controls on the use of FME screens and to provide further guidance on restoring power to the screen refuse pumps.

Analysis: The inspectors determined that the licensee's failure to establish adequate instructions for coping with a Dresden lock and dam failure was contrary to the requirements of 10 CFR Part 50, Appendix B, Criterion V, and was a performance deficiency. The performance deficiency was determined to be more than minor because it adversely affected the availability, reliability, and capability of mitigating systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, procedure DOA 0010-01, lacked controls to ensure an adequate FME screen configuration and specific instructions on EDG loading and restoration of power to the bus associated with the screen refuse pumps. Without adequate controls and instructions, the licensee may not be able to perform the procedure within the two hour requirement to support isolation condensers operability during a seismic event assuming a loss of offsite power.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," Table 4a for the Mitigating System Cornerstone. The finding screened as of very low safety significance (Green) because the finding was a qualification deficiency confirmed not to result in a loss of operability or functionality. Specifically, the licensee estimated that the additional time required to install the inner screens and restore power to the screen refuse pumps was within the two hours required. As a result, the licensee provided reasonable assurance the isolation condenser did not experience a complete loss of function during this event.

The inspectors determined the cause of this finding did not represent current licensee performance and no cross-cutting aspect was assigned.

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

Contrary to the above, as of February 11, 2011, procedure DOA 0010-01, "Dresden Lock and Dam Failure," the procedure used to mitigate the consequences of a dam failure and safely shutdown both reactors, was not appropriate for the circumstances. Specifically, the procedure lacked controls to ensure proper configuration of the FME screens and lacked specific instructions on how to shed load off the EDGs and restore power to the bus associated with equipment relied upon during this event. Because this violation was of very low safety significance and it was entered into the licensee's CAPs as IR 1221386 and IR 1202741, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy (**NCV 05000237/2011003-04; 05000249/2011003-04**, "Inadequate Instructions for Coping with a Dresden Lock and Dam Failure").

(4) Inadequate Scoping of a Non-Safety-Related Pump Into Maintenance Rule

Introduction: A finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50.65(b)(2)(ii), “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” was identified by the inspectors for the licensee’s failure to adequately scope a non-safety-related component relied upon to mitigate an accident. Specifically, the licensee failed to include the non-safety-related screen refuse pumps, credited during a dam failure as a support component for the safety-related isolation condensers, as part of their maintenance effectiveness monitoring program.

Description: On February 9, 2011, the inspectors identified that the licensee failed to scope the screen refuse pumps into their maintenance rule program.

Section 9.2.5 of Dresden UFSAR, titled “Ultimate Heat Sink,” describes the station’s ability to safely shutdown both units after a failure of Dresden’s lock and dam. Following a dam failure, the level in the intake structure would drop drastically and both reactors would automatically shutdown on condenser low vacuum. Relief valves from the primary system to the suppression chamber would open to prevent overpressurizing the reactor vessels. Each of the reactors would be depressurized at a controlled rate using the isolation condenser for each unit. During this event, the credited source of water for the isolation condensers is river water pumped by the diesel-driven fire pumps (DFP). The DFPs take their suction from bay 13, the fire pump enclosure. To operate the DFPs, water level needs to be restored by reflooding bay 13 due to the decrease in level as a result of the failure of the dam. In order to reflood bay 13, the non-safety-related screen refuse pumps need to operate. The screen refuse pumps take suction at a lower level in the intake structure and discharge into bay 13. Once flooded, water level in bay 13 is maintained using the screen refuse pumps.

The inspectors identified the licensee did not scope the screen refuse pumps into their Maintenance Rule program. As described in UFSAR Section 9.2.5.3.1, “Dam Failure during Normal Plant Operation,” a failure of these refuse pumps could prevent the isolation condensers, safety-related systems, from performing their safety-related function of decay heat removal. In addition, the Safety Evaluation Report (SER) issued on December 20, 2001, by the Office of Nuclear Reactor Regulation (NRR) related to power up-rate Amendments No. 191 and No. 185, reiterates the reliance on the screen refuse pumps to maintain operability of the isolation condensers by providing a suction source for the Unit 2 and 3 diesel fire pump water. In addition, the SER recognizes bay 13 also provides suction for the containment cooling service water pumps.

The inspectors discussed these concerns with NRR. NRR acknowledged the issues and concurred on this inspection report.

As a corrective action, the licensee initiated IR 1221421, to evaluate the need to include the refuse screen pumps into their maintenance rule program.

Analysis: The inspectors determined that the licensee’s failure to scope the screen refuse pumps into the Maintenance Rule program was contrary to the requirements of 10 CFR Part 50.65(b)(2)(ii), and was a performance deficiency. The performance deficiency was determined to be more than minor because it adversely affected the availability, reliability, and capability of mitigating systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Failure to monitor the

performance or condition of these pumps in a manner sufficient to provide reasonable assurance the pumps were capable of fulfilling the intended functions affected the reliability of the isolation condensers to perform their safety function during a dam failure.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," Table 4a for the Mitigating System Cornerstone. The finding screened as of very low safety significance (Green) because the finding was a qualification deficiency confirmed not to result in a loss of operability or functionality. Specifically, the screen refuse pumps are in a four year overhaul and motor replacement program. In addition, the licensee performed further review of related maintenance and provided reasonable assurance the screen refuse pumps had not experienced a complete loss of function. The inspectors did not have further concerns.

The inspectors determined the cause of this finding did not represent current licensee performance and no cross-cutting aspect was assigned.

<u>Enforcement</u>: Title 10 CFR Part 50.65(b)(2)(ii), "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," requires, in part, non-safety-related structures, systems and components whose failure could prevent safety-related structures, systems and components from fulfilling their safety-related functions shall be included in a maintenance monitoring program.

Contrary to the above, as of February 11, 2011, the licensee failed to include a non-safety-related component whose failure could prevent safety-related structures, systems, and components from fulfilling their safety-related functions in a maintenance monitoring program. Specifically, the inspectors identified the screen refuse pumps, non-safety-related pumps which provide a support function for the safety-related isolation condensers during a dam failure as described in Section 9.2.5 of the UFSAR and the SER for power up-rate, were not included in the maintenance monitoring program. Because this violation was of very low safety significance and it was entered into the licensee's CAP as IR 1221421, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy (**NCV05000237/2011003-05; 05000249/2011003-05**, Inadequate Scoping of a Non-Safety-related Pump Into Maintenance Rule).

(5) <u>Concerns with the Licensing and Design Basis</u>

<u>Introduction</u>: An unresolved item was identified by the inspectors for potential concerns with the licensing and design basis related to a seismic event that causes a dam failure and a loss of offsite power (LOOP).

<u>Description</u>: A seismic event that causes a dam failure is a credible event. In a letter to the NRC dated March 31, 1998, (ML073540225), the licensee stated: "In a Stability Investigation that was performed by the U.S. Army Corps of Engineers in 1973, the Dresden Lock and Dam was evaluated for earthquakes with a static equivalent horizontal acceleration of 0.05g for the dam and 0.10g for the lock and found to be acceptable. The safe shutdown earthquake (SSE) defined for Dresden Station has a magnitude of 0.2g zero period acceleration (ZPA) horizontal ground acceleration."
As described in previous sections of this report, during this scenario, the isolation

condensers are the only credited systems available to remove decay heat from both reactors.

The inspectors are concerned with the following:

- the current licensing bases which relies on non-safety-related components (screen refuse pumps) to mitigate the effects of a credible postulated UFSAR accident;
- the reliance of one success path to mitigate the consequences of a seismic event which results in a dam failure and a loss of offsite power. Specifically, during this event, the containment cooling service water (CCSW) pumps' piping drains to the intake structure due to lower intake water level. Therefore, the CCSW pumps which are designed for containment heat removal and as a support system for the low pressure coolant injection (LPCI) in cold shutdown mode and high pressure coolant injection (HPCI) by means of room cooling, are inoperable. The licensee does not credit the CCSW pumps during this event. The CCSW pumps take suction from bay 13, the same enclosure as the DFPs. If the licensee required CCSW in service, trapped air would need to be rapidly vented from the CCSW piping by cutting into the top of the pipe. This evolution would be performed by two individuals in the Maintenance Department who may not be onsite at the time of the event and may take up to one hour to arrive at the site. In addition, the licensee has not evaluated the capability of the screen refuse pumps to provide enough water to meet the demands of the CCSW pumps and the DFPs; and
- a seismically qualified method to supply water to the isolation condenser would not be available for about 5 hours. This path entails connecting the discharge of the Unit 2 diesel generator cooling water (DGCW) pump, which is safety-related and draws water at a lower elevation of the intake structure, to the isolation condenser make-up pumps. The Unit 2 DGCW pump would provide make-up water to both Unit 2 and Unit 3 isolation condensers. With no makeup water, the water level in the isolation condenser shell approaches the bottom of the tube bundles in 20 minutes which was viewed as a reasonable length of time allowed for initiating makeup water flow to the shell side of the isolation condenser to minimize water losses from the core. Although the licensee evaluated the potential 5-hour delay and considerably more time is available before the reactor core begins to uncover, the inspectors questioned whether 5 hours to restore water was reasonable.

This issue is unresolved pending further clarification of Dresden's licensing and design basis during a seismic event (**URI 05000237/2011003-06; 05000249/2011003-06**, "Concerns with the Licensing and Design Basis").

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope

On May 4, 2011, and again on June 27, 2011, the inspectors observed two crews of licensed operators in the plant's simulator during licensed operator requalification examinations to verify that operator performance was adequate, evaluators were identifying and documenting crew performance problems, and training was being

conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

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

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

This inspection constituted two quarterly licensed operator requalification program samples as defined in IP 71111.11.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

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

- Unit 3 core spray; and
- 480 volt breakers.

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

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

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

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

b. Findings

No findings were identified.

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

a. Inspection Scope

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

- Unit 3 on line risk (OLR) Yellow during lube of Unit 3 turbine building closed cooling water (TBCCW) pump motor;
- Unit 2 OLR Yellow during 2B standby liquid control planned maintenance;
- Unit 2 emergency diesel generator cooling water pump inoperable; and
- Unit 2 B TBCCW heat exchanger out-of-service.

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

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

b. Findings

No findings were identified.

### 1R15 Operability Evaluations and Functional Assessments (71111.15)

a. Inspection Scope

The inspectors reviewed the following issues:

- IR 1161361, "Historical Operability of MS [main steam] Lines due to Failed Snubbers";
- IR 1094902, "Apparent Valve Seat Leakage 3 DGCW [diesel generating cooling water] Flow Directing Valves";
- IR 1017020, "MSLB [main steam line break] Calc 3C2-0181-001 Did Not Consider Opening to Reactor Bldg";
- IR 1207712, "Crack Discovered in the 3 EDG [emergency diesel generator] Inlet Air Turning Box"; and
- IR 1208503, "3-1402-8B Didn't Meet Acceptance Criteria During DOS 1400-09."

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)

a. Inspection Scope

The inspectors reviewed the following modification(s):

- Engineering Change 362432, "3-1501-19B Install 4 Rotor Limit Switch, Add TS Open Protection and Change Close Scheme to LS Control."

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 permanent plant modification sample as defined in IP 71111.18 05.

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

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

- WO 1026062-01, "D2 4Y PM SBO [station blackout] D/G [diesel generator] Overhaul AMOT Valve 2-6629-24A";
- WO 1293696-02, "OP PMT [post maintenance testing] Functional Test 2-1401-2A CS [core spray] Pump"; and
- WO 826064, "D3 12Y PM [preventative maintenance] Replace 'D' LPCI [low pressure coolant injection] Pump Mechanical Seal."

These activities were selected based upon the structure, system, or component's ability to impact risk. The inspectors evaluated these activities for the following (as applicable): the effect of testing on the plant had been adequately addressed; testing was adequate for the maintenance performed; acceptance criteria were clear and demonstrated operational readiness; test instrumentation was appropriate; tests were performed as written in accordance with properly reviewed and approved procedures; equipment was returned to its operational status following testing (temporary modifications or jumpers required for test performance were properly removed after test completion); and test documentation was properly evaluated. The inspectors evaluated the activities against 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 three post-maintenance testing samples as defined in IP 71111.19-05.

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

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

- WO 1399468, "D2 Qtr TS 2A SBLC [standby liquid control] Pump Test for In-Service Testing Surv" (IST);
- DOP 2000-180, "Drywell Sump Operations with Unit On-line," Rev 00 (RCS);
- WO 1374038, "D2 SA PM Emergency Diesel Pump (CCSW Pump) Operation";
- WO 1308656-01, "Op D2/3 Annual PM Control Room HVAC System Smoke Detector Test"; and
- WO 1229061, "D2 24M TS D/G Test/Endur & Margin/Full Load REJ/ECCS."

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

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

- equipment was returned to a position or status required to support the performance of its safety functions; and
- all problems identified during the testing were appropriately documented and dispositioned in the CAP.

Documents reviewed are listed in the Attachment to this report.

This inspection constituted three routine surveillance testing samples, one inservice testing sample, and one reactor coolant system leak detection inspection sample, as defined in IP 71111.22, Sections -02 and -05.

b. Findings

No findings were identified.

1EP6 Drill Evaluation (71114.06)

.1 Training Observation

a. Inspection Scope

The inspector observed a simulator training evolution for licensed operators on May 4, 2011, which required emergency plan implementation by a licensee operations crew. This evolution was planned to be evaluated and included in performance indicator data regarding drill and exercise performance. The inspectors observed event classification and notification activities performed by the crew. The inspectors also attended the post-evolution critique for the scenario. The focus of the inspectors' activities was to note any weaknesses and deficiencies in the crew's performance and ensure that the licensee evaluators noted the same issues and entered them into the corrective action program. As part of the inspection, the inspectors reviewed the scenario package and other documents listed in the Attachment to this report.

This inspection of the licensee's training evolution with emergency preparedness drill aspects constituted one sample as defined in IP 71114.06-05.

b. Findings

No findings were identified.

## 2. RADIATION SAFETY

**Cornerstones: Occupational and Public 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 Inspection Planning (02.01)

a. Inspection Scope

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

b. Findings

No findings were identified.

.2 Radiological Hazard Assessment (02.02)

a. Inspection Scope

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

The inspectors reviewed the last two radiological surveys from selected plant areas and evaluated whether the thoroughness and frequency of the surveys were appropriate for the given radiological hazard.

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 assess conditions.

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

b. Findings

No findings were identified.

.3 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."

The inspectors reviewed the following radiation work permits used to access high radiation areas and evaluated the specified work control instructions or control barriers.

- Trouble shooting off-gas system; and
- Dresden-2 TIP room maintenance.

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

b. Findings

No findings were identified.

.4 Contamination and Radioactive Material Control (02.04)

a. Inspection Scope

The inspectors reviewed the licensee's procedures and assessed 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.

The inspectors selected several sealed sources from the licensee's inventory records and assessed whether the sources were accounted for and verified to be intact.

The inspectors evaluated whether any transactions, since the last inspection, involving nationally tracked sources were reported in accordance with 10 CFR 20.2207.

b. Findings

No findings were identified.

.5 Radiological Hazards Control and Work Coverage (02.05)

a. Inspection Scope

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

The inspectors assessed whether radiation monitoring devices were placed on the individual's body consistent with licensee procedures. The inspectors determined whether the dosimeter was placed in the location of highest expected dose or that the

licensee properly employed an NRC-approved method of determining effective dose equivalent.

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

The inspectors 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 assess conformance with the occupational performance indicator.

b. Findings

No findings were identified.

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

a. Inspection Scope

The inspectors discussed with the radiation protection manager 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 reduced 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.

.7 Radiation Worker Performance (02.07)

a. Inspection Scope

The inspectors reviewed radiological problem reports since the last inspection that found the cause of the event to be human performance errors. The inspectors evaluated whether there was an observable pattern traceable to a similar cause. The inspectors assessed whether this perspective matched the corrective action approach taken by the

licensee to resolve the reported problems. The inspectors discussed with the radiation protection manager any problems with the corrective actions planned or taken.

b. Findings

No findings were identified.

.8 Problem Identification and Resolution (02.09)

a. Inspection Scope

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

b. Findings

No findings were identified.

2RS3 In-Plant Airborne Radioactivity Control and Mitigation (71124.03)

This inspection constituted one complete sample as defined in IP 71124.03-05.

.1 Inspection Planning (02.01)

a. Inspection Scope

The inspectors reviewed the plant UFSAR to identify areas of the plant designed as potential airborne radiation areas and any associated ventilation systems or airborne monitoring instrumentation. Instrumentation review included continuous air monitors (continuous air monitors and particulate-iodine-noble-gas-type instruments) used to identify changing airborne radiological conditions such that actions to prevent an overexposure may be taken. The review included an overview of the respiratory protection program and a description of the types of devices used. The inspectors reviewed UFSAR, TS, and emergency planning documents to identify location and quantity of respiratory protection devices stored for emergency use.

Inspectors reviewed the licensee's procedures for maintenance, inspection, and use of respiratory protection equipment including self-contained breathing apparatus as well as procedures for air quality maintenance.

The inspectors reviewed reported performance indicators to identify any related to unintended dose resulting from intakes of radioactive material.

b. Findings

No findings were identified.

.2 Engineering Controls (02.02)

a. Inspection Scope

The inspectors reviewed the licensee's use of permanent and temporary ventilation to determine whether the licensee uses ventilation systems as part of its engineering controls (in lieu of respiratory protection devices) to control airborne radioactivity. The inspectors reviewed procedural guidance for use of installed plant systems, such as containment purge, spent fuel pool ventilation, and auxiliary building ventilation, and assessed whether the systems are used, to the extent practicable, during high-risk activities (e.g., using containment purge during cavity flood-up).

The inspectors selected installed ventilation systems used to mitigate the potential for airborne radioactivity, and evaluated whether the ventilation airflow capacity, flow path (including the alignment of the suction and discharges), and filter/charcoal unit efficiencies, as appropriate, were consistent with maintaining concentrations of airborne radioactivity in work areas below the concentrations of an airborne area, to the extent practicable.

The inspectors selected temporary ventilation system setups (high-efficiency particulate air/charcoal negative pressure units, down draft tables, tents, metal "Kelly buildings," and other enclosures) used to support work in contaminated areas. The inspectors assessed whether the use of these systems is consistent with licensee procedural guidance and as-low-as-is-reasonably-achievable concept.

The inspectors reviewed airborne monitoring protocols by selecting installed systems used to monitor and warn of changing airborne concentrations in the plant and evaluating whether the alarms and setpoints are sufficient to prompt licensee/worker action to ensure that doses are maintained within the limits of 10 CFR Part 20 and the as-low-as-is-reasonably-achievable concept.

The inspectors assessed whether the licensee had established trigger points (e.g., the Electric Power Research Institute's "Alpha Monitoring Guidelines for Operating Nuclear Power Stations") for evaluating levels of airborne beta-emitting (e.g., plutonium-241) and alpha-emitting radionuclides.

b. Findings

No findings were identified.

.3 Use of Respiratory Protection Devices (02.03)

a. Inspection Scope

For those situations where it is impractical to employ engineering controls to minimize airborne radioactivity, the inspectors assessed whether the licensee provided respiratory protective devices such that occupational doses are as-low-as-is-reasonably-achievable. The inspectors selected work activities where respiratory protection devices were used to limit the intake of radioactive materials, and assessed whether the licensee performed an evaluation concluding that further engineering controls were not practical and that the use of respirators is as-low-as-is-reasonably-achievable. The inspectors also evaluated whether the licensee had established means (such as routine bioassay) to determine if

the level of protection (protection factor) provided by the respiratory protection devices during use was at least as good as that assumed in the licensee's work controls and dose assessment.

The inspectors assessed whether respiratory protection devices used to limit the intake of radioactive materials were certified by the National Institute for Occupational Safety and Health/Mine Safety and Health Administration or have been approved by the NRC per 10 CFR 20.1703(b). The inspectors selected work activities where respiratory protection devices were used. The inspectors evaluated whether the devices were used consistent with their National Institute for Occupational Safety and Health/Mine Safety and Health Administration certification or any conditions of NRC approval.

The inspectors reviewed records of air testing for supplied-air devices and self-contained breathing apparatus bottles to assess whether the air used in these devices meets or exceeds Grade D quality. The inspectors reviewed plant breathing air supply systems to determine whether they meet the minimum pressure and airflow requirements for the devices in use.

The inspectors selected several individuals qualified to use respiratory protection devices, and assessed whether they have been deemed fit to use the devices by a physician.

The inspectors selected several individuals assigned to wear a respiratory protection device and observed them donning, doffing, and functionally checking the device as appropriate. Through interviews with these individuals, the inspectors evaluated whether they knew how to safely use the device and how to properly respond to any device malfunction or unusual occurrence (loss of power, loss of air, etc.) and requested a demonstration of device use (donning, doffing, functional checks, and device malfunction) from selected operation and radiation protection staff.

The inspectors chose multiple respiratory protection devices staged and ready for use in the plant or stocked for issuance for use. The inspectors assessed the physical condition of the device components (mask or hood, harnesses, air lines, regulators, air bottles, etc.) and reviewed records of routine inspection for each. The inspectors selected several of the devices and reviewed records of maintenance on the vital components (e.g., pressure regulators, inhalation/exhalation valves, hose couplings).

b. Findings

No findings were identified.

.4 Self-Contained Breathing Apparatus for Emergency Use (02.04)

a. Inspection Scope

Based on the UFSAR, TS, and emergency operating procedure requirements, the inspectors reviewed the status and surveillance records of self-contained breathing apparatuses staged in-plant for use during emergencies. The inspectors reviewed the licensee's capability for refilling and transporting self-contained breathing apparatus air bottles to and from the control room and operations support center during emergency conditions.

The inspectors selected several individuals on control room shift crews and from designated departments currently assigned emergency duties (e.g., onsite search and rescue duties) to assess whether control room operators and other emergency response and radiation protection personnel (assigned in-plant search and rescue duties or as required by emergency operating procedures or the emergency plan) were trained and qualified in the use of self-contained breathing apparatuses (including personal bottle change-out). The inspectors evaluated whether personnel assigned to refill bottles were trained and qualified for that task.

The inspectors determined whether appropriate mask sizes and types are available for use (i.e., in-field mask size and type match what was used in fit-testing). The inspectors determined whether on-shift operators had no facial hair that would interfere with the sealing of the mask to the face and whether vision correction (e.g., glasses inserts or corrected lenses) was available as appropriate.

The inspectors reviewed the past two years of maintenance records for select self-contained breathing apparatus units used to support operator activities during accident conditions and designated as "ready for service" to assess whether any maintenance or repairs on any self-contained breathing apparatus unit's vital components were performed by an individual, or individuals, certified by the manufacturer of the device to perform the work. The vital components typically are the pressure-demand air regulator and the low-pressure alarm. The inspectors reviewed the onsite maintenance procedures governing vital component work to determine any inconsistencies with the self-contained breathing apparatus manufacturer's recommended practices. For those self-contained breathing apparatuses designated as "ready for service," the inspectors determined whether the required, periodic air cylinder hydrostatic testing was documented and up-to-date, and the retest air cylinder markings required by the U.S. Department of Transportation were in place.

b. Findings

No findings were identified.

.5 Problem Identification and Resolution (02.05)

a. Inspection Scope

The inspectors evaluated whether problems associated with the control and mitigation of in-plant airborne radioactivity were being identified by the licensee at an appropriate threshold and were properly addressed for resolution in the licensee corrective action program. The inspectors assessed whether the corrective actions were appropriate for a selected sample of problems involving airborne radioactivity and were appropriately documented by the licensee.

b. Findings

No findings were identified.

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

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

.1 Inspection Planning (02.01)

a. Inspection Scope

The inspectors reviewed the solid radioactive waste system description in the UFSAR, the process control program, and the recent radiological effluent release report for information on the types, amounts, and processing of radioactive waste disposal.

The inspectors reviewed the scope of any quality assurance audits in this area since the last inspection to gain insights into the licensee's performance and performed the "smart sampling" inspection planning.

b. Findings

No findings were identified.

.2 Radioactive Material Storage (02.02)

a. Inspection Scope

The inspectors selected areas where containers of radioactive waste are stored, and evaluated whether the containers were labeled in accordance with 10 CFR 20.1904, "Labeling Containers," or controlled in accordance with 10 CFR 20.1905, "Exemptions to Labeling Requirements," as appropriate.

The inspectors assessed whether the radioactive material storage areas were controlled and posted in accordance with the requirements of 10 CFR Part 20, "Standards for Protection against Radiation." For materials stored or used in the controlled or unrestricted areas, the inspectors evaluated whether they were secured against unauthorized removal and controlled in accordance with 10 CFR 20.1801, "Security of Stored Material," and 10 CFR 20.1802, "Control of Material Not in Storage," as appropriate.

The inspectors evaluated whether the licensee established a process for monitoring the impact of long term storage (e.g., buildup of any gases produced by waste decomposition, chemical reactions, container deformation, loss of container integrity, or re-release of free-flowing water) that was sufficient to identify potential unmonitored, unplanned releases or nonconformance with waste disposal requirements.

The inspectors selected containers of stored radioactive material, and assessed each container for signs of swelling, leakage, and deformation.

b. Findings

No findings were identified.

.3 Radioactive Waste System Walkdown (02.03)

a. Inspection Scope

The inspectors walked down accessible portions of select radioactive waste processing systems to assess whether the current system configuration and operation agreed with

the descriptions in the UFSAR, Offsite Dose Calculation Manual, and process control program.

The inspectors reviewed administrative and/or physical controls (i.e., drainage and isolation of the system from other systems) to assess whether the equipment which is not in service or abandoned in place would not contribute to an unmonitored release path and/or affect operating systems or be a source of unnecessary personnel exposure. The inspectors assessed whether the licensee reviewed the safety significance of systems and equipment abandoned in place in accordance with 10 CFR 50.59, "Changes, Tests, and Experiments."

The inspectors reviewed the adequacy of changes made to the radioactive waste processing systems since the last inspection. The inspectors evaluated whether changes from what is described in the UFSAR were reviewed and documented in accordance with 10 CFR 50.59, as appropriate and to assess the impact on radiation doses to members of the public.

The inspectors selected processes for transferring radioactive waste resin and/or sludge discharges into shipping/disposal containers and assessed whether the waste stream mixing, sampling procedures, and methodology for waste concentration averaging were consistent with the process control program, and provided representative samples of the waste product for the purposes of waste classification as described in 10 CFR 61.55, "Waste Classification."

For those systems that provide tank recirculation, the inspectors evaluated whether the tank recirculation procedures provided sufficient mixing.

The inspectors assessed whether the licensee's process control program correctly described the current methods and procedures for dewatering and waste stabilization (e.g., removal of freestanding liquid).

b. Findings

No findings were identified.

.4 Waste Characterization and Classification (02.04)

a. Inspection Scope

The inspectors selected the following radioactive waste streams for review:

- Dry Active Waste Stream;
- Waste Sludge/Spent Resin Stream;
- Unit-2 Torus Filter Stream;
- Spent Fuel Pool Crud Stream; and
- Condensate Waste Resin Stream.

For the waste streams listed above, the inspectors assessed whether the licensee's radiochemical sample analysis results (i.e., "10 CFR Part 61" analysis) were sufficient to support radioactive waste characterization as required by 10 CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste." The inspectors evaluated whether the licensee's use of scaling factors and calculations to account for

difficult-to-measure radionuclides was technically sound and based on current 10 CFR Part 61 analysis for the selected radioactive waste streams.

The inspectors evaluated whether changes to plant operational parameters were taken into account to: (1) maintain the validity of the waste stream composition data between the annual or biennial sample analysis update; and (2) assure that waste shipments continued to meet the requirements of 10 CFR Part 61 for the waste streams selected above.

The inspectors evaluated whether the licensee had established and maintained an adequate quality assurance program to ensure compliance with the waste classification and characterization requirements of 10 CFR 61.55 and 10 CFR 61.56, "Waste Characteristics."

b. Findings

No findings were identified.

.5 Shipment Preparation (02.05)

a. Inspection Scope

The inspectors observed shipment packaging, surveying, labeling, marking, placarding, vehicle checks, emergency instructions, disposal manifest, shipping papers provided to the driver, and licensee verification of shipment readiness. The inspectors assessed whether the requirements of applicable transport cask certificate of compliance had been met. The inspectors evaluated whether the receiving licensee was authorized to receive the shipment packages. The inspectors evaluated whether the licensee's procedures for cask loading and closure procedures were consistent with the vendor's current approved procedures.

The inspectors observed radiation workers during the conduct of radioactive waste processing and radioactive material shipment preparation and receipt activities. The inspectors assessed whether the shippers were knowledgeable of the shipping regulations and whether shipping personnel demonstrated adequate skills to accomplish the package preparation requirements for public transport with respect to:

- The licensee's response to NRC Bulletin 79-19, "Packaging of Low-Level Radioactive Waste for Transport and Burial," dated August 10, 1979; and
- Title 49 CFR Part 172, "Hazardous Materials Table, Special Provisions, Hazardous Materials Communication, Emergency Response Information, Training Requirements, and Security Plans," Subpart H, "Training."

Due to limited opportunities for direct observation, the inspectors reviewed the technical instructions presented to workers during routine training. The inspectors assessed whether the licensee's training program provided training to personnel responsible for the conduct of radioactive waste processing and radioactive material shipment preparation activities.

b. Findings

No findings were identified.

.6 Shipping Records (02.06)

a. Inspection Scope

The inspectors evaluated whether the shipping documents indicated the proper shipper name; emergency response information and a 24-hour contact telephone number; accurate curie content and volume of material; and appropriate waste classification, transport index, and UN number for the following radioactive shipments:

- DW-11-012; Radioactive Material; Low Specific Activity (LSA-II); 7; UN3321; DAW;
- DW-11-020; Radioactive Material; Low Specific Activity (LSA-II); 7; UN3321; Laundry;
- DW-11-008; Radioactive Material; Low Specific Activity (LSA-II); 7; UN3321;
- DW-11-011; Radioactive Material; Low Specific Activity (LSA-II); 7; UN3321: Metal Cask containing Condensate Resin; and
- DW-10-019; Radioactive Material, Type B(U) Package, 7, UN-2916; Fissile Excepted of Dewatered Filters to Clive Utah; dated April 16, 2010.

Additionally, the inspectors assessed whether the shipment placarding was consistent with the information in the shipping documentation.

b. Findings

No findings were identified.

.7 Identification and Resolution of Problems (02.07)

a. Inspection Scope

The inspectors assessed whether problems associated with radioactive waste processing, handling, storage, and transportation, were being identified by the licensee at an appropriate threshold, were properly characterized, and were properly addressed for resolution in the licensee corrective action program. Additionally, the inspectors evaluated whether the corrective actions were appropriate for a selected sample of problems documented by the licensee that involve radioactive waste processing, handling, storage, and transportation.

The inspectors reviewed results of selected audits performed since the last inspection of this program and evaluated the adequacy of the licensee's corrective actions for issues identified during those audits.

b. Findings

No findings were identified.

## 4. OTHER ACTIVITIES

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

4OA1 Performance Indicator Verification (71151)

.1 Mitigating Systems Performance Index - Emergency AC Power System

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index (MSPI) - Emergency AC Power System performance indicator for Unit 2 and Unit 3 for the period from the first quarter 2010 through the second quarter 2011. To determine the accuracy of the Performance Indicator (PI) data reported during those periods, PI definitions and guidance contained in the Nuclear Energy Institute (NEI) Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, dated October 2009, were used. The inspectors reviewed the licensee's operator narrative logs, MSPI derivation reports, issue reports, event reports and NRC Integrated Inspection Reports for the period of January 2010 through May 2011 to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings were identified.

.2 Mitigating Systems Performance Index - High Pressure Injection Systems

a. Inspection Scope

The inspectors sampled licensee submittals for the MSPI - High Pressure Injection Systems performance indicator for Units 2 and 3 for the period from the first quarter 2010 through the second quarter 2011. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, dated October 2009, were used. The inspectors reviewed the licensee's operator narrative logs, issue reports, MSPI derivation reports, event reports and NRC Integrated Inspection Reports for the period of January 2010 through the May 2011 to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or

transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Routine Review of Items Entered into the Corrective Action Program

a. Inspection Scope

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

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

b. Findings

No findings were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

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

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

b. Findings

No findings were identified.

.3 Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a review of the licensee's CAP and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review was focused on repetitive equipment issues, but also considered the results of daily inspector CAP item screening discussed in Section 4OA2.2 above, licensee trending efforts, and licensee human performance results. The inspectors' review nominally considered the 24-month period of January 1, 2009, through December 31, 2010, although some examples expanded beyond those dates where the scope of the trend warranted.

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

The inspectors focused their review on local leak rate testing (LLRT) and 480V breaker issues.

The inspectors identified one trend:

Main steam isolation valve (MSIV) failures to meet LLRT allowable limits.

- IR 1133829, "LLRT on 3-0203-1C Leakage Exceeded 34 SCFH";
- IR 1133832, "LLRT 3-0203-1D Leakage Exceeded 34 SCFH";
- IR 1133833, "LLRT 3-0203-2D Leakage Exceeded 34 SCFH Limit."

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

b. Findings and Observations

The inspectors reviewed the equipment apparent cause evaluation (EACE) report generated by the licensee to address the issues described in IRs 1133829, 1133832, and 1133833. The licensee documented in the report that the apparent cause for the MSIVs failure to meet LLRT allowable limits was a non-optimal valve design that allows the main valve plug to become misaligned with the valve seat ring.

The design issue was originally identified as a long term chronic problem in 2004 (Refer to MREQ [Management Request] AT [Action Tracking] 171393-28, Chronic Problem #91 – MSIV Leak Rate Test Failures). To date, there has been no conclusion reached for an engineering recommended solution.

The inspectors reviewed the Unit 3 MSIV LLRT history for all Unit 3 MSIVs and noticed an increasing trend in leakage rate for the three MSIVs in question (3-0203-1C, 3-0203 1D and 3-0203-2D). Leakage rate for all three valves doubled and in some instances exceeded twice the leakage rate results from previous outages (D3R18 through D3R21).

The licensee decided during D3R20 (2008) and previous outages to not perform maintenance on the MSIVs that demonstrated little margin to failure during the test. The subsequent tests (2010) found the leakage rate above the technical specification limit.

The inspectors determined that the failure to meet the allowable leakage rate limits specified in the technical specifications was a licensee-identified violation and is documented in Section 4OA7 of this report.

.4 Selected Issue Follow-Up Inspection Associated with IR 1222832, "Unit 3 Emergency Diesel Generator Jacket Water Temperatures Trend"

a. Inspection Scope

The inspectors reviewed the corrective actions associated with an adverse trend in the temperature of the Unit 3 emergency diesel generator (EDG) jacket water system that was identified by the licensee and documented in IR 1222832. The inspectors chose this issue for an in-depth review due to the safety significance of the EDG, the potential for a common issue between all EDGs, and recent issues involving degradation of the EDG cooling water system. The inspectors reviewed the surveillance test results and the troubleshooting activities to verify that the licensee was appropriately addressing the adverse trend in their CAP. The inspectors also reviewed the operability evaluation to verify that the actions taken by the licensee were appropriate based on the operability of the system.

b. Findings and Observations

During the system engineer's review of a successful monthly surveillance run of the Unit 3 EDG on May 24, 2011, the engineer identified that the jacket water inlet and outlet temperatures observed during the run were both high outside the nominal range. The system engineer noted that this was a step increase compared to past surveillance runs on this EDG. Therefore, the system engineer documented the adverse trend in the licensee's CAP. Based on the successful surveillance test and the expectation that parameters will not exceed the trip setpoint, operations determined the U3 EDG to be operable with a more rigorous evaluation required. The licensee assigned the condition report a Significance Level 4 based on the issue not having any actual impact on nuclear safety and an Investigation Class D because no formal investigation into the issue was required in accordance with the licensee's CAP. The licensee then initiated an operability evaluation to further evaluate the operability of the EDG. The licensee's operability evaluation identified that while margin was reduced on the maximum EDG cooling water intake temperature, the EDG was still able to perform its required function at the design maximum intake temperature. The operability evaluation also determined that although the cause of the elevated jacket water temperature was not known, the other two EDGs (Unit 2 and Unit 2/3) did not possess the same condition based on the jacket water temperature data collected from their surveillance runs.

The inspectors determined that the licensee's documentation of the adverse jacket water temperature trend in the CAP was complete, accurate, and timely. The inspectors also determined that the licensee took appropriate action to evaluate the operability of the EDG, and appropriately considered extent of condition. The inspectors also determined that the classification and prioritization of the resolution of the issue was appropriate commensurate with its safety significance.

No findings were identified.

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

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

a. Inspection Scope

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

b. Findings and Observations

.1 Change In Unit 2 Reactor Water Level Due To A Failure To Follow Procedure

Introduction: A finding of very low safety significance and associated non-cited violation of Technical Specification 5.4.1 was self-revealed for a Nuclear Station Operator (NSO) failing to follow step G.14.a of procedure DOP 0600-06, "Feedwater Regulating Valve (FWRV) Operation," Revision 39.

Description: On May 22, 2011, a NSO was returning the 2A FWRV to service after maintenance and failed to place the valve in test before performing post-maintenance testing by cycling the valve. The initial action was for the NSO to open the 2A FWRV which was isolated. Because the 2A FWRV was not in test, the opening of the 2A FWRV caused the 2B FWRV to go closed. Since the 2A FWRV was isolated closing of the 2B FWRV caused reactor water level to go down. The NSO took manual control of the 2B FWRV and restored water level to normal. Reactor water level decreased as low as 20 inches. Normal water level is 30 inches. The administrative manual scram point was 15 inches and the automatic scram set point for reactor water low level was about 8 inches.

The licensee documented in the prompt investigation of the event that the NSO misread the step in the procedure. The NSO thought he was required to verify that the 2A FWRV was NOT in test prior to operating the valve.

In addition, the inspectors interviewed the unit supervisor that was supposed to conduct the pre-job brief for the evolution. The unit supervisor stated that he had not performed a pre-job brief for the return to service of the 2A FWRV. The licensee also identified this fact in the prompt investigation and the apparent cause evaluation.

The inspectors reviewed DOP 0600-06. The procedure states in the Precautions section E, step 3, that moving a FWRV in MAN (manual) will cause the other FWRV to

move if it is in AUTO due to the demand balancing logic; and in step 9, placing an isolated FWRV in the TEST mode assures that any movement of the valve will NOT result in the demand balancing logic acting on the operating FWRV and causing a RPV [reactor pressure vessel] level response prior to assertion of level dominance mode by the feedwater level control (FWLC) logic. The inspectors interviewed the NSO who stated that he had read the precautions and limitations sections of DOP 0600-06 when the 2A FWRV was taken out-of-service but not when he was restoring the valve to operation. The NSO estimated the time period between reading the precaution steps and needing to apply them during the 2A FWRV restoration to be about 2-3 hours.

Based on the information in the procedure and the interviews conducted, the inspectors concluded that the operator not only misread the step, but that the operating crew also failed to properly prepare to perform the evolution. The unit supervisor did not conduct a pre-job brief for the restoration of the 2A FWRV and the NSO also failed to properly prepare for the evolution by completely reviewing the precautions and applicable steps of DOP 0600-06 just prior to the restoration of the 2A FWRV.

<u>Analysis</u>: The inspectors determined that the failure to place the 2A FWRV into test before performing the post-maintenance test was contrary to step G.14.a of procedure DOP 0600-06, “Feedwater Regulating Valve (FWRV) Operation,” Revision 39, and was a performance deficiency.

The finding was determined to be more than minor because the finding could be reasonably viewed as a precursor to a significant event. Specifically, the event could have lead to a reactor scram. The inspectors concluded this finding was associated with the Initiating Events Cornerstone.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, “Significance Determination Process,” Attachment 0609.04, “Phase 1 - Initial Screening and Characterization of Findings,” Table, 4a for the Initiating Events Cornerstone. Since the finding did not contribute to both the likelihood of a reactor scram and the likelihood that mitigation equipment or functions will not be available, the finding screened as Green.

This finding has a cross-cutting aspect in the area of human performance, work practices because, as stated the above, the licensee did not ensure the proper use of human error prevention techniques. Specifically, the failure to place the 2A FWRV into test was a human error which could have been prevented by performing a pre-job brief and/or by the operator properly preparing by reading the precautions section of the procedure before proceeding. [H.4 (a)]

The licensee disagreed with the inspectors’ characterization of the cross-cutting issue. Licensee management personnel stated that, because the operator stated that he had read the precautions earlier and that he misread the procedure step, performing a pre-job brief would not have changed the outcome of this event. The licensee stated that even if a pre-job brief had been performed this error would still have occurred. The inspectors disagreed with the licensee’s conclusion that this event was unpreventable even if a pre-job brief had been performed. The licensee’s apparent cause evaluation gave four causal factors for this event. Two of the causal factors were associated with the lack of a pre-job brief.

Enforcement: Technical Specification Section 5.4.1.a, states, in part, that "Written procedures shall be established, implemented, and maintained covering the following activities: The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978." Paragraph 4.o, of this Regulatory Guide, states, in part, that procedures for operation of the feedwater system shall be prepared and activities shall be performed in accordance with these procedures. The licensee established DOP 0600-06, "Feedwater Regulating Valve (FWRV) Operation," Revision 39, as an implementing procedure for operation of the feedwater regulating valves. DOP 0600-06, Step G.14.a, "Returning FWRV to Service," states, "Verify on-coming FWRV in TEST per Step G.12."

Contrary to the above, on May 22, 2011, an NSO failed to follow procedure DOP 0600-06, "Feedwater Regulating Valve (FWRV) Operation," Revision 39. Step G.14.a during the return to service of the 2A FWRV. The 2A FWRV was not put into test before post-maintenance testing of the 2A FWRV was performed. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program as IR 1218997, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy (**NCV 05000237/2011003-07; 05000249/2011003-07**, "Change In Unit 2 Reactor Water Level Due To A Failure To Follow Procedure"). The licensee's immediate corrective action was to restore reactor water level and relieve the reactor operator from duty. The licensee required each of the shift managers to reinforce the procedure definition of critical steps to be identified at all pre-job briefs for 60 days.

.2 Failure to Place One Channel of the Rod Block Monitor Into Trip

Introduction: A Green NCV of Technical Specification (TS) 3.3.2.1 was identified by the inspectors for the licensee's failure to place one channel of the Rod Block Monitor (RBM) into trip within 1 hour after discovering that both channels of the RBM on Unit 3 were inoperable.

Description: On January 15, 2011, while performing control rod exercising on Unit 3, the NSO noticed that when selecting certain non-peripheral control rods near the edge of the core, specifically, rods that have three local power range monitor (LPRM) strings associated with them, a fourth LPRM reading was indicated on the control board. The operations crew at the time determined that the RBM remained operable and proceeded with control rod exercising. Afterward, an Issue Report (IR 1163326) was written to troubleshoot the RBM anomaly. During troubleshooting on January 21, 2011, the licensee discovered that a diode had shorted in a RBM relay card, causing the signals from three specific LPRM strings to be included in the RBM circuit whenever any of the control rods with three associated LPRM strings was selected.

Since each LPRM string has four LPRMs, and each channel of the RBM uses two LPRMs from each LPRM string, then each RBM channel has six inputs when a control rod with three associated LPRM strings is selected. The RBM also has a count circuit to fulfill the "Inop" function required by TS 3.3.2.1 that verifies that each RBM channel has at least 50 percent of the total possible inputs. Because of the shorted diode, when some of these control rods were selected, instead of having six inputs, each RBM channel now had eight. Even though at the time of the failure there were sufficient real LPRM inputs into the RBM, because of this condition, the count circuit would not be able

to identify if there were fewer than 50 percent of the real inputs, rendering the Inop functions of both RBM channels inoperable.

Upon discovery of the shorted diode on January 21, 2011, the licensee declared the RBM inoperable and placed one channel into trip in accordance with TS 3.3.2.1. The licensee then replaced the relay card that contained the failed diode and returned the RBM to service that same day. These activities were documented in IR 1165508. The licensee then performed a past operability evaluation to determine if the RBM had been inoperable from the time of discovery on January 15, 2011, (IR 1183073). In this evaluation, the license determined that the Upscale and Downscale functions of the RBM were maintained, but did not identify that the count circuit in the RBM was not able to fulfill the Inop function required by TS 3.3.2.1. Also, the licensee noted that the Rod Withdrawal Error (RWE) analysis for the reactor does not assume the RBM. Therefore, the licensee determined that the RBM had been operable until it was taken out-of-service for repair.

The inspectors questioned this operability call, noting that previous IRs had determined the Inop function to be non-conservative with these additional false inputs. After further review, the licensee agreed with the inspectors and submitted a Licensee Event Report within the 60 day timeframe required by 10 CFR 50.73(a)(1) and initiated an Apparent Cause Evaluation (IR 1189950) to evaluate the issue and human performance gaps in not identifying the failed instrument and to determine corrective actions. In this Apparent Cause Evaluation (ACE), the licensee found that when the anomaly was identified, the NSO brought the anomaly to the attention of the Unit Supervisor (US) and then proceeded to exercise rods. As the ACE states, "this is contrary to OP-AA-101-111 Section 4.2.10 which requires the US to ensure abnormal conditions are investigated, including initial troubleshooting, verifying proper information is gathered and verifying appropriate corrective actions are established." The ACE also identified that "the Shift Manager failed to utilize OP-AA-101-111-1003, Attachment 7, to help guide him in determining operability by engaging proper assistance as necessary to ensure that immediate operability is thoroughly assessed." The ACE determined this to be the apparent cause for the licensee not meeting the Technical Specification Required Action.

Analysis: The inspectors determined that the licensee's failure to place one RBM channel on Unit 3 in trip within 1 hour as required upon meeting Condition B, "Two RBM channels inoperable," of Technical Specification 3.3.2.1 was a performance deficiency warranting a significance evaluation in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," issued on December 24, 2009.

The inspectors determined that the finding was more than minor because, if left uncorrected, it could lead to a more significant safety concern. Specifically, if a RBM channel had fewer than three real inputs, but was counting false inputs due to the shorted diode, it would be unable to fulfill its Inop function, and would not insert a rod block to prevent a RWE.

The inspectors performed a Phase 1 Significance Determination in accordance with IMC 0609, Attachment 4. The inspectors concluded that this finding affected the Mitigating Systems Cornerstone since it was a degradation of reactivity control. The finding screened as Green, or very low safety significance, in Table 4a of Attachment 4, since it was not a design or qualification deficiency and did not represent

an actual loss of safety system function. This is because the RBM maintained a sufficient number of real inputs and because the licensee's RWE analysis does not assume the RBM function.

The inspectors determined that this finding has a cross-cutting aspect in the area of Problem Identification and Resolution in the component of the Corrective Action Program, since it involved the licensee's failure to thoroughly evaluate problems. Specifically, when the RBM anomaly was identified, the Unit Supervisor did not ensure abnormal conditions were investigated, nor did the Shift Manager engage proper assistance as necessary to ensure that immediate operability was thoroughly assessed. In addition, the licensee did not properly evaluate for operability and reportability after the troubleshooting and repair of the RBM had occurred, until prompted by the inspectors. [P.1(c)]

Enforcement: Technical Specification 3.2.1.1, Required Action B.1, requires the licensee to place one channel of the Rod Block Monitor into trip within one hour when it is identified that both channels of the RBM are inoperable. Contrary to the above, on January 15, 2011, when the licensee identified that false inputs were being provided to both channels of the RBM, the licensee did not declare both channels inoperable and did not place one channel into trip. The licensee also proceeded to exercise control rods, which would have been prevented by a rod block if one channel of the RBM had been placed into trip. Action Request 1163326 was generated to troubleshoot the RBM issue.

The licensee was in this condition until January 21, 2011, when they performed troubleshooting on the issue and declared the RBM inoperable to repair the failed component. There was no actual safety consequence because there were still sufficient real inputs into the RBM, and the Rod Withdrawal Error analysis for the plant does not assume the RBM function. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program, this violation is being treated as an NCV, consistent with the Section 2.3.2 of the NRC Enforcement Policy. (**NCV 05000249/2011003-08**, "Failure to Place One Channel of the Rod Block Monitor Into Trip").

.3 (Closed) Licensee Event Reports (LERs) 05000249/2011-001-00 and 05000249/2011-001-01, "Control Rod Block Instrumentation Failure"

a. Inspection Scope

The inspectors reviewed LERs 249/2011-001-00 and 249/2011-001-01, "Control Rod Block Instrumentation Failure," to ensure that the issues documented in the report were adequately addressed in the licensee's corrective action program.

b. Findings

On January 15, 2011, while performing control rod exercising on Unit 3, the NSO noticed that when selecting certain non-peripheral control rods near the edge of the core, specifically, rods that have three LPRM strings associated with them, a fourth LPRM reading was indicated on the control board. The operations crew at the time determined that the RBM remained operable and proceeded with control rod exercising. Documents reviewed as part of this inspection are listed in the Attachment to this report.

The enforcement aspects of this finding are discussed in Section 4OA3, Event Follow-Up.

These LERs are closed.

These event follow-up reviews constituted two samples as defined in IP 71153-05.

.4 <u>(Closed) Licensee Event Report 05000249/2010-002-00, "MSIV Leakage Exceeds Technical Specifications Allowable Limits"</u>

a. <u>Inspection Scope</u>

The inspectors reviewed LER 249/2010-002-00, "MSIV Leakage Exceeds Technical Specifications Allowable Limits," to ensure that the issues documented in the report were adequately addressed in the licensee's corrective action program. Documents reviewed as part of this inspection are listed in the Attachment to this report.

b. <u>Findings</u>

On November 1, 2010, Dresden Unit 3 had been shutdown for a refueling outage. After entering Mode 4, plant personnel performed the local leak rate test (LLRT) for the main steam isolation valves (MSIVs). During the testing it was identified that leakage rates on three MSIVs exceeded the allowable limits specified in the Technical Specifications.

The enforcement aspects of this finding are discussed in Section 4OA7, Licensee-Identified Violations.

This LER is closed.

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

4OA5 <u>Other Activities</u>

.1 <u>(Closed) Temporary Instruction 2515/179, "Verification of Licensee Responses to NRC Requirement for Inventories of Materials Tracked in the National Source Tracking System Pursuant to Title 10, Code of Federal Regulations, Part 20.2207 (10 CFR 20.2207)"</u>

a. <u>Inspection Scope</u>

The inspectors confirmed that the licensee has reported the initial inventories of sealed sources pursuant to 10 CFR 20.2207 and determined that the National Source Tracking System database correctly reflects the Category 1 and 2 sealed sources in custody of the licensee. Inspectors interviewed personnel and performed the following:

- Reviewed the licensee's source inventory;
- Determined the presence of any Category 1 or 2 sources;
- Reviewed procedures for and evaluated the effectiveness of storage and handling of sources;
- Reviewed documents involving transactions of sources; and

- Reviewed adequacy of licensee maintenance, posting, and labeling of nationally tracked sources.

b. Findings

No findings were identified.

.2 (Closed) NRC Temporary Instruction 2515/183, “Followup to the Fukushima Daiichi Nuclear Station Fuel Damage Event”

The inspectors assessed the activities and actions taken by the licensee to assess its readiness to respond to an event similar to the Fukushima Daiichi nuclear plant fuel damage event. This included (1) an assessment of the licensee’s capability to mitigate conditions that may result from beyond design basis events, with a particular emphasis on strategies related to the spent fuel pool, as required by NRC Security Order Section B.5.b issued February 25, 2002, as committed to in severe accident management guidelines, and as required by 10 CFR 50.54(hh); (2) an assessment of the licensee’s capability to mitigate station blackout (SBO) conditions, as required by 10 CFR 50.63 and station design bases; (3) an assessment of the licensee’s capability to mitigate internal and external flooding events, as required by station design bases; and (4) an assessment of the thoroughness of the walkdowns and inspections of important equipment needed to mitigate fire and flood events, which were performed by the licensee to identify any potential loss of function of this equipment during seismic events possible for the site.

Inspection Report 05000237/2011009; 05000249/2011009 (ML111320349) documented detailed results of this inspection activity. Following issuance of the report, the inspectors conducted detailed follow-up on selected issues.

.3 (Closed) NRC Temporary Instruction 2515/184, “Availability and Readiness Inspection of Severe Accident Management Guidelines (SAMGs)”

On May 19, 2011, the inspectors completed a review of the licensee’s severe accident management guidelines (SAMGs), implemented as a voluntary industry initiative in the 1990’s, to determine (1) whether the SAMGs were available and updated, (2) whether the licensee had procedures and processes in place to control and update its SAMGs, (3) the nature and extent of the licensee’s training of personnel on the use of SAMGs, and (4) licensee personnel’s familiarity with SAMG implementation.

The results of this review were provided to the NRC task force chartered by the Executive Director for Operations to conduct a near-term evaluation of the need for agency actions following the Fukushima Daiichi fuel damage event in Japan. Plant-specific results for Dresden Nuclear Power Station, Units 2 and 3 were provided as an Enclosure to a memorandum to the Chief, Reactor Inspection Branch, Division of Inspection and Regional Support, dated June 1, 2011, (ML111520396).

.4 (Closed) Unresolved Item (URI) 05000237/2005002-01; 05000249/2005002-01: “Post-Fire Safe Shutdown Circuit Analysis not Consistent with RIS 2004-003”

During the 2005 triennial fire protection inspection, the NRC identified an URI concerning the licensee's Post-Fire Safe Shutdown (SSD) circuit analysis not being consistent with NRC Regulatory Issue Summary (RIS) 2004-003, Revision 1, “Risk-Informed Approach

for Post-Fire Safe Shutdown Circuit Inspection." Specifically, the licensee's post-fire SSD circuit analysis considered only single instead of multiple fire-induced spurious actuations of safe shutdown components.

Subsequent to the inspection in 2005, two specific aspects of fire-induced circuit cable faults were addressed by the NRC. The first issue involved fire-induced single circuit cable faults and associated operator manual actions. The NRC Enforcement Guidance Memorandum (EGM) 07-004, "Enforcement Discretion for Post-Fire Manual Actions used as Compensatory Measures for Fire Induced Circuit Failures," authorized enforcement discretion for such non-compliance issues until March 6, 2009.

The second issue involved fire-induced multiple circuit cable faults and associated operator manual actions. The NRC Enforcement Guidance Memorandum 09-002, "Enforcement Discretion for Fire Induced Circuit Faults," dated May 14, 2009, authorized enforcement discretion for such non-compliance issues, provided that licensees identified the non-compliances, entered them into their corrective action programs, and instituted appropriate compensatory measures until the issues were corrected, within the six-month period following a planned revision to Regulatory Guide (RG) 1.189, "Fire Protection for Nuclear Power Plants." RG 1.189, Revision 2, issued in October 2009, provided a method acceptable to the NRC to evaluate and resolve multiple fire-induced circuit faults. After the six-month period for identification of non-compliances, the EGM further authorized enforcement discretion for an additional 30-month period, for the licensee to resolve the identified multiple fire-induced circuit fault issues.

EGM 07-004, EGM 09-002 and RG 1.189, Revision 2, provided adequate technical guidance and an acceptable time table to evaluate and resolve the non-compliances identified during the six months including the issue of fire-induced cable faults tracked by this URI. The adequacy of licensee actions to address these issues will continue to be reviewed within the framework of the NRC's reactor oversight process, which includes the triennial fire protection team inspections and problem identification and resolution inspections. Therefore, URI 05000373/2005006-02; 05000374/2005006-02 is no longer necessary to track this issue and is closed.

The inspectors' review of this issue was considered to be a part of the original inspection effort, and as such did not constitute any additional inspection samples.

.5 <u>(Closed) Unresolved Item 05000237/2008002-03; 05000249/2008002-03: "LPCI Room Equipment EQ Qualification Deficiencies and Discrepancies"</u>

During the 2008 heat sink inspection, the NRC identified an unresolved item regarding the ability of the Low Pressure Coolant Injection (LPCI) Room safety-related equipment to perform post-loss-of-coolant accident (LOCA) design safety functions without LPCI room cooling. The inspectors noted that Dresden equipment qualification (EQ) binders concluded that safety-related equipment throughout the Reactor Building was adequately environmentally qualified for post-extended power uprate (EPU), post-LOCA conditions without the LPCI room coolers and the associated fans. The inspectors questioned whether the motor used in testing was acceptable. Specifically, the licensee's analysis relied on an EQ test that used a motor that was not identical to the motors installed at Dresden.

Subsequent to the inspection in 2008, additional questions were raised by the NRC requiring additional clarification and analysis by the licensee. The inspectors reviewed the environmental qualification report for the LPCI motor. While the motor tested was not exactly the same as the motor in the pump, the motor in the EQ report was Type tested and qualified by similarity analysis. For the qualification, the licensee evaluated any dissimilarities and justified those within the report. Based upon their review, the inspectors could find no violation of NRC requirements.

Therefore, URI 05000237/2008002-03; 05000249/2008002-03 is no longer necessary to track this issue and is closed.

The inspectors' review of this issue was considered to be a part of the original inspection effort, and as such did not constitute any additional inspection samples.

4OA6 Management Meetings

.1 Exit Meeting Summary

On July 14, 2011, the inspectors presented the inspection results to Mr. D. Czufin, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

.2 Interim Exit Meetings

Interim exits were conducted for:

- The radioactive solid waste processing and radioactive material handling, storage, and transportation under the public and occupational radiation safety cornerstones, and a performance indicator verification was discussed with Mr. T. Hanley, Site Vice President, on February 18, 2011.

- The results of the review of Unresolved Item 05000237/2005002-01; 05000249/2005002-01 concerning the licensee's post-fire SSD circuit analysis failure to consider multiple fire induced spurious actuations of safe shutdown components were discussed with Mr. S. Marik on April 13, 2011.

- Inventories of materials tracked in the national source tracking system, radiological hazard assessment and exposure control, and in-plant airborne radioactivity control/ mitigation with Mr. D. Czufin, Site Vice President, on June 10, 2011.

- The results of the review of Unresolved Item 05000237/2008002-03; 05000249/2008002-03 concerning the licensee's LPCI Room Equipment EQ Qualification analysis were discussed with Mr. T. Loch on June 22, 2011.

- On Thursday, July 7, 2011, the inspectors presented the Triennial Heat Sink inspection results to Mr. S. Marik, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

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

4OA7 Licensee-Identified Violations

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

**Cornerstone: Barrier Integrity**

Technical Specification (TS) Surveillance Requirement 3.6.1.3.10 requires that the leakage rate through each MSIV leakage path is less than or equal to 34 standard cubic feet per hour (scfh) when tested at greater than or equal to 25 pound-force per square inch gauge (psig), and the combined leakage rate for all MSIV leakage paths is less than or equal to 86 scfh when tested at greater than or equal to 25 psig. Contrary to this, on November 1, 2010, the licensee identified that the leakage rate on three MSIVs exceeded the allowable limits specified in TS Surveillance Requirement 3.6.1.3.10. The 3-0203-1C, 3-0203-1D and 3-0203-2D valves were found to have 52.8, 34.6 and 36.9 scfh leakages, respectively. Based on the as-found leakage rates, Surveillance Requirement 3.6.1.3.10 was not met. The licensee documented this condition in issue reports (IRs) 1133829, 1133832 and 1133833, respectively. The inspectors evaluated this finding using IMC 0609.04, “Phase 1 – Initial Screening and Characterization of Findings.” The inspectors answered “No” to all the questions on Table 4a, “Characterization Worksheet for IE, MS, and BI Cornerstone,” under the Containment Barrier column, therefore, the finding screened as of very low safety significance (Green). In addition, the combined MSIV leakage was within the allowable leakage limits for overall primary containment. The three MSIVs that exhibited excessive leakage were repaired and successfully retested.

ATTACHMENT: SUPPLEMENTAL INFORMATION

## SUPPLEMENTAL INFORMATION

## KEY POINTS OF CONTACT

Licensee

D. Czufin, Site Vice President
S. Marik, Station Plant Manager
H. Bush, Radiation Protection Manager
J. Cady, Manager of RP Technical Support
P. DiSalvo, GL 89-13 Program Owner
J. Fox, Design Engineer
G. Gates, Operations
D. Glick, Shipping Specialist
G. Graff, Nuclear Oversight Manager
D. Gronek, Operations Director
M. Hosain, Site EQ Engineer
G. Ice, Regulatory Assurance – NRC Coordinator
L. Jordan, Training Director
B. Kapellas, Work Control Manager
J. Knight, Chemistry Manager
M. Knott, Instrument Maintenance Manager
D. Leggett, Regulatory Assurance Manager
T. Loch, Design Engineering Manager
G. Marrow, Operations
M. McDonald, Maintenance Director
T. Mohr, Engineering Program Manager
P. O'Brien, Site Corrective Action Manager
P. O'Connor, Licensed Operator Requalification Training Lead
D. O'Flannagan, Security Manager
M. Overstreet, Lead Radiation Protection Supervisor
M. Pavey, Sr., Radiation Protection Technical Specialist
P. Quealy, Emergency Preparedness Manager
B. Rybak, Reg. Assurance
R. Ruffin, Licensing Engineer
J. Sipek, Engineering Director

Nuclear Regulatory Commission

M. Ring, Chief, Division of Reactor Projects, Branch 1
B. C. Dickson, Chief, DRS/RIII
A. M. Stone, Chief, Engineering Branch 2
J. Bowen, NRR/DIRS/IRIB

IEMA

R. Schulz, Illinois Emergency Management Agency

## LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

| | | |
|---|---|---|
| 05000237/2011003-01<br>05000249/2011003-01 | URI | Failure to Include Adequate Acceptance Criteria in a Surveillance Test (1R01) |
| 05000237/2011003-02<br>05000249/2011003-02 | NCV | Inadequate Instructions for the Inspection of Safety-Related Portions of the Intake Structure (1R07T.b(1)) |
| 05000237/2011003-03<br>05000249/2011003-03 | NCV | Inadequate Acceptance Criteria for Testing Equipment Relied Upon to Mitigate the Consequences of a Lock and Dam Failure (1R07T.b(2)) |
| 05000237/2011003-04<br>05000249/2011003-04 | NCV | Inadequate Instructions for Coping with a Dresden Lock and Dam Failure (1R07T.b(3)) |
| 05000237/2011003-05<br>05000249/2011003-05 | NCV | Inadequate Scoping of a Non-Safety-related Pump Into Maintenance Rule (1R07T.b(4)) |
| 05000237/2011003-06<br>05000249/2011003-06 | URI | Concerns with the Licensing and Design Basis (1R07T.b(5)) |
| 05000237/2011003-07<br>05000249/2011003-07 | NCV | Change In Unit 2 Reactor Water Level Due To A Failure To Follow Procedure (71153.1) |
| 05000249/2011003-08 | NCV | Failure to Place One Channel of the Rod Block Monitor Into Trip (71153.2) |

Closed

| | | |
|---|---|---|
| 05000237/2011003-02<br>05000249/2011003-02 | NCV | Inadequate Instructions for the Inspection of Safety-related Portions of the Intake Structure (1R07T.b(1)) |
| 05000237/2011003-03<br>05000249/2011003-03 | NCV | Inadequate Acceptance Criteria for Testing Equipment Relied Upon to Mitigate the Consequences of a Lock and Dam Failure (1R07T.b(2)) |
| 05000237/2011003-04<br>05000249/2011003-04 | NCV | Inadequate Instructions for Coping with a Dresden Lock and Dam Failure (1R07T.b(3)) |
| 05000237/2011003-05<br>05000249/2011003-05 | NCV | Inadequate Scoping of a Non-Safety-related Pump Into Maintenance Rule (1R07T.b(4)) |
| 05000237/2011003-07<br>05000249/2011003-07 | NCV | Change In Unit 2 Reactor Water Level Due To A Failure To Follow Procedure (71153.1) |
| 05000249/2011003-08 | NCV | Failure to Place One Channel of the Rod Block Monitor Into Trip (71153.2) |

| | | |
|---|---|---|
| 2515/179 | TI | Verification of Licensee Responses to NRC Requirement for Inventories of Materials Tracked in the National Source Tracking System Pursuant to Title 10, Code of Federal Regulations, Part 20.2207 (4OA5.1) |
| 2515/183 | TI | Followup to the Fukushima Daiichi Nuclear Station Fuel Damage Event (4OA5.2) |
| 2515/184 | TI | Availability and Readiness Inspection of Severe Accident Management Guidelines (4OA5.3) |
| 05000237/2005002-01<br>05000249/2005002-01 | URI | Post-Fire Safe Shutdown Circuit Analysis not Consistent with RIS 2004-003 (4OA5.4) |
| 05000237/2008002-03<br>05000249/2008002-03 | URI | LPCI Room Equipment EQ Qualification Deficiencies and Discrepancies (4OA5.5) |
| 05000249/2011-001-00 | LER | Control Rod Block Instrumentation Failure (71153.3) |
| 05000249/2011-001-01 | LER | Control Rod Block Instrumentation Failure (Supplement) (71153.3) |
| 05000249/2010-002-00 | LER | MSIV Leakage Exceeds Technical Specifications Allowable Limits (71153.4) |

## LIST OF DOCUMENTS REVIEWED

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

1R01 Adverse Weather Protection (71111.01)

- DOA 6500-12, "Low Switchyard Voltage," Revision 22
- WC-AA-101, "On-Line Work Control Process," Revision 18
- WC-AA-107, "Seasonal Readiness," Revision 9
- OP-AA-108-107-1001, "Station Response to Grid Capacity Conditions," Revision 3
- OP-AA-108-107-1002, "Interface Procedure Between ComEd/PECO and Exelon Generation (Nuclear/Power) for Transmission Operations," Revision 5
- WO 872864, "D2/3 6Y PM Emergency Diesel Pump (Flood Pump) Operation"
- DOA 0010-04, "Floods," Revision 31
- IR 1203243, "Flood Pump Flow During Test Point Was Lower Than Expected"
- IR 1198074, "IER 11-1 Review of DOA 0010-04 Floods"
- IR 1209642, "NRC Identified URI With Flood Pump Acceptance Criteria"

1R04 Equipment Alignment (71111.04Q&S)

- DOP 1100-M1, "Unit 2 Standby Liquid Control (SBLC) System," Revision 13
- DOP 1100-E1, "Unit 2 Standby Liquid Control Electrical Checklist," Revision 7
- DOP 2300-M1/E1, "Unit 3 HPCI System Checklist," Revision 37
- DOP 1400-E1, "Unit 2 Core Spray Electrical," Revision 3
- DOP 1400-M1, "Unit 2 Core Spray System," Revision 23
- DOP 1400-01, "Core Spray System Preparation for Standby Operation," Revision 13
- M-27, "Diagram of Core Spray Piping," Revision AAN

1R05 Fire Protection (71111.05)

- IR 1120945 – "remove TCP for SBLC and chem permit, SBLC chem. gone extend due date as required, inform fire marshall"
- Fire Load Calculation, Calculation No. DRE97-0105, Revision 08, Amendment 16
- IR 1213881, "Initial Fire Brigade Students Lacking Training"
- Fire Drill Scenario: U3 RX MCC 38-4 Cube A4
- OP-AA-201-003, "Fire Drill Performance," Revision 12
- OP-AA-201-005, "Fire Brigade Qualification," Revision 07

1R07 Triennial Heat Sink Performance (71111.07T)

- ER-AA-340-1002, Service Water Heat Exchanger Inspection Guide 4
- ER-AA-340-1001, GL 89-13 Program Implementation Instructional Guide 7
- ER-AA-450, Structures Monitoring 0
- DCP 1008-04, Heat Exchanger Inspection Program 8
- DGA-12, Partial or Complete Loss of AC Power 63
- DMP 1500-03, Containment Cooling (LPCI) Heat Exchanger Maintenance 28
- DOA 0010-01, Dresden Lock and Dam Failure 29

- DOS 0010-01, Dresden Dam Failure Equipment Test 20
- DOP 1300-03, Manual Operation of the Isolation Condenser 32
- DTS 3900-07, Crib House/Intake Structure Inspections 13
- IR 01045554, Abnormal Noise When Running U3 EDGCWP, dated 03/21/2010
- IR 00989609, D2R21 Inspection Results for 2A LPCI Heat Exchanger, dated 11/05/2009
- M-355, Diagram of Service Water Piping, Revision SB
- M-22, Diagram of Service Water Piping, Revision EB
- WO 00840066, D2/3 4YR COM SCRN REFUSE PMP 'B' OVRHL AND MOTOR, dated 10/01/08
- WO 00851878, EP Perform Pressure Testing of DGSW Piping Block 2DG01, dated 12/27/2006
- WO 00993106, D2 2Yr PM Crib Hse PP Bay A Insp/Clean/Chem Add, dated 11/03/2008
- WO 01055959, D2/3 2Y PM Screen Refuse Pit Discharge Line to Bay 13 Flush, dated 05/19/09
- WO 01306101, D2/3 Qtr Com Insp Fire Pump Bay/Dwnstrm Screen With Diver, dated 02/17/2010
- WO 01316876, D2/3 AN PM Grease Pmp Bearings 2/3 A Screen Refure Pp, dated 06/10/11
- WO 01354572, Thermal Performance Test – 2A LPCI HX, dated 12/27/2010
- WO 01302369, Thermal Performance Test – 2A LPCI HX, dated 6/15/2010
- 2-1503-A, 2A LPCI - Eddy Current Examination Final Report, November 2007
- 2-1503-A, 2A LPCI - Eddy Current Examination Final Report, October 2003
- ER-AA-340-1002 Attachment 1, Unit 2 EDG Heat Exchanger A Inspection and Service, dated 02/16/2009
- ER-AA-340-1002 Attachment 1, Unit 2 EDG Heat Exchanger A Inspection and Service, dated 02/13/2007
- ER-AA-340-1002 Attachment 1, Unit 2 LPCI Heat Exchanger A Inspection and Service, dated 11/05/2009
- Engineering Evaluation, Evaluation of Re-routing of DGCW Line Discharge Piping on System Performance
- OP-AA-108-115, Operability Evaluation for Failed Thermal Performance Test of 2A LPCI Heat Exchanger, Revision 1
- IR 01173062, NRC Identifies UFSAR Discrepancy, dated 02/09/2011
- IR 01173579, NRC Heat Sink Inspection – Bay 13 Inspection Corrosion, dated 02/10/2011
- IR 01202741, Enhancement to DGA 12, dated 04/14/2011
- IR 01212196, Error Identified in DOA 0010-01, dated 05/05/2011
- IR 01221386, NRC Id's Issue with Procedure for Dam Failure, dated 05/27/2011
- IR 01221390, NRC Id's Issues With Surveillance Procedure for Dam Failure, dated 05/27/2011
- IR 01221421, 2/3 Screen Refuse Not in Maintenance Rule, dated 05/27/2011

1R11 Licensed Operator Requalification Program (71111.11)

- Simulator Scenario Guide OPEX AJ
- Simulator Scenario Guide OPEX 09-11-12B

1R12 Maintenance Effectiveness (71111.12)

- IR 927865,"Core Spray Keep Fill Stop Check Valve Stuck Closed"
- Expert Panel Meeting Minutes August, 26, 2010
- Expert Panel Meeting Minutes September 16, 2010
- Expert Panel Meeting Minutes October 21, 2010
- Expert Panel Meeting Minutes June 7, 2010

- Expert Panel Meeting Minutes July 14, 2010
- Expert Panel Meeting Minutes August 26, 2010

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

- IR1210760, “NRC SR. Resident Concern Identified”
- IR 1211156,”Procedure Enhancement Identified”
- OP-AA-108-103, “Locked Equipment Program,” Revision 2

1R15 Operability Evaluations (71111.15)

- EC 382211, “Historical Operability Evaluation of Main Steam Loop “A” Line Due to Failed Snubbers 3-3001A-42, 3-3001A-46, 3-3001A-47, and 3-0203-34 (Dwg. M-564J Sht. 3, Dwg. M-564J Sht. 1, Dwg. M-564J Sht. 2, and Dwg. M-1143 Sht. 25),” Revision 0
- EC 382208, “Historical Operability Evaluation of Main Steam Loop “B” Line Due to Failed Snubber 3-3001B-41 (Dwg. M-564K, Sht. 3),” Revision 0
- EC 382212, “Historical Operability Evaluation of Main Steam Loop C Line Due to Failed Snubber 3-3001C-50 (Drawing M-564L, Sht. 2),” Revision 0
- EC 382210, “Historical Operability Evaluation of Main Steam Loop “D” Line Due to Failed Snubbers 3-3001D-48 and 3-3019D-55 (Dwg. M-564M, Sht. 2 and M-564M, Sht. 8),” Revision 0
- IR 1137413, “Snubber 3-3001A-42 Failed Functional Test”
- IR 1137959, “Snubber 3-3001A-46 Failed Functional Testing”
- IR 1138468, “Snubber 3-0203-34 Failed Functional Testing”
- IR 1137195, “Snubber 3-3001A-47 Failed Functional Testing”
- WO 1306122, “Add Snubbers to D3R21 Scope for Service Life Monitoring”
- WO 1194624, “D3 RFL TS Perform 10% Sample Selection Criteria per DTS 0020”
- IR 1137434, “Snubber 3-3001B-41 Failed Functional Testing”
- IR 1135133, “Snubber 3-3001C-50 Failed Functional Testing”
- IR 1135809, “D3R21 Snubber 3-3019D-55 Exceeded Piston Setting Tolerance”
- IR 1136705, “Snubber 3-3019D-55 Failed Functional Test”
- IR 1135144, “Snubber 3-3001D-48 Failed Functional Testing”
- NES-MS-03.2, “Evaluation of Discrepant Piping and Support Systems,” Revision 6, dated March 19, 2008
- IR 1191149, “Corp Eng Oversight Review of Dresden Op-Eval 11-002”
- IR 1094902, “Apparent Vavle Seat Leakage 3 DGCW Flow Directing Valves”
- IR 1107448, “Op-Eval Actions Cannot Be Completed As Directed”
- IR 1108123, “NOS ID Incomplete Basis Included in DGCW OP Eval”
- IR 1167380, “DGCW 3-Way Valve Disassembly Inspection Results”
- IR 1186363, Trending-Possible Unit 2 DGCW 3-Way Valve Degradation”
- IR 1199727, “NOS ID Issues With Op Eval For EDG Cooling Water”
- IR 1191149,”Corp Eng Oversight Review of Dresden Op-Eval 11-002”
- EC 378526, “X-Area MSLB Effects not Evaluated for Adjacent SDC Pump and SDC HX Rooms – Op Eval 10-001,” Revision 005
- EC 378389, “Determine the Bounding Parameters for RB 517’-6” Elevation and RB 545’-6” Elevation (EQ Zone 28) from a HELB in the MS Tunnel with Open Side Hatch,” Revision 000
- EC 378565, “Install Conduit Seals on Pressure Switches 2(3)-0263-111A, B, C & D to Address Op Eval 10-001,” Revision 000
- IR 1208568, “EDG Cracked Air Box Weld Extent of Condition”
- IR 1216522, “NOS ID – Op Eval Incomplete for EDG Air Box Support Cracks”
- 3B CS declared prompt operable. Engineering working on an Op.Eval.

1R18 Plant Modifications (71111.18)

- EC 362432, “3-1501-19B Install 4 Rotor Limit Switch, Add TS Open Protection, and Change Close Scheme to LS Control”
- WO 945926, “3-1501-19B Install 4 Rotor Lim Switch & Change to LS Control”
- EC 362432, “3-1501-1901 Install 4 Rotor Limit Switch, Add TS Open protection, and Change Close Scheme to LS Control,” Revision 001
- WO 927689, “D3 10Y Com MOV Diagnostic * Limitorque Surv 3-1501-19B”
- WO 945926, “3-1501-19B Install 4 Rotor Lim Switch & Change to LS Control”
- DEP 0040-09, “Limitorque Valve Operator Maintenance,” Revision 12
- MA-AA-723-300, “Diagnostic Testing of Motor Operated Valves,” Revision 4a

1R19 Post-Maintenance Testing (71111.19)

- WO 01026062-01 D2 4Y PM SBO D/G Overhaul AMOT Valve 2-6629-24A
- IR 1211325, “NRCIdentified Issue”
- IR 1197539, “Receipt of NRC Minor Violation”
- DMP 1500-05, “LPCI Pump Maintenance,” Revision 10
- DOS 1500-10, “LPCI System Pump Operability and Quarterly Test With Torus Available and In-service Testing (IST) Program,” Revision 66
- DOS 1400-05, “Core Spray System Pump Operability and Quarterly IST Test with Torus Available,” Revision 43
- DOP 1400-03, “ECCS Fill System,” Revision 54
- WO 1242544, “OP D2 Div 1 Core Spray System Venting”
- DOS 1400-07, “ECCS Venting,” Revision 29
- IR 1223405, “MOV 2-1402-4A, Flow Test Valve Timing”

1R22 Surveillance Testing (71111.22)

- WO 1230111, “D2 2Y TS 2B SBLC Pump Comprehensive Test for In-Serv Testing”
- DOS 1100-04, “Standby Liquid Control System Quarterly/Comprehensive Pump Test for the Inservice Testing (IST) Program,” Revision 44
- IR 1068175, “SBLC Test Tank Drain Valve Leaking By”
- IR 1092108, “Unable to Adjust Level Indicator 2-1156 to within Tolerance”
- IR 1132718, “LaSalle CDBI Issue, SBLC Test Tank Seismic Evaluation”
- IR 1169325, “Leak at Union on U2 SBLC Air Sparge Line”
- IR 1188330, “Need SBLC Test Tank Seismic Evaluation”
- IR 1190040, “Need Risk Assessment of SBLC Storage Tank Wall Thickness”
- IR 1199106, “Risk Associated with SBLC Tank Wall Thinning (If Found)”
- DOS 1500-19, “Operation of the Dresden Lock and Dam Failure CCSW Emergency Pump,” Revision 06
- DOA 0010-01, “Dresden Lock and Dam Failure,” Revision 29
- IR 1200223, “Enhancement Needed for DOS 1500-19”
- IR 1209183, “NRC Identified Change to DOS 1500-19 is Required”
- DFPS 4183-14, “Unit 2/3 Control Room HVAC Smoke Detector Annual Surveillance Procedure,” Revision 9
- DFPS 4183-14, “Unit 2/3 Control Room HVAC Smoke Detector Annual Surveillance Procedure,” Revision 14
- DFPS 4183-14, “Unit 2/3 Control Room HVAC Smoke Detector Annual Surveillance Procedure,” Revision 15

- DFPS 4183-14, “Unit 2/3 Control Room HVAC Smoke Detector Annual Surveillance Procedure,” Revision 16
- Diesel Fuel Sample Results, 4/11/2011 and 4/12/2011.

1EP6 Drill Evaluation (71114.06)

- Simulator Scenario Guide OPEX AJ

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

- RP-AA-376-1001; Radiological Posting, Labeling, and Marking Standard; Revision 5
- RP-AA-460; Controls for High and Locked High Radiation Areas; Revision 20
- RP-AA-460-001; Controls for Very High Radiation Areas; Revision 2
- RP-AA-460-002; Additional High Radiation Exposure Control; Revision 0
- RP-AB-460; Initial TIP Room Entry Data Sheet; Revision 1; April 19, 2011
- RP-AB-460; TIP Drive Mechanism Access Data Sheet; April 5, 2011; Revision 1
- RP-AA-460; Approval for HRS/LHRA Deviations; D-2 TIP Room; Revision 20; January 1, 2011
- RP-AA-460; Approval for HRS/LHRA Deviations; D-3 TIP Room; Revision 19; February 2, 2011
- RWP-10012301; Trouble Shooting and Repair of D-3 Off-Gas; Revision 0
- RWP-10011122; D-2 Tip Room Activities; March 16, 2010
- RWP-10012307; Repair “B” Max Recycle Concentrator Dump Valve; May 13, 2011
- LS-AA-126-1001; 1164214-02; Functional or Cross Functional Area Self Assessment; April 3, 2011
- IR-01227107; Catch Funnel in Rad Waste Basement Clogged and Overflowing During NRC Inspection; June 10, 2011
- IR-01035344; LHRA Enhancement for Main Turbine Fence; February 25, 2010
- IR-01110566; Unplanned Exposure Results from Issues with Concentrator Recirc Pumps; September 7, 2010
- IR-01205674; Safety Concern of LHRA Door will not Open from Inside; April 21, 2011
- IR-01077659; Radiation Concern Unit-3 at Elevation 570; June 7, 2010
- DMS-0040-06; Locked High Radiation Door Inspection Checklist; Revision 2
- D-2 Radiological Surveys for Noble Metal Injection Survey Data; from May 5 through May 31, 2011
- IR-01224170; Unidentified Material Discovered in Rad Waste Tank Farm; June 3, 2011
- Dresden RP-AA-460; HRA Key Log; from April 12 through May 26, 2011
- IR-01159034; Emergent Exposure Received for Unit-2 and Unit-3 SJAE Entries; January 5, 2011
- RP-AA-460; HRA and LHRA Briefing Form (CM-3); D-2 Fuel Pool Cooling, D-3 Shutdown Cooling Heat Exchangers’ D-2 Regen Room, D-2 Moisture Separator; June 7, 2011

2RS3 In-Plant Airborne Radioactivity Control and Mitigation (71124.03)

- RP-DR-870; DOP Testing Portable HEPA Filter Units; Revision 0
- DTS-7500-07; Standby Gas Treatment System Air Filter Unit Performance Requirements (Methyl Iodide Removal and Charcoal Leak Test) Unit 2 and 3; Revision 16; September 14, 2010
- NCS Corporation; Radioiodine Penetration/Efficiency Test Report; Dresden-2 and 3; September 23, 2010
- PSI Inc. Quarterly Service Air and Self Contained Breathing Apparatus-Performed April 26, 2011

- DTS-7500-11; DOP Testing of Dresden 2 and 3 Standby Gas Treatment HEPA Filters; Revision 10; September 14, 2010
- DOS 4650-01; Control Room Emergency Breathing Air System Surveillance; Revision 6
- RP-DR-831; MSA Self-Contained Breathing Apparatus Inspection; Revision 3
- RP-DR-831; MSA SCBA Spare Cylinder Inspection Logs; June 3, 2011
- RP-DR-831; MSA SCBA Inspection Checklist; June 3, 2011
- IR-01162919; Respirator Chemical Cartridges Greater than 3 years from the Manufacturer Recommendation; Communication from Byron Issues; January 14, 2011
- IR-01199023; Asbestos Vacuum within the RCA Not Controlled by RP; April 6, 2011
- IR-01201491; NOS Identified Outdated RP Procedure Associated with Airborne Alpha Concentration; April 12, 2011

<u>2RS8 Radioactive Solid Waste Processing and Radioactive Material Handling, Storage, and Transportation (71124.08)</u>

- RP-AA-600-1012; Use and Operation of WMG Software for Direct Sample Characterization and Generation of Shipping Paperwork; Revision 0
- RP-AA-600-1013; Use and Operation of WMG Software RAMSHP; Revision 0
- RP-AA-600-1004; Radioactive Waste Shipments to Energy Solutions' Clive Utah Disposal Site Containerized Waste Facility; Revision 08
- RP-AA-600-1007; Radioactive Waste Shipments to Energy Solutions' Clive Utah Disposal Facility Bulk Waste Facilit; Revision 05
- RP-AA-600-1011; Use and Operation of WMG Software for Gross Gamma Characterization and Generation of Shipping Paperwork; Revision 01
- RP-AA-600-1010; Use and Operation of WMG Software for Creating Containers Samples, Waste Streams and Waste Streams and Waste Types; Revision 00
- RP-AA-600-1005; Radioactive Material and Non Disposal Site Waste Shipments; Revision 12
- RP-AA-605; "Sample" Waste Stream Results Review; Concentrated Waste (CW-2009); L39296; dated April 23, 2009
- RP-AA-605; "Sample" Waste Stream Results Review; Condensate Resin; L39280; dated May 05, 2009
- RP-AA-605; "Sample" Waste Stream Results Review; Unit-2 Torus filters Stream; L42157-2; dated April 22, 2010
- RP-AA-605; "Sample" Waste Stream Results Review; Dry Active Waste 2010; L42157-1; dated April 22, 2010
- DW-11-001; Radioactive Material, Low Specific Activity (LSA-II), 7, UN-3321; Fissile Excepted; Units 2/3 Concentrated Waste Shipment Cask to Bear Creek, Oakridge; dated January 13, 2011
- DW-11-012; Radioactive Material, Low Specific Activity (LSA-II), 7, UN-3321; 2/3 DAW; Metal Box to Duratek, Kingston; dated February 16, 2011
- DW-11-020; Radioactive Material, Low Specific Activity (LSA-II), 7, UN-3321; Laundry to UniTech, Morris, Illinois; dated February 15, 2011
- DW-11-011; Radioactive Material, Low Specific Activity (LSA-II), 7, UN-3321; Metal Cask Containing Condensate Resin to Clive Utah; dated February 14, 2011
- DW-10-014; Radioactive Material, Low Specific Activity (LSA-II), 7, UN-3321; RQ-Radionuclides; Metal Cask Containing Condensate Resin to Clive Utah ; dated March 08, 2010
- DW-09-030; Radioactive Material, Low Specific Activity (LSA-II), 7, UN-3321; DAW to Duratek Bear Creek, Oak Ridge; dated August 04, 2009

- DM-09-076; Radioactive Material, Type A Package, 7, UN2915 (non-special form, non fissile or fissile excepted); Part-61 Samples to Teledyne Brown of Knoxville, TN; dated October 13, 2009
- DW-10-022; Radioactive Material, Low Specific Activity (LSA-II), 7, UN-3321; Fissile Excepted of DAW Metal Box to Duratek Services at Bear Creek, Oak ridge, TN; dated May 08, 2010
- DW-10-019; Radioactive Material, Type B(U) Package, 7, UN-2916; Fissile Excepted of Dewatered Filters to Clive Utah; dated April 16, 2010
- Docket No. 71-9168; Certificate of Compliance (CoC) No. 9168 for Model No. CNS 8-120B Cask; USA/9168/B(U); Revision No. 17
- IR-01159083; a Follow-up to IR01159083; Dresden Received an NOV Letter from State of Utah due to Missing Unique Package Identification Number; dated January 04, 2011
- DW-10-058; Radioactive Material, Type A Package, 7, UN 2915; Metal cask containing Impeller shipped as Yellow-II to Clive Utah Disposal Site; dated December 07, 2010
- IR-00882662; Package Arriving through Warehouse Exceeds Dose Limit; dated February 19, 2009
- IR-00919723; Vendor Identifies Error in Report of Part 61 Analysis; dated April 06, 2009
- IR-01024746; Insufficient Detail on Radioactive Shipping Log; dated January 29, 2010
- IR-01030469; Enhancement Needed to More Precisely Classify Waste; dated February 15, 2010
- IR-01152751; Management Direction Conflicts with Procedure and Operating Manual; dated December 15, 2010

4OA2 Identification and Resolution of Problems (71152)

- DTP 47, “Leak Rate Testing Program,” Revision 18
- DOS 7000-01, “Local Leak Rate Testing of Main Steam Isolation Valves (Dry Test),” Revision 6
- DOS 7000-08, “Local Leak Rate Testing of Primary Containment Isolation Valves,” Revision 9
- ER-AA-380, “Primary Containment Leak Rate Testing Program,” Revision 7
- IR 1136704, “D3R21 LLRT 3-0205-27 Exceeded Admin Alarm Limit”
- IR 1135334, “D3R21 – LLRT on 3-1201-1 Exceeded Admin Alarm Limit < 20 SCFH”
- IR 987852, “D2R21 As Found LLRT on 2-0203-1D Exceeded Leakage Limit”
- A/R 171393-28, Chronic Problem #91 – MSIV Leak Rate Test Failures
- IR 987850, “D2R21 As Found LLRT on 2-0203-2C Exceeded Leakage Limit”
- IR 691510, “D2R20 2-0203-2B As Found LLRT Exceeds TS Limit <34 SCFH”
- IR 1135779, “D3R21 FW 220-58B LLRT Exceeded Admin Alarm Limit”
- IR 693270-08, “D2R20 FW 0220-58B Failed As Found LLRT”
- WO 977606, “D3 30M/RFL TS LLRT MSIV 203-1A & 203-2A Dry Test”
- NEI 94-01, “Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,” Revision 1 (12/08/05)
- IR 1133829, “LLRT on 3-0203-1C Leakage Exceeded 34 SCFH”
- IR 1133832, “LLRT 3-0203-1D Leakage Exceeded 34 SCFH”
- IR 1133833, “LLRT 3-0203-2D Leakage Exceeded 34 SDFH Limit”
- IR 1222832, “Unit 3 EDG Jacket Water Temperatures Trend”
- IR 1223961, “Instrumentation Needed to Monitor Unit 3 EDG Cooling”
- IR 1228228, “U3 EDG Test Data Indicated a Potential Hx Problem”
- EC 380926, “Unit 3 EDG Operability Evaluation 10-006,” Revision 000
- EC 384722, “Unit 3 EDG Operability Evaluation 11-004,” Revision 000
- WO 1432882, “D3 1M TS Unit Diesel Generator Operability”
- WO 1441645, “D3 Bwk PM Run D/G Cooling Water Pump”

- DOS 6600-01, “Diesel Generator Surveillance Tests,” Revision 117
- LS-AA-120, “Issue Identification and Screening Process,” Revision 12

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

- DTP 47, “Leak Rate Testing Program,” Revision 18
- DOS 7000-01, “Local Leak Rate Testing of Main Steam Isolation Valves (Dry Test),” Revision 6
- DOS 7000-08, “Local Leak Rate Testing of Primary Containment Isolation Valves,” Revision 9
- ER-AA-380, “Primary Containment Leak Rate Testing Program,” Revision 7
- IR 1133833, “Apparent Cause of MSIV Failures to Meet LLRT Allowable Limits during D3R21”
- IR 1163326, “Unit Three RBM / LPRM Anomaly (U3)”
- IR 1165508, “Followup to IR 1163326 for RBM Anomaly”
- IR 1183073, “Ops Review for RBM Historical Operability Determination”
  IAR 1189950, “LER Identifies Potential Knowledge Deficiency of the RBM Ckt”
- 12E-3473V, “Power Range Neutron Monitoring System RBM Input Relay Selection,” Revision A
- 12E-3473X, “Power Range Neutron Monitoring System RBM Input Signal Selection,” Revision A
- 12E-3473Y, “Power Range Neutron Monitoring System RBM Input Signal Selection,” Revision A 12E-3473X, “Power Range Neutron Monitoring System RBM7 (AR5),” Revision A
- 12E-3473AA, “Power Range Neutron Monitoring System RBM7 (AR5),” Revision A
- 12E-3473AB, “Power Range Neutron Monitoring System RBM8 (AR6),” Revision A
- 12E-3473AC, “Power Range Neutron Monitoring System RBM8 (AR6),” Revision A

4OA5 Other Activities

- DRE00-0075, “Evaluation of Dresden EQ Binders for Extended Power Uprate Environmental Condition,” Revision 2

4OA5 Temporary Instruction 2515/179

- IR-01227156; Enhancement Noted for Clarification of Source ID during Verification of Nationally Tracked Sources; June 10, 2011
- RP-AA-800; Source Leak Test Record of J.L. Shepherd and Associates Model 6810; February 2, 2011
- National Source Tracking System Database for Dresden Nuclear Power Stations; June 1, 2011

## LIST OF ACRONYMS USED

| | |
|---|---|
| AC | Alternating Current |
| ADAMS | Agencywide Document Access Management System |
| AOP | Abnormal Operating Procedure |
| CAP | Corrective Action Program |
| CCSW | Component Cooling Service Water |
| CFR | Code of Federal Regulations |
| DFP | Diesel Fire Pump |
| DGCW | Diesel Generator Cooling Water |
| DRP | Division of Reactor Projects |
| EACE | Equipment Apparent Cause Evaluation |
| EDG | Emergency Diesel Generator |
| EGM | Enforcement Guidance Memorandum |
| FME | Foreign Material Exclusion |
| FWLC | Feedwater Level Control |
| FWRV | Feedwater Regulating Valve |
| gpm | Gallons Per Minute |
| HPCI | High Pressure Coolant Injection |
| HVAC | Heating, Ventilation, Air Conditioning |
| IMC | Inspection Manual Chapter |
| INPO | Institute of Nuclear Power Operations |
| IP | Inspection Procedure |
| IR | Inspection Report |
| IR | Issue Report |
| IST | In-Service Testing |
| kw | Kilowatt |
| LER | Licensee Event Report |
| LLRT | Local Leak Rate Testing |
| LOCA | Loss-of-Coolant Accident |
| LOOP | Loss of Offsite Power |
| LPCI | Low Pressure Coolant Injection |
| LPRM | Local Power Range Monitor |
| MSIV | Main Steam Isolation Valve |
| MSPI | Mitigating Systems Performance Index |
| NCV | Non-Cited Violation |
| NEI | Nuclear Energy Institute |
| NRC | U.S. Nuclear Regulatory Commission |
| NRR | Nuclear Reactor Regulation |
| NSO | Nuclear Station Operator |
| OLR | On Line Risk |
| PARS | Publicly Available Records System |
| PI | Performance Indicator |
| PM | Planned, Post or Preventative Maintenance |
| PMF | Probable Maximum Flood |
| psig | Pounds Per Square Inch Gauge |
| QATR | Quality Assurance Topical Report |
| RBM | Rod Block Monitor |
| RCS | Reactor Coolant System |
| RG | Regulatory Guide |
| RIS | Regulatory Issue Summary |

| | |
|---|---|
| RPV | Reactor Pressure Vessel |
| RWE | Rod Withdrawal Error |
| SBLC | Standby Liquid Control |
| scfh | Standard Cubic Feet Per Hour |
| SDP | Significance Determination Process |
| SER | Safety Evaluation Report |
| SSC | Structure, System, and Component |
| SSD | Safe Shutdown |
| SSE | Safe Shutdown Earthquake |
| TBCCW | Turbine Building Closed Cooling Water |
| TI | Temporary Instruction |
| TS | Technical Specification |
| TSO | Transmission System Operator |
| UFSAR | Updated Final Safety Analysis Report |
| UHS | Ultimate Heat Sink |
| URI | Unresolved Item |
| WO | Work Order |
| ZPA | Zero Period Acceleration |

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Sincerely,

***/RA/***

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

Docket Nos. 50-237; 50-249
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Letter to M. Pacilio from M. Ring dated August 4, 2011

SUBJECT: DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3, INTEGRATED INSPSECTION REPORT 05000237/2011-003; 05000249/2011-003

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