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

May 14, 2009

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

SUBJECT: DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3 INTEGRATED INSPECTION REPORT 05000237/2009-002; 05000249/2009-002

Dear Mr. Pardee:

On March 31, 2009, the U.S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your Dresden Nuclear Power Station, Units 2 and 3. The enclosed report documents the inspection results, which were discussed on April 15, 2009, with Mr. T. Hanley 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.

The report documents three NRC-identified findings and three self-revealing findings of very low safety significance (Green). Five of these findings were determined to involve violations of NRC requirements. Additionally, three licensee-identified violations which were determined to be of very low safety significance are listed in this report. However, because of the very low safety significance and because they are entered into your corrective action program, the NRC is treating these findings as non-cited violations (NCVs) consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest any NCV, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region III; 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352, the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at Dresden. In addition, if you disagree with the characterization of 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 Dresden. The information you provide will be considered in accordance with Inspection Manual Chapter 0305.

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 made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

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

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

Enclosure: Inspection Report 05000237/2009-002; 05000249/2009-002
w/Attachment: Supplemental Information

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

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Plant Manager - Dresden Nuclear Power Station
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Senior Vice President - Midwest Operations
Senior Vice President - Operations Support
Vice President - Licensing and Regulatory Affairs
Director - Licensing and Regulatory Affairs
Manager Licensing - Clinton, Dresden, and Quad Cities
Associate General Counsel
Document Control Desk - Licensing
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J. Klinger, State Liaison Officer,
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Letter to C. Pardee from M. Ring dated May 14, 2009

SUBJECT: DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3
INTEGRATED INSPECTION REPORT 05000237/2009-002;
05000249/2009-002

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REGION III

Docket Nos:	50-237; 50-249
License Nos:	DPR-19; DPR-25
Report No:	05000237/2009-002; 05000249/2009-002
Licensee:	Exelon Generation Company
Facility:	Dresden Nuclear Power Station, Units 2 and 3
Location:	Morris, IL
Dates:	January 1 through March 31, 2009
Inspectors:	C. Phillips, Senior Resident Inspector D. Meléndez-Colón, Resident Inspector W. Slawinski, Senior Health Physicist J. Draper, Reactor Engineer E. Coffman, Reactor Engineer
Approved by:	M. Ring, Chief Projects Branch 1 Division of Reactor Projects

Enclosure

TABLE OF CONTENTS

SUMMARY OF FINDINGS 1

REPORT DETAILS 5

Summary of Plant Status 5

1. REACTOR SAFETY 5
- 1R01 Adverse Weather Protection (71111.01) 5
- 1R04 Equipment Alignment (71111.04) 5
- 1R05 Fire Protection (71111.05) 7
- 1R06 Flooding (71111.06) 9
- 1R11 Licensed Operator Requalification Program (71111.11Q) 12
- 1R12 Maintenance Effectiveness (71111.12) 12
- 1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13) 13
- 1R15 Operability Evaluations (71111.15) 14
- 1R18 Plant Modifications (71111.18) 15
- 1R19 Post-Maintenance Testing (71111.19) 15
- 1R22 Surveillance Testing (71111.22) 16
- 1EP6 Drill Evaluation (71114.06) 18

2. RADIATION SAFETY 18
- 2PS1 Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems (71122.01) 18
- 2PS2 Radioactive Material Processing and Transportation (71122.02) 19

4. OTHER ACTIVITIES 22
- 4OA1 Performance Indicator Verification (71151) 22
- 4OA2 Identification and Resolution of Problems (71152) 23
- 4OA3 Event Follow-up (71153) 30
- 4OA6 Management Meetings 34
- 4OA7 Licensee-Identified Violations 34

SUPPLEMENTAL INFORMATION 1

Key Points of Contact 1

List of Items Opened, Closed, and Discussed 2

List of Documents Reviewed 3

List of Acronyms Used 7

SUMMARY OF FINDINGS

IR 05000237/2009-002, 05000249/2009-002; 01/01/2009 - 03/31/2009; Dresden Nuclear Power Station, Units 2 & 3; Fire Protection, Flood Protection, Identification and Resolution of Problems, 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. Six Green findings were identified by the inspectors. Five of the findings were considered Non-Cited Violations (NCVs) of NRC regulations. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealed Findings

Cornerstone: Initiating Events

- Green. The inspectors identified a NCV of 10 CFR 50, Appendix B, Criterion III, "Design Control", of very low safety significance, for the failure to ensure that the control of the design basis was correctly translated into station procedures. The procedures used to control the temporary placement of 480V heaters in safety-related areas did not meet the station procedural requirements for a temporary configuration change. The violation was placed into the licensee's corrective action program (CAP) in Issue Report (IR) 876126. The licensee's corrective actions included planning to change all the station procedures that control the installation and removal of temporary heaters.

  Using the guidance contained in IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," dated December 4, 2008, the inspectors determined that the finding was more than minor because, if left uncorrected, the performance deficiency had the potential to lead to a more significant safety concern. The inspectors evaluated the finding using IMC 0609, Appendix M, "Significance Determination Process Using Qualitative Criteria," dated December 22, 2006. Per IMC 0609, Appendix M, a bounding quantitative and/or qualitative (i.e., worst case analysis) was performed. The resultant risk significance of the inspection finding was determined to be of very low safety significance and is determined to be Green. The inspectors determined that this issue also affected the cross-cutting area of Problem Identification and Resolution because the licensee failed to take corrective actions to address a safety issue in a timely manner, (P.1(d)). (Section 4OA2.3)

Cornerstone: Mitigating Systems

- Green. The inspectors identified an NCV of the Dresden Nuclear Power Station Renewed Facility Operating License having very low safety significance for the licensee's failure to develop a pre-fire plan for Fire Zone 18.6. This issue was entered into the licensee's CAP as issue reports 873977 and 875688. The licensee's corrective actions included the development of a pre-fire plan for Fire Zone 18.6.

The finding was more than minor because it involved the Mitigating Systems attribute of protection against external factors (i.e., fire), where the failure to develop a pre-fire plan for Fire Zone 18.6 could have adversely impacted the fire brigade's ability to fight a fire. The inspectors completed a Phase 1 significance determination of this issue using IMC 0609, "Significance Determination Process," Appendix A, Attachment 0609.04. However, as discussed by Attachment 0609.04, issues related to performance of the fire brigade are not included in IMC 0609, Appendix F, "Fire Protection SDP," and require management review. The finding was reviewed by NRC management, and was determined to be a finding of very low safety significance because no safe shutdown equipment was located in this fire zone. The inspectors determined that this issue also affected the cross-cutting area of Problem Identification and Resolution (e.g., corrective action program) because the licensee failed to thoroughly evaluate the problem addressed in NCV 05000237/2008008-02; 05000249/2008008-02, "Failure to Develop a Pre-fire Plan for Fire Zone 18.6," such that appropriate corrective actions to address safety issues and adverse trends were not taken in a timely manner, commensurate with their safety significance and complexity, (P.1(d)). (Section 1R05)

- Green. A self-revealed NCV of the Dresden Nuclear Power Station Renewed Facility Operating License having very low safety significance was identified for the licensee's failure to implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Updated Final Safety Analysis Report (UFSAR). Specifically, the licensee failed to ensure that the floor penetrations to Fire Zone 2.0 were sealed as described in the Fire Hazards Analysis. Licensee corrective actions included revising the Fire Hazard Analysis and sealing the floor penetrations.

  The finding was more than minor because it involved the Mitigating Systems attribute of protection against external factors (i.e., flood hazard, fire) and impacted the Mitigating Systems objective to ensure availability, reliability, and capability of systems that respond to initiating events (i.e., flood hazard, fire) to prevent undesirable consequences. The inspectors performed a Phase 1 qualitative screening and the finding screened to very low safety significance. The inspectors determined that because the modifications took place in the 1985 to 1986 timeframe, the performance deficiency is not reflective of current licensee performance and therefore no cross-cutting area was affected. (Section 1R06)

Cornerstone: Barrier Integrity

- Green. The inspectors identified an NCV of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action", for the failure to correct degraded safety-related equipment in a timely manner. A degraded 4-way solenoid valve for the reactor building ventilation damper 2-5741B actuator was not replaced during the work window that started on January 5, 2009. The solenoid valve failed on January 13, 2009, when it was called upon during a reactor building ventilation isolation. The violation was placed into the licensee's corrective action program in IR 877591. The licensee's corrective action included replacing all the 4-way solenoid valves in the actuators for all the Unit 2 and Unit 3 reactor building ventilation secondary containment isolation boundary dampers.

  Using the guidance contained in IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," dated December 4, 2008, the inspectors determined that the finding was more than minor because it was associated with the Reactor Safety Barrier Integrity Cornerstone objective of maintaining the functionality of the secondary containment. The inspectors evaluated the finding using IMC 0609, Attachment 0609.04, dated January 10, 2008. Per Table 4a, under Containment Barrier, question 1, “Does the finding only represent a degradation of the radiological barrier function provided for the … Standby Gas Treatment System,” the inspectors answered, YES. The secondary containment isolation valves isolate the secondary containment to ensure the effectiveness of the Standby Gas Treatment System. Therefore the finding was determined to be Green. The inspectors determined that this issue also affected the cross-cutting area of Problem Identification and Resolution, (P.1(c)). (Section 4OA2.4)

- Green. On January 13, 2009, a Finding with no violation of regulatory requirements was self-revealed when an operator performed an incorrect response to an unexpected alarm in the control room that resulted in a reactor building ventilation isolation and a standby gas treatment system actuation. This action required entry into TS 3.6.4.1 Limiting Condition of Operation, Action A for reactor building low differential pressure.

  The finding was more than minor because it impacted the structures, systems, and components attribute of the Barrier Integrity Cornerstone objective. The finding was of very low safety significance because it impacted the reactor building differential pressure for a time period of less than one hour. The finding was placed into the licensee’s CAP as IR 866445. As an immediate corrective action, the individual was temporarily removed from licensed shift duties and no manipulation of any equipment in the plant or the control room was allowed without a peer check until January 18, 2009. The inspectors also concluded that this finding affected the cross-cutting issue of Human Performance (Personnel) because the operator failed to utilize human performance error prevention techniques, (H.4(a)). (Section 4OA3.1)

- Green. A self-revealed NCV of Dresden Station Improved Technical Specification (TS) 3.6.1.3, “Primary Containment Isolation Valves (PCIVs),“ of very low safety significance was identified for the failure to declare primary containment isolation valve 3-3702 inoperable and take actions in accordance with the requirements of TS 3.6.1.3 required action A. The licensee generated IR 837675 and IR 839009 to address this issue. Corrective actions included: the initiation of a training request to re-enforce with Operations personnel the potential operability issues when light indications are not functioning properly, and the revision of Operations procedures to include guidance to alert users that a failed or flickering indication light associated with a motor operated valve may indicate problems that could affect valve operation, and that valve operability must be verified.

  The finding was more than minor because it impacted the Barrier Integrity objective to provide reasonable assurance that physical design barriers (i.e., containment) protect the public from radionuclide releases caused by accidents or events. The inspectors completed a Phase 1 significance determination of this issue using IMC 0609, Attachment 0609.04, “Phase 1 – Initial Screening and Characterization of Findings,” The inspectors answered NO to all questions in the Containment Barrier column of Table 4a, therefore the finding screened as Green (i.e., very low safety significance). The inspectors determined that this finding also affected the cross-cutting area of Human Performance, resources aspect (H.2(c)) because the licensee failed to provide complete, accurate and up-to-date procedures. (Section 4OA3.3)

B. Licensee-Identified Violations

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

REPORT DETAILS

Summary of Plant Status

Unit 2

On March 7, 2009, power was reduced to 66 percent to perform rod sequence exchange and control rod drive (CRD) scram timing, and the unit returned to full power on the same day.

On March 28, 2009, power was reduced to 78 percent power to perform channel distortion testing. The unit returned to full power on the same day.

Unit 3

On February 28, 2009, Unit 3 was down powered to perform a drywell entry to access leakage of the 3C electromagnetic relief valve (ERV). The unit returned to full power on March 1, 2009.

1. REACTOR SAFETY

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 event of the failure of the Dresden Island Lock and Dam. 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 a loss of heat sink caused by failure of the lock and dam. Additionally, the inspectors performed a walkdown of the protected area to identify any modification to the site which would inhibit the use of equipment designed to mitigate the loss of the external heat sink. The inspectors also reviewed the abnormal operating procedure (AOP) for mitigating the loss of heat sink due to a lock and dam failure, 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

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Quarterly Partial System Walkdowns

a. Inspection Scope

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

- Unit 2 DIV 1 low pressure coolant injection/containment cooling service water systems;
- Unit 2 emergency diesel generator (EDG);
- Unit 2/3 EDG;
- Unit 2 EDG cooling water system

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

.2 Semi-Annual Complete System Walkdown

a. Inspection Scope

On March 11, 2009, the inspectors performed a complete system alignment inspection of the Unit 2 high pressure coolant injection system to verify the functional capability of the system. This system was selected because it was considered both safety-significant and risk-significant in the licensee's probabilistic risk assessment. The inspectors walked down the system to review mechanical and electrical equipment 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 work orders (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.

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

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Routine Resident Inspector Tours (71111.05Q)

a. Inspection Scope

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

- Fire Zone 1.1.2.5.D, elevation 589', Unit 2 standby liquid control;
- Fire Zone 1.1.2.5.C, elevation 545', Unit 2 isolation condenser pipe chase;
- Fire Zone 11.2.1, elevation 476', Unit 2 southwest corner;
- Fire Zone 18.6, elevation 541', station black out (SBO) Battery Room (U2 125VDC Alt. Battery Room); and
- Fire Zone 7.0.A.1, elevation 549', Unit 2 battery room.

The inspectors reviewed areas to assess if the licensee had implemented a fire protection program that adequately controlled combustibles and ignition sources within the plant, effectively maintained fire detection and suppression capability, maintained passive fire protection features in good material condition, and had implemented adequate compensatory measures for out of service, degraded or inoperable fire protection equipment, systems, or features in accordance with the licensee's fire protection 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 five quarterly fire protection inspection samples as defined in IP 71111.05-05.

b. Findings

Introduction: The inspectors identified a non-cited violation (NCV) of the Dresden Nuclear Power Station Renewed Facility Operating License having very low safety significance (Green) for the licensee's failure to develop a pre-fire plan for Fire Zone 18.6.

Description: On January 28, 2009, the inspectors performed a fire protection inspection of Fire Zone 18.6. As shown in the Dresden Station Units 2 and 3 Fire Protection

Reports, Volume 1, “Updated Fire Hazards Analysis,” Figure 3.3-26, Fire Zone 18.6 encompasses the Unit 2 125 volt (V) alternate batteries. Alternate batteries are provided in order to allow the unit’s main 125V batteries to undergo rated discharge testing while the unit remains at power. The alternate batteries are also available to supply system loads upon a failure of the unit’s main 125V batteries. The alternate batteries are of a similar type as the unit’s main batteries and have been sized to support the same loads. The alternate batteries are normally disconnected from the system and are kept on a float charge.

The inspectors could not locate a pre-fire plan associated with this fire area. The inspectors identified that the existing plan for Fire Zone 18.6 was referencing the SBO diesel generator battery room, specifically the 6A 125V battery room. These batteries provide the necessary power for the SBO diesel generator control and indications. The 6A 125V batteries are nonsafety-related.

As described in the Enforcement section, fire plans shall be developed for all safety-related areas and areas representing a hazard to safety-related equipment. The Unit 2 125V alternate batteries are classified as safety-related equipment; therefore, Fire Zone 18.6 should have had a pre-fire plan. The licensee generated issue reports (IR) 873977 and 875688 to address this issue.

The inspectors had identified this issue previously as NCV 05000237/2008008-02; 05000249/2008008-02, “Failure to Develop a Pre-fire Plan for Fire Zone 18.6.”

Analysis: The inspectors determined that the failure to develop a pre-fire plan for Fire Zone 18.6 was a performance deficiency warranting a significance evaluation. Using Inspection Manual Chapter (IMC) 0612, Appendix B, “Issue Screening,” issued on December 04, 2008, the inspectors determined that this finding was more than minor because it involved the Mitigating Systems attribute of protection against external factors (i.e., fire), where the failure to develop a pre-fire plan for Fire Zone 18.6 could have adversely impacted the fire brigade’s ability to fight a fire. As such, this finding impacted the Mitigating Systems objective to ensure availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Although Fire Zone 18.6 did not have a pre-fire plan associated with it, no safe shutdown equipment was located in this fire zone. The inspectors determined that this issue also affected the cross-cutting area of Problem Identification and Resolution (e.g., corrective action program) because the licensee failed to thoroughly evaluate the problem addressed in NCV 05000237/2008008-02; 05000249/2008008-02, “Failure to Develop a Pre-fire Plan for Fire Zone 18.6,” such that appropriate corrective actions to address safety issues and adverse trends were not taken in a timely manner, commensurate with their safety significance and complexity, P.1(d)

The inspectors completed a Phase 1 significance determination of this issue using IMC 0609, “Significance Determination Process,” Appendix A, Attachment 0609.04, dated January 10, 2008. The inspectors determined that the finding affected fire protection defense-in-depth strategies. However, as discussed by IMC 0609, Appendix A, Attachment 0609.04, issues related to performance of the fire brigade are not included in IMC 0609, Appendix F, “Fire Protection SDP,” and require management review. Therefore, the finding was reviewed by NRC management, and was determined to be a finding of very low safety significance (Green) because no safe shutdown equipment was located in this fire zone.

Enforcement: The inspectors determined that the licensee's failure to develop a pre-fire plan for Fire Zone 18.6 was a violation of Dresden Nuclear Power Station Renewed Operating License. License conditions 2.E and 3.G of the Unit 2 and Unit 3, respectively, Dresden Nuclear Power Station Renewed Facility Operating Licenses states, in part, that "The licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report for the facility...." Section 9.5.1, "Fire Protection System," of Dresden UFSAR states that "The design bases, system descriptions, safety evaluations, inspection and testing requirements, NFPA [National Fire Protection Association] conformance reviews, personnel qualifications, and training are described in Reference 1."

Section 9.5.10, "References," of Dresden UFSAR, reference 1, lists "Dresden Units 2 and 3 Fire Protection Reports," Volumes 1 through 5, and "Fire Protection Program Documentation Package," Volumes 1 through 13, as the documents to follow for compliance with the fire protection program.

Section 2.5.4, "Fire Fighting Strategies," of Dresden Station Units 2 and 3 Fire Protection Reports, Volume 1, "Updated Fire Hazards Analysis," specifies that "Pre-fire plans are provided for all safety-related areas of the plant." In addition, Dresden Station Units 2 and 3 Fire Protection Program Documentation Package, Volume 12, "References," also specifies "Pre-fire plans have been developed for the safety-related areas in the plant."

Also, in procedure OP-AA-201-008, Revision 2, "Pre-Fire Plans," Paragraph 4.1.1, "Main Body," the licensee stated, "A pre-fire plan shall be established for all safety-related areas, areas representing a hazard to safety-related equipment, and insured buildings."

Contrary to the above, the licensee failed to develop a pre-fire plan for Fire Zone 18.6. This failure could have adversely impacted the fire brigade's ability to fight a fire. This issue was entered into the licensee's corrective action program as issue reports 873977 and 875688. Corrective actions by the licensee included the development of a pre-fire plan for Fire Zone 18.6. 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 a NCV, consistent with Section VI.A.1 or the NRC Enforcement Policy.
(NCV 05000237/2009002-01; 05000249/2009002-01).

1R06 Flooding (71111.06)

.1 Internal Flooding

a. Inspection Scope

The inspectors reviewed selected risk important plant design features and licensee procedures intended to protect the plant and its safety-related equipment from internal flooding events. The inspectors reviewed flood analyses and design documents, including the UFSAR, engineering calculations, and abnormal operating procedures to identify licensee commitments. In addition, the inspectors reviewed licensee drawings to identify areas and equipment that may be affected by internal flooding caused by the failure or misalignment of nearby sources of water, such as the fire suppression or the circulating water systems. The inspectors also reviewed the licensee's corrective action documents with respect to past flood-related items identified in the corrective action program to verify the adequacy of the corrective actions. The inspectors performed a walkdown of the following plant area to assess the adequacy of drains, and that the licensee complied with its commitments:

- Unit 2 battery room.

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

b. Findings

Introduction: A self-revealed NCV of the Dresden Nuclear Power Station Renewed Facility Operating License having very low safety significance (Green) was identified for the licensee's failure to implement and maintain in effect all provisions of the approved Fire Protection Program as described in the UFSAR.

Description: On January 16, 2009, operations personnel in the Main Control Room discovered water dripping at the rear of the 902-4 and 902-5 main control boards onto various components of the Unit 2 main control panels (horseshoe area). The source of the water was the Unit 2 east turbine building heating coil, located in the Unit 2 battery room, directly above the Main Control Room. The water was leaking around floor mounting bolts for the Unit 2 and Unit 2A 125 volt DC battery chargers. Two nonsafety-related components failed as a result of the water leak (the Unit 2 source range monitor level recorder (RR 2-750-2) and the Unit 2 computer digital display unit (2-0942-8A)). Main control boards 902-4 and 902-5 are classified as safe shutdown equipment under Fire Zone 2.0, Main Control Room. Leakage into the Main Control Room stopped after approximately 61 minutes. The licensee generated IR 867770 describing this event.

The Unit 2 125 volt DC battery chargers were installed under modification M12-2-85-32 in the 1985 to 1986 timeframe. The bolts holding the charger supports in place were routed through the battery room area floor. This is the ceiling to the Main Control Room and a rated 3-hour fire barrier. Further investigation determined no previous documentation existed that evaluated the as-built configuration as an acceptable 3-hour fire barrier when it was installed. As a result, the licensee declared the fire barrier inoperable based on the lack of evaluation. The licensee generated IR 892096 to address this issue. As part of the extent of condition, at least three more penetrations were identified as not being shown on the associated fire barrier drawings (IR 896456). Also, the Fire Hazards Analysis (FHA) described all penetrations to Fire Zone 2.0 (Main Control Room) as being sealed to prevent water from being released into this area.

Analysis: The inspectors determined that the failure to ensure that the floor penetrations to Fire Zone 2.0 were sealed, as described in the Fire Hazards Analysis, was a performance deficiency warranting a significance evaluation. Using IMC 0612, Appendix B, "Issue Screening," issued on December 4, 2008, the inspectors determined that this finding was more than minor because it involved the Mitigating Systems attribute of protection against external factors (i.e., flood hazard, fire). As such, this finding impacted the Mitigating Systems objective to ensure availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The licensee performed an evaluation (EC 374653) to determine if the configuration described above (bolts that extend through a fire barrier) was an acceptable 3-hour fire barrier and concluded that the configuration was acceptable. Although the penetrations were determined to be an acceptable 3-hour fire barrier, water was still able to be released into Fire Zone 2.0. The inspectors determined that because the modifications took place in 1985 and 1986 the performance deficiency is not reflective of current licensee performance and therefore no cross-cutting area was affected.

The inspectors completed a Phase 1 significance determination of this issue using IMC 0609, "Significance Determination Process," Appendix A, Attachment 0609.04, dated January 10, 2008. The inspectors determined that the finding affected the fire protection defense-in-depth strategies. As discussed by IMC 0609, Appendix A, Attachment 0609.04, issues related to fire protection defense-in-depth strategies are to be evaluated using IMC 0609, Appendix F, "Fire Protection Significance Determination Process," dated February 28, 2005. The inspectors performed a Phase 1 qualitative screening, assigned a Finding Category of Fire Confinement, and a Degradation Rating of LOW, because even though the 3-hour fire barrier was degraded, the licensee determined that it was acceptable. Therefore the finding screened as of very low safety significance (Green).

Enforcement: The inspectors determined that the licensee's failure to ensure that the floor penetrations to Fire Zone 2.0 were sealed, as described in the Fire Hazards Analysis, to prevent water from being released into this area was a violation of the Dresden Nuclear Power Station Renewed Operating License. License conditions 2.E and 3.G of the Unit 2 and Unit 3 Dresden Nuclear Power Station Renewed Facility Operating Licenses state, in part, "The licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report (UFSAR) for the facility...." Section 9.5.1, "Fire Protection System," of the Dresden UFSAR states that, "The design bases, system descriptions, safety evaluations, inspection and testing requirements, National Fire Protection Association (NFPA) conformance reviews, personnel qualifications, and training are described in Reference 1."

Section 9.5.10, "References," of Dresden UFSAR, Reference 1, lists, "Dresden Units 2 and 3 Fire Protection Reports," Volumes 1 through 5, and, "Fire Protection Program Documentation Package," Volumes 1 through 13, as the documents to follow for compliance with the fire protection program.

Section 2.3.1.8, "Drainage," specifies, "Suppression effects are described in the Fire Hazards Analysis (Section 4.0) for each fire zone." Section 4.8.1, "Turbine Building – Station Battery Rooms Elevation 549 feet 0 inch (Fire Zone 7.0.A)," specifies that, "water runoff would be controlled through utilization of floor drains in Fire Zone 8.2.7 or in the battery rooms. The floor penetrations are sealed to Fire Zone 2.0 to prevent water from being released into this area." Fire Zone 2.0 describes the Main Control Room.

Contrary to the above, from 1985 through 2009, the licensee failed to properly seal the floor penetrations to Fire Zone 2.0, as described in the Fire Hazards Analysis, to prevent water from being released into this area. Although the licensee determined that the penetrations were acceptable as a 3-hour fire barrier, water was still able to be released into Fire Zone 2.0. No safety-related components failed as a result of the water leak into the Main Control Room. Some of the corrective actions to address this issue included a revision to the Fire Hazards Analysis to describe the as-built configuration (IR 889472) and sealing the penetrations in the Main Control Room ceiling (EC 374656). 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 Section VI.A.1 or the NRC Enforcement Policy. **(NCV 05000237/2009002-02; 05000249/2009002-02)**

1R11 Licensed Operator Requalification Program (71111.11Q)

.1 Resident Inspector Quarterly Review (71111.11Q)

a. Inspection Scope

On February 9, 2009, the inspectors observed a crew of licensed operators in the plant's simulator during licensed operator requalification examinations to verify that operator performance was adequate, evaluators were identifying and documenting crew performance problems and training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

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

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

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

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Routine Quarterly Evaluations

a. Inspection Scope

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

- Unit 2 high pressure coolant injection; and
- Unit 3 feedwater.

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

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

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

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

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

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

- Unit 2 DIV 2 low pressure coolant injection/containment cooling service water is out-of-service for maintenance;
- Unit 3 EDG is out-of-service for maintenance;
- Unit 2 EDG is out-of-service for maintenance;
- Safety-related service water pump Bay 13 cleaning – 2/3 diesel fire pump out-of-service; and
- Switchyard breaker 8-15 failure.

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

These activities constituted five samples as defined in IP 71111.13-05.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the following issues:

- IR 667672, "TR 3-5741-19 Point 10 TC Open. No Tech Spec Violation;"
- IR 872894, "Unknown Substance Found in Bay 13;"
- Event Notification #44849 "Configuration of CST Level Instrumentation, Discovered at Hatch 2;" and
- IR 886454, "Low Pressure Coolant Injection CST Suction Isolation Valve 2-1501-31B Seat is Leaking."

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

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

b. Findings

No findings of significance were identified.

1R18 Plant Modifications (71111.18)

.1 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the following temporary modification:

- Temporary Heater Placement In Station Blackout Diesel Building

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

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

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

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

- WO 01004735-01, "2C CCSW (containment cooling service water) pump packing requires replacement;"
- WO 01148961-03, "Replace U3 EDG cooling water HX (heat exchanger) Flex Hoses;"
- WO 778501-01, "D2 4Y Preventative Maintenance (PM) Replace Ball Checks, Inspect Air Hose, and Replace/Rebuild VERSA Valve;"
- WO 1141992-02, "Replace Identified Degraded CCSW Piping;" and
- Unit 2 EDG outage, Review of Operating Experience Smart Sample: OpESS FY2008-01, "Negative Trend and Recurring Events Involving Emergency Diesel Generators."

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

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

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

.1 Surveillance Testing

a. Inspection Scope

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

- WO 01175809-01, "D3 Qtr TS CS Pump Test with Torus Available for IST (in-service testing) Data Surveillance";
- WO 999967-01, "D2 2Y TS CS Pump Comp Test with Torus Available, for IST Surveillance;"
- DOP 2000-24, "Drywell Sump Operation," Revision 18 (RCS Unit 2);
- WO 01175812-01, "D3 Qtr TS Core Spray Motor Operator Valve Operability and Timing Surveillance;"
- WO 01011536-02, Unit 3 Reactor Feedwater Flow Transmitter/Indicator/ Recorder Calibration;" and
- U3 Down power and Drywell entry to Perform Thermography on 3C ERV (emergency relief valve).

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

Documents reviewed are listed in the Attachment to this report.

This inspection constituted six surveillance testing samples, one inservice testing sample, one RCS leakage detection sample, and four routine samples as defined in IP 71111.22, Sections -02 and -05.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

.1 Emergency Preparedness Drill Observation

a. Inspection Scope

The inspectors evaluated the conduct of a routine licensee emergency drill on February 11, 2009, to identify any weaknesses and deficiencies in classification, notification, and protective action recommendation development activities. The inspectors observed emergency response operations in the simulator to determine whether the event classification, notifications, and protective action recommendations were performed in accordance with procedures. The inspectors also attended the licensee drill critique to compare any inspector-observed weakness with those identified by the licensee staff in order to evaluate the critique and to verify whether the licensee staff was properly identifying weaknesses and entering them into the corrective action program. As part of the inspection, the inspectors reviewed the drill package and other documents listed in the Attachment to this report.

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

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Public Radiation Safety

2PS1 Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems (71122.01)

.1 Inspection Planning

a. Inspection Scope

In 2008, the NRC performed confirmatory measurements of water samples to evaluate the licensee's proficiency in collecting and analyzing samples for the presence of tritium. Specifically, in November 2008, the NRC split samples with the licensee that were collected from 12 different onsite monitoring wells and sent those samples to the NRC's contract laboratory for independent tritium analyses. The samples were collected from onsite monitoring wells which were part of the licensee's Radiological Groundwater Protection Program and from other supplemental onsite monitoring wells established by the licensee. The inspectors confirmed, through these confirmatory measurements, that the licensee demonstrated adequate radiological capabilities necessary to quantify environmental samples and radioactive effluents, as required by 10 CFR 20 and the licensee's Offsite Dose Calculation Manual.

No samples were accredited for this inspection effort.

b. Findings

No findings of significance were identified.

2PS2 Radioactive Material Processing and Transportation (71122.02)

.1 Radioactive Waste System

a. Inspection Scope

The inspectors reviewed the liquid and solid radioactive waste system description in the UFSAR for information on the types and amounts of radioactive waste (radwaste) generated and disposed. The inspectors reviewed the scope of the licensee's audit program with regard to radioactive material processing and transportation programs to verify that it met the requirements of 10 CFR 20.1101(c).

This inspection constituted one sample as defined in IP 71122.02-5.

b. Findings

No findings of significance were identified.

.2 Radioactive Waste System Walkdowns

a. Inspection Scope

The inspectors performed walkdowns of the liquid and solid radwaste processing systems to verify that the systems agreed with the descriptions in the UFSAR and the Process Control Program and to assess the material condition and operability of the systems. The inspectors reviewed the status of radwaste processing equipment that was not operational and/or was abandoned in place. The inspectors reviewed the licensee's administrative and physical controls to ensure that the equipment would not contribute to an unmonitored release path or be a source of unnecessary personnel exposure.

The inspectors reviewed changes to the waste processing system to verify that the changes were reviewed and documented in accordance with 10 CFR 50.59 and to assess the impact of the changes on radiation dose to members of the public. The inspectors reviewed the current processes for transferring waste resin into shipping containers to determine if appropriate waste stream mixing and/or sampling procedures were utilized. The inspectors also reviewed the licensee's methods for waste concentration averaging to determine if representative samples of the waste product were provided for the purposes of waste classification, as required by 10 CFR 61.55.

This inspection constituted one sample as defined in IP 71122.02-5.

b. Findings

No findings of significance were identified.

.3 Waste Characterization and Classification

a. Inspection Scope

The inspectors reviewed the licensee's radiochemical sample analysis results for each of the licensee's waste streams, including dry active waste (DAW), spent resins, concentrator wastes and filter media. The inspectors also reviewed the licensee's use of scaling factors to quantify difficult-to-measure radionuclides (e.g., pure alpha or beta emitting radionuclides). The reviews were conducted to verify that the licensee's program assured compliance with 10 CFR 61.55 and 10 CFR 61.56, as required by Appendix G of 10 CFR Part 20. The inspectors also reviewed the licensee's waste characterization and classification program to ensure that the waste stream composition data accounted for changing operational parameters and thus remained valid between the annual sample analysis updates.

This inspection constituted one sample as defined in IP 71122.02-5.

b. Findings

No findings of significance were identified.

.4 Shipment Preparation and Shipment Manifests

a. Inspection Scope

The inspectors reviewed the documentation of shipment packaging, radiation surveys, package labeling and marking, vehicle inspections and placarding, emergency instructions, determination of waste classification/isotopic identification, and licensee verification of shipment readiness for eight non-excepted material and radwaste shipments made between February 2007 and January 2009. The shipment documentation reviewed consisted of the following shipments consigned as low specific activity (LSA) and Type B:

- Three Irradiated/Contaminated Equipment Shipments to Vendor;
- One Contaminated Equipment Shipment to Waste Processor;
- Two Radwaste Resin/Concentrator Wastes to Waste Burial Site; and
- Two Irradiated Hardware Shipments to Waste Burial Site.

For each shipment, the inspectors determined if the requirements of 10 CFR Parts 20 and 61 and those of the Department of Transportation (DOT) in 49 CFR Parts 170-189 were met. Specifically, records were reviewed and staff involved in shipment activities was interviewed to determine if packages were labeled and marked properly, if package and transport vehicle surveys were performed with appropriate instrumentation, if radiation survey results satisfied DOT requirements, and if the quantity and type of radionuclides in each shipment were determined accurately. The inspectors also determined whether shipment manifests were completed in accordance with DOT and NRC requirements, if they included the required emergency response information, if the recipient was authorized to receive the shipment, and if shipments were tracked as required by 10 CFR Part 20, Appendix G.

This inspection constitutes one sample as defined in IP 71122.02-5.

Radiation protection staff involved in shipment activities were observed and interviewed by the inspectors to determine if they had adequate skills to accomplish shipment related tasks and to determine if the shippers were knowledgeable of the applicable regulations to satisfy package preparation requirements for public transport with respect to NRC Bulletin 79-19, "Packaging of Low-Level Radioactive Waste for Transport and Burial," and 49 CFR Part 172 Subpart H. Also, lesson plans for safety training and function specific training for radiation protection technicians, laborers, and warehouse staff were reviewed for compliance with the hazardous material training requirements of 49 CFR 172.704.

This inspection constitutes one sample as defined in IP 71122.02-5.

b. Findings

No findings of significance were identified.

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed condition reports, audits and self-assessments that addressed radioactive waste and radioactive materials shipping program deficiencies since the last inspection to verify that the licensee had effectively implemented the corrective action program and that problems were identified, characterized, prioritized and corrected. The inspectors reviewed the licensee's progress since the last inspection to address long-standing material condition issues in various areas of the Radwaste Building including vaults and tanks. Additionally, the inspectors reviewed and discussed with the licensee its plans to rectify re-emerging material condition problems in the radwaste building basement. The inspectors also verified that the licensee's self-assessment program was capable of identifying repetitive deficiencies or significant individual deficiencies in problem identification and resolution.

The inspectors reviewed corrective action reports from the radioactive material and shipping programs since the previous inspection, interviewed staff and reviewed documents to determine if the following activities were being conducted in an effective and timely manner commensurate with their importance to safety and risk:

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

This inspection constituted one sample as defined in IP 71122.02-5.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Unplanned Scrams per 7000 Critical Hours

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Scrams per 7000 Critical Hours performance indicator (PI) for Unit 2 and Unit 3 for the period from the first quarter 2008 through the fourth quarter 2008. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the Nuclear Energy Institute (NEI) Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5, were used. The inspectors reviewed the licensee's operator narrative logs, issue reports, event reports and NRC Inspection Reports for the period of January 2008 through January 2009 to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings of significance were identified.

.2 Unplanned Scrams with Complications

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Scrams with Complications performance indicator for Unit 2 and Unit 3 for the period from the first quarter 2008 through the fourth quarter 2008. 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 5, were used. The inspectors reviewed the licensee's operator narrative logs, issue reports, event reports and NRC Inspection Reports for the period of January 2008 through January 2009 to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings of significance were identified.

.3 Unplanned Transients per 7000 Critical Hours

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Transients per 7000 Critical Hours performance indicator for Unit 2 and Unit 3 for the period from the first quarter 2008 through the fourth quarter 2008. 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 5, were used. The inspectors reviewed the licensee's operator narrative logs, issue reports, event reports and NRC Inspection Reports for the period of January 2008 through January 2009 to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Routine Review of items Entered Into the CAP

a. Scope

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

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

b. Findings

No findings of significance were identified.

.2 Daily CAP Reviews

a. Scope

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

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

b. Findings

No findings of significance were identified.

.3 In-depth Review – IR 728641, "NOS (Nuclear Oversite) Identified that a TCCP (Temporary Configuration Change Package) is Needed for the U2 Alternative Battery Heater"

a. Inspection Scope

The inspectors conducted an in-depth review related to problem identification, prioritization and corrective actions associated with IR 728641. This review constituted one sample as defined in IP 71152.

The inspectors reviewed the additional following documents:

IR 728641, "NOS IDS That A TCCP Is Needed For the U2 Alt Battery Heater;"

IR 726400, "DOS 0010-20 U1 and Out Buildings Cold Weather Operations Rev;"

IR 729793, "NOS IDS TCCP Process Not Used For Temp Heaters;"

IR 876126, "NRC Resident Identified Issues;"

CC-AA-112, "Temporary Configuration Changes," Revision 12;

OP-AA-201-006, "Control of Temporary Heat Sources," Revision 4;

DOS 0010-19, "Preparation for Cold Weather Operations for Unit 1 & Out Buildings," Revision 26;

DOS 0010-20, "Cold Weather Operations for Unit 1 & Out Buildings," Revision 11; and

DOS 0010-21, "Securing Cold Weather Operations for U1 & Out Buildings," Revision 12.

b. Findings

Introduction: The inspectors identified a NCV of 10 CFR 50, Appendix B, Criterion III, "Design Control", of very low safety significance (Green) for the failure to ensure the control of the design basis was correctly translated into station procedures. The addition of temporary space heaters is a modification that can adversely impact plant equipment design. A temporary modification that is made repeatedly can be controlled by an operation or maintenance procedure provided the procedure meets certain requirements. The procedures used to control the temporary placement of 480V heaters in safety-related areas did not meet the station procedural requirements for a temporary configuration change.

Description: The licensee identified an issue with the control of temporary heaters in outbuildings on January 28, 2008, in the following corrective action documents:

IR 728641, "NOS IDS That A TCCP Is Needed For The U2 Alt Battery Heater;"

IR 726400, "DOS 0010-20 U1 and Out Buildings Cold Weather Operations Rev;" and

IR 729793, "NOS IDS TCCP Process Not Used For Temp Heaters."

These IRs identified that procedure DOS 0010-20 was used to control the placement of 480V temporary heaters in out buildings, and that DOS 0010-20 did not meet the requirements of CC-AA-112, "Temporary Configuration Changes," Revision 12, Section 4.2, for procedural control of temporary configuration changes.

Procedure CC-AA-112 stated that a temporary configuration change that was performed repeatedly could be controlled by a procedure (i.e., DOS 0010-20) as long as that procedure met certain requirements. An evaluation must have been done at least once, the evaluation had to be referenced in the procedure, and the temporary change needed to be independently verified to have been installed and removed. These verifications had to be documented. In addition, shift management had to concur with the installation and that concurrence had to be documented.

The inspectors identified in January 2009 that the corrective actions the licensee took to address the three above IRs were incomplete. Verification and authorization signatures were placed in DOS 0010-20, but there was no guidance on the time frame between initial installation and verification. The inspectors identified two examples where the time frame between installation and verification were 5 days and 10 days respectively. In addition, there were no justifications for heater placement performed.

The inspectors identified in January 2009 that the licensee was also using procedure DOS 0010-19, "Preparation For Cold Weather Operations For Unit 1 & Out Buildings," Revision 26; to install temporary heaters and DOS 0010-19 also did not meet the requirements of CC-AA-112. The licensee did not identify that this procedure had the same problems as DOS 0010-20 had in January 2008. The licensee was using DOS 0010-19 and DOS 0010-20 to install temporary heaters that could impact safety-related or augmented quality equipment without proper installation verification or technical evaluations.

Safety-related or augmented quality equipment existing in out buildings included the SBO diesel generators, Unit 1 and 2/3 fire pumps, and the isolation condenser makeup pumps.

Analysis: The inspectors determined that the failure to ensure control of the design basis, by adding temporary space heaters, was correctly translated into station procedures was a performance deficiency. Using the guidance contained in IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," dated December 4, 2008, the inspectors determined that the finding was more than minor because if left uncorrected the performance deficiency had the potential to lead to a more significant safety concern.

The inspectors evaluated the finding using IMC 0609, Appendix M, "Significance Determination Process Using Qualitative Criteria," dated December 22, 2006. Per IMC 0609, Appendix M, a bounding quantitative and/or qualitative (i.e., worst case analysis) should be initially performed, using best available information to determine the significance of the issue. If the bounding evaluation shows that the finding is of very low safety significance, the finding is Green. The inspectors assumed losing 1 train of isolation condenser makeup pump or 1 station black out diesel, or 1 diesel fire pump as the worst case scenario. The inspectors performed a Phase 2 Significance Determination assuming no recovery credit and using different initiating event likelihoods. The resultant risk significance of the inspection finding was determined to be of very low safety significance and Green. The inspectors determined that this issue also affected the cross-cutting area of Problem Identification and Resolution because the licensee identified issues with the control of temporary heaters in January 2008 and failed to take effective corrective actions to address a safety issue in a timely manner, P.1(d).

Enforcement: The Code of Federal Regulations, Title 10, Part 50, Appendix B, Criterion III, "Design Control", states, in part, "Measures shall be established to assure that … the design basis, as defined in §50.2 and as specified in the license application, for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions. These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled."

Exelon Generation Company Quality Assurance Topical Report, NO-AA-10, Revision 82, Section 3, "Design Control," Section 2.6, states in part, … modifications to operating facilities, … shall be justified and subject to design control measures commensurate with those applied to the original design.

Procedure CC-AA-112, "Temporary Configuration Changes," Revision 12, Section 4.2, "Procedurally Controlled Temporary Configuration Changes," states in part, that temporary configuration changes may be controlled through procedures that are approved in accordance with this section. Procedurally controlled temporary configuration changes are used to control changes that are performed on a regular basis and would benefit from a more specifically detailed process. Procedures used to control temporary configuration changes must:

- Provide a documented technical evaluation, using CC-AA-309-101, of the procedure that is being used as a procedural control. Documentation of the approved technical evaluation review must be contained in the review/approval package as an applicable procedure reference.
- Include verification per procedural requirements when the temporary change is installed or removed.
- Provide requirements for and obtain operations shift management notification and authorization/sign off when temporary changes are installed and removed.

Contrary to the above, from January 24, 2008, until March 18, 2009, the licensee utilized procedures DOS 0010-19, "Preparation For Cold Weather Operations For Unit 1 & Out Buildings," Revision 26; and DOS 0010-20, "Cold Weather Operations For Unit 1 & Out Buildings," Revisions 10 and 11, to procedurally control the addition of 480V heaters (a temporary configuration change) in outbuildings that had the potential to impact safety-related or augmented quality equipment without:

- A documented technical evaluation, of any kind, for heaters installed under either DOS 0010-19 or 20;
- Verification when the temporary configuration change was installed under DOS 0010-19 or removed under DOS 0010-19 or DOS 0010-20; or
- Requirements for operations shift management notification and authorization/sign off when temporary changes were installed under DOS 0010-19 or removed under DOS 0010-19 or DOS 0010-20.

The licensee's corrective actions included planning to change all the station procedures that control the installation and removal of temporary heaters. Because this violation was of very low safety significance and it was entered into the licensee's CAP as IR 876126, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. **(NCV 05000237/2009002-03)**

.4 <u>In-depth Review - IR 866417, "Reactor Building (RB) Vent Damper 2-5741B Slow to Close."</u>

a. <u>Inspection Scope</u>

The inspectors reviewed the problem identification, prioritization, and corrective action effectiveness aspects of IR 866417, "RB Vent Damper 2-5741B Slow to Close." Damper 2-5741B is a secondary containment isolation valve (SCIV). The damper closes upon a reactor building isolation signal to prevent the escape of radiation. This review constituted one sample as defined in IP 71152.

The inspectors reviewed the following documents:

IR 546603, "U3 RBV Damper INOP;"

IR 664903, "2-5741B Reactor (Rx) Building (Bldg) Inlet Damper Exceeded Stroke Time (Quad Cities);"

EACE 664903, "2-5741B Rx Bldg Inlet Damper Exceeded Stroke Time (Quad Cities);"

Nuclear Event Report (NER) QC-07-100;

IR 729845, “Problems with Solenoid Valves at QC;”

IR 829039, “SCIV 2-5741B Failed to Stroke Closed within 60 Second Requirement;”

IR837265, “MRFF for SCIV 2-5742B;”

IR 837291, “MR Reliability Criteria Exceeded for RBV [reactor building ventilation] SCIVs;”

IR 863042, “2-5741B will not Pass Fail Safe Testing after Valve Work;”

IR 866417, “RB Vent Damper 2-5741B Slow to Close;”

IR 870258, “RBV Experienced Maintenance Rule Functional Failure on SCIV (Repeat Maintenance Preventable Functional Failure);” and

IR 877591, “Potential 10 CFR 50 Part 21 Notification of Versa Air Solenoid.”

b. Findings

Introduction: The inspectors identified a NCV of 10 CFR 50, Appendix B, Criterion XVI, “Corrective Action”, of very low safety significance (Green) for the failure to correct a degraded actuator solenoid valve in the 2-5741B reactor building ventilation secondary containment isolation valve (SCIV) in a timely manner.

Description: The 2-5741B reactor building SCIV failed to close automatically during a secondary containment isolation on January 13, 2009.

Failures of the reactor building ventilation SCIV actuators had been identified at both Dresden Station and Quad Cities Station as early as 2006. The actuators at Dresden and Quad Cities are identical and the 4-way solenoid valves (4WSVs) in the actuators are purchased from the same vendor. Multiple failures at Quad Cities in 2007 prompted an equipment apparent cause evaluation (EACE) that identified that there were manufacturing flaws with the existing 4WSVs in the actuators (a condition adverse to quality) and that a new solenoid valve with a stronger spring and different lubrication was required to ensure proper operation of the actuators. The 4WSVs in the SCIV actuators were safety-related components.

The information that the safety-related SCIV actuator components were improperly manufactured was reported to Dresden Station via a Green Nuclear Event Report (NER QC-07-099) on November 15, 2007. The need to replace the SCIV actuator 4WSVs with a different style was entered into the Dresden Corrective Action Program in IR 729845 on January 31, 2008. A compensatory measure, recommended by the vendor and put in place at Quad Cities until the actuator 4WSVs could be replaced, was to cycle the dampers several times per month. The work to replace the solenoid on 2-5741B at Dresden was not scheduled until 2009. Dresden station management did not put the same compensatory measure in place to cycle the dampers several times per month.

The licensee identified a condition adverse to quality on the SCIV actuator 4WSVs in January 2008. Title 10 of the Code of Federal Regulations, Part 50, Appendix B, Criterion XVI, "Corrective Action", states in part, "measures shall be established to ensure that conditions adverse to quality…are promptly identified and corrected."

Regulatory Issue Summary 2005-20, Revision 1, explains the NRC policy on what is determined to be promptly corrected. Section 7.2, states, in part, "If the licensee does not resolve the degraded or non-conforming condition at the first available opportunity or does not appropriately justify a longer completion schedule, the staff would conclude that the corrective action has not been timely."

A maintenance window was scheduled and took place for the 2-5741B SCIV for the week of January 5, 2009. However, the 4WSV within the actuator was not replaced. The 2-5741B SCIV failed when it was called upon for a secondary containment isolation on January 13, 2009, because the 4WSV failed. Therefore, the condition adverse to quality was neither promptly corrected, nor was a longer completion time justified.

<u>Analysis</u>: The inspectors determined that the failure to coordinate work activities in order to replace the degraded 4WSV for the 2-5741B SCIV during the work window that started on January 5, 2009, prior to its failure on January 13, 2009, was a performance deficiency. The resident inspectors used the guidance provided in Regulatory Issue Summary 2005-20, Revision 1, Section 7.2, "If the licensee does not resolve the degraded or non-conforming condition at the first available opportunity or does not appropriately justify a longer completion schedule, the staff would conclude that the corrective action has not been timely." Using the guidance contained in IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," dated December 4, 2008, the inspectors determined that the finding was more than minor because it was associated with the Reactor Safety Barrier Integrity Cornerstone objective of maintaining the functionality of secondary containment.

The inspectors evaluated the finding using IMC 0609, Attachment 0609.04, dated January 10, 2008. Per Table 4a, under Containment Barrier, question 1, "Does the finding only represent a degradation of the radiological barrier function provided for the …Standby Gas Treatment System," the inspectors answered, YES. The SCIVs isolate the secondary containment to ensure the effectiveness of the Standby Gas Treatment System. Therefore the finding was determined to be Green. The inspectors determined that this issue also affected the cross-cutting area of Problem Identification and Resolution. The licensee was aware of the equipment issue but did not evaluate and prioritize the issue to ensure corrective actions were taken in a timely manner. P.1(c).

<u>Enforcement</u>: The Code of Federal Regulations, Title 10, Part 50, Appendix B, Criterion XVI, Corrective Action, states, in part, "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and non-conformances are promptly identified and corrected."

A condition adverse to quality for all 4WSVs in the actuators for the Unit 2 and Unit 3 reactor building ventilation secondary containment isolation boundary dampers was identified by the licensee in IR 729845, "Problems with Solenoid Valves at QDC," on January 31, 2008.

Contrary to the above, on January 5, 2009, the licensee had an opportunity but failed to promptly correct a condition adverse to quality associated with the 4WSV in the actuator for the Unit 2 reactor building ventilation secondary containment isolation boundary damper 2-5741B during a scheduled maintenance window. The licensee's corrective action included replacing all the 4WSVs in the actuators for all the Unit 2 and Unit 3 reactor building ventilation secondary containment isolation boundary dampers. Because this violation was of very low safety significance and it was entered into the licensee's CAP as IR 877591, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. **(NCV 05000237/2009002-04)**

4OA3 Event Follow-up (71153)

.1 Failure to Properly Reset the Unit 2 Reactor Building Fuel Pool Channel A Area Radiation Monitor Resulted in an Emergency Safety Feature Systems Actuation

a. Inspection Scope

The inspectors reviewed the circumstances involved in a reactor building ventilation isolation and standby gas treatment (SBGT) system start on January 13, 2009. The inspectors reviewed IR 866445 and interviewed the operators involved in the event.

This represented one inspection sample.

b. Finding

Introduction: A Green finding with no violation of regulatory requirements was self-revealed when an operator performed an incorrect response to an unexpected alarm in the control room that resulted in a reactor building ventilation isolation and SBGT start.

Description: On January 13, 2009, the reactor building fuel pool channel A area radiation monitor alarmed unexpectedly. The alarm was only in momentarily. Upon finding the area radiation monitor indicating normally the operator intended to take action to press the "Reset" push button to reset the alarm. Instead, the operator mistakenly pressed the "Trip Check" push button. This caused the radiation monitor to increase to the "H" trip setpoint, causing a Secondary Containment Isolation. Reactor building ventilation isolated and SBGT actuated. The licensee entered the TS Limiting Condition for Operation (LCO) 3.6.4.1 A.1 when secondary containment differential pressure dropped below -.25 inches of water and remained in that LCO for about 17 minutes.

Analysis: The inspectors determined that the failure to press the correct button on the reactor building fuel pool channel A area radiation monitor, was a performance deficiency warranting a significance evaluation. The inspectors concluded that the finding was more than minor in accordance with Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," issued on December 04, 2008, because it impacted the structures, systems, and components attribute of the Barrier Integrity Cornerstone (containment) objective. This deficiency challenged a safety system and could have affected the availability and capability of components and systems that respond to initiating events.

The inspectors completed a Phase 1 significance determination of this issue using IMC 0609, "Significance Determination Process," Attachment 0609.04,Table 4a dated January 8, 2008, and determined that this finding impacted the Barrier Integrity Cornerstone column. The inspectors answered YES to question #1 under the Barrier

Integrity column on page E1-6. Therefore, the issue screened out as having very low significance (Green). The inspectors also concluded that this finding affected the cross-cutting issue of Human Performance (Personnel) because the operator failed to utilize human performance error prevention techniques. H.4(a).

Enforcement: Resetting an area radiation alarm is considered an activity within the "skill of the craft," and therefore did not require a procedure. Resetting the area radiation alarm was not part of the alarm response procedure. The inspector's interview with the operator confirmed that the operator knew how to reset the area radiation monitor and made a mistake by pushing the wrong button. Therefore, this event was not determined to be in violation of regulatory requirements. This event was documented in IR 866445. The licensee performed the following corrective actions: the individual was temporarily removed from licensed shift duties and no manipulation by the operator of any equipment in the plant or the control room was allowed without a peer check until January 18, 2009. **(FIN 05000237/2009002-05)**

.2 (Closed) Licensee Event Report (LER) 05000249/2008-001-00, "Unit 3 Drywell Floor Drain Sump Monitoring System Declared Inoperable"

On August 16, 2008, operations personnel unsuccessfully attempted to pump the Unit 3 drywell floor drain sump to partially satisfy TS Surveillance Requirement (SR) 3.4.4.1. Operations personnel started a shutdown of Unit 3 to fulfill the requirements of TS 3.4.4, "RCS Operational Leakage." The licensee requested and received a Notice of Enforcement Discretion and stopped the Unit 3 shutdown at about 30 percent reactor power. The licensee identified later when the drywell floor drain sump isolation valves were disassembled during the Unit 3 outage in November 2008 that the valve plug had separated from the stem. The licensee determined that the maintenance procedure governing valve replacement was inadequate which could result in higher than desired actuator output and seating forces. A licensee identified violation is documented in Section 4OA7. The inspectors reviewed the licensee's corrective actions. The inspectors had no issues with the licensee's corrective actions and determined that they were completed or had an acceptable time table for completion.

This represented one inspection sample. This LER is closed.

.3 (Closed) LER 05000249/2008-002-00, "Unit 3 Primary Containment Isolation Valve Declared Inoperable"

Introduction: The inspectors identified an NCV of Dresden Station Improved TS 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)," of very low safety significance (Green) for the failure to declare Primary Containment Isolation Valve 3-3702 inoperable and take actions in accordance with the requirements of TS 3.6.1.3 Required Action A.1.

Description: On October 29, 2008, operations personnel identified that the OPEN indication light for the normally open PCIV 3-3702 was flickering on and off. Valve 3-3702 is the Unit 3 Reactor Building Closed Cooling Water (RBCCW) drywell supply header isolation to the containment and is designed to require a manual close signal from the Main Control Room during certain accident events.

The RBCCW system is a closed loop system that, in part, cools primary containment heat loads. Primary Containment Isolation Valve 3-3702 is one of two isolation valves in the RBCCW supply line to the containment. The second valve is a check valve. The 3-3702 valve can not be closed during normal plant operation without a plant shutdown. Loss of RBCCW cooling flow to the primary containment heat loads requires immediate plant shutdown if flow can not be reestablished within two minutes. Damage to electrical equipment and reactor recirculation pump seals and bearings may result. Loss of RBCCW to the drywell coolers results in drywell atmosphere heat up and subsequent drywell pressure increase.

The light bulb for PCIV 3-3702 was replaced several times but the flickering continued. Operations personnel performed a prompt operability review for valve 3-3702 and concluded that the flickering light indicated that the valve was operable but the light socket was failing. The indication light was declared inoperable and actions were taken in accordance with the requirements of TS 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation." Technical Specification 3.3.3.1 Required Action A.1 required the light to be restored to operable within 30 days. On November 1, 2008, an investigation into the flickering of the indication light discovered an intermittent high resistance electrical connection between the primary and secondary stab on the motor control center (MCC) that provides control power to PCIV 3-3702. The intermittent high resistance electrical connection caused the flickering indicator light and also would have prevented valve closure from the Main Control Room.

On November 1, 2008, at approximately 10:50 a.m., PCIV 3-3702 was declared inoperable and actions were taken in accordance with the requirements of TS 3.6.1.3, "Primary Containment Isolation Valves (PCIV)." Technical Specification 3.6.1.3 Required Action A.1 required the valve to be restored to operable status within 4-hours or PCIV 3-3702 would have to be closed, isolating the RBCCW flow. The valve was restored to operable status on November 1, 2008, at approximately 1:09 p.m.

The cause of the intermittent high resistance electrical connection was indeterminate. The licensee concluded that the most probable cause was that the electrical connection was marginal at installation and over time surrounding equipment vibration caused the connection to be intermittent.

Procedure OP-AA-103-102, "Watch-Standing Practices," Revision 8, Section 4.2, "Control Room Watch-Standing Practices," directs operations personnel, in part, to "be aware if indicating light status changes since a loss of control power can render components inoperable from the Control Room." In addition, procedure OP-AA-108-115-1002, "Supplemental Consideration for On-Shift Immediate Operability Determinations (CM-1)," Revision 1, provides guidance to the on-shift Senior Reactor Operator (SRO) to ensure that a thorough review of immediate operability of systems, structures and components (SSC) is performed and documented. The procedure directs the SRO, in part, to "evaluate the effects of the condition and possible failure modes on the ability of the SSC to perform its required function(s)." The failure to follow the instructions on OP-AA-103-102 and OP-AA-108-115-1002 to validate PCIV 3-3702 function and the lack of urgency in responding to the problem caused PCIV 3-3702 to be inoperable for a period of time that exceeded the TS required action completion time.

The flickering indication light that was discovered on October 29, 2008, provides evidence that PCIV 3-3702 was historically inoperable for a period of time that exceeded the 4-hour limit of TS 3.6.1.3 Required Action A.1.

In addition, the procedures used by Operations personnel on October 29, 2008, lacked specific guidance to alert users that a failed or flickering indication light associated with a motor operated valve (MOV) may indicate problems that could affect valve operation and that valve operability must be verified. An extent of condition review of operating procedures identified that revisions to enhance guidance must be made to at least five procedures: DGP 03-02, "Normal Control Room Inspection," DOP 0040-01, "Station Motor Operated Valve Operations," DOP 0040-04, "Control Panel Light Bulb and LED Replacement," DOP 6700-20, "480V Circuit Breaker Trip," and DOS 0040-12, "Penetration Flow Path PCIV Position Channel Check and Control Room PCIV Position Verification."

The licensee generated IR 837675 and IR 839009 to address this issue.

Analysis: The inspectors determined that the licensee's failure to follow the instructions contained in procedures OP-AA-103-102, "Watch-Standing Practices," and OP-AA-108-115-1002, "Supplemental Consideration for On-Shift Immediate Operability Determinations (CM-1)," to validate PCIV 3-3702 functionality and the lack of urgency in responding to the problem that resulted in PCIV 3-3702 being inoperable for a period of time that exceeded the 4-hour limit of TS 3.6.1.3, "Primary Containment Isolation Valves (PCIV)," Required Action A.1, was a performance deficiency warranting a significance evaluation. Using IMC 0612, Appendix B, "Issue Screening," issued on December 4, 2008, the inspectors determined that this finding was more than minor because it impacted the Barrier Integrity objective to provide reasonable assurance that physical design barriers (i.e., containment) protect the public from radionuclide releases caused by accidents or events. Although PCIV 3-3702 was inoperable for a period of time that exceeded the TS required action completion time, the RBCCW system is a closed loop system that provides a boundary to prevent a radioactive release to the environment during normal plant operation. In addition, PCIV 3-3702 is one of two isolation valves in the RBCCW supply line to the containment. The second valve is a check valve that was operable and capable of isolating the supply line if required. As described above in the description section, the procedures used by Operations personnel on October 29, 2008, lacked specific guidance to alert users that a failed or flickering indication light associated with a MOV may indicate problems that could affect valve operability. The inspectors considered this an additional factor that contributed to the lack of urgency in responding to the problem. Therefore, the inspectors determined that this finding also affected the cross-cutting area of human performance, resources aspect H.2(c) because the licensee failed to provide complete, accurate and up-to-date procedures.

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

Enforcement: Dresden Station Improved Technical Specification 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)," states, in part, "Each PCIV,..., shall be OPERABLE." Technical Specification 3.6.1.3 Required Action A.1 requires an inoperable PCIV to be restored to operable status within 4-hours; otherwise the affected penetration flow path would have to be isolated by use of at least one closed and de-activated valve.

Contrary to the above, from October 29, 2008, until November 1, 2008, the licensee failed to declare PCIV 3-3702 inoperable and take actions in accordance with the requirements of TS 3.6.1.3 Required Action A.1. Valve 3-3702 was inoperable for a period of time that exceeded the 4-hour limit of TS 3.6.1.3 Required Action A.1. The licensee generated IR 837675 and IR 839009 to address this issue. Some of the corrective actions implemented by the licensee to address this issue include: the initiation of a training request to re-enforce with Operations personnel the potential operability issues when light indications are not functioning properly, and the revision of Operations procedures to include guidance to alert users that a failed of flickering indication light associated with a motor operated valve may indicate problems that could affect valve operation, and that valve operability must be verified. 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 Section VI.A.1 or the NRC Enforcement Policy. **(NCV 05000249/2009002-06)**

This LER is closed.

4OA6 Management Meetings

.4 Exit Meeting Summary

On April 15, 2009, the inspectors presented the inspection results to Mr. T. Hanley, 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 Meeting

An interim exit was conducted for:

- Public Radiation Safety Radwaste Processing and Transportation Inspection with Mr. J. Sipek and others on February 13, 2009.

The inspectors confirmed that none of the potential report input discussed was considered proprietary.

4OA7 Licensee-Identified Violations

The following violations of very low significance (Green) were identified by the licensee and are violations of NRC requirements, which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as Non-Cited Violations.

- Technical Specification 5.4.1 requires that written procedures be established and implemented for activities provided in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Procedures specified in Regulatory Guide 1.33 include procedures for performing maintenance. Contrary to this requirement, on August 16, 2008, operations personnel unsuccessfully attempted to pump the

Unit 3 drywell floor drain sump to partially satisfy TS Surveillance Requirement 3.4.4.1. The licensee identified later that the valve plug had separated from the stem in the Drywell Floor Drain Sump Containment Isolation Valve 3-2001-1005. The licensee determined that the maintenance procedure governing valve replacement was inadequate which resulted in higher than desired actuator output and seating forces. The inspectors reviewed the licensee's corrective actions. The inspectors had no issues with the licensee's corrective actions and determined that they were completed or had an acceptable time table for completion. This incident was identified in the licensee's corrective action program as Issue Report 807914 and documented in LER 249/2008-001-00, "Unit 3 Drywell Floor Drain Sump Monitoring System Declared Inoperable." This violation was determined to be of very low safety significance because the containment isolation valve failed in a closed condition ensuring its ability to perform a containment isolation function.

- Technical Specification 5.4.1 requires that written procedures be established and implemented for activities provided in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Procedures specified in Regulatory Guide 1.33 include radiation protection procedures for radiation surveys and contamination control which are provided by licensee procedure RP-AA-503, "Unconditional Release Survey Method," Revision 1. That procedure requires that material or equipment that is unconditionally released outside the radiologically controlled area (RCA) have no detectable licensed radioactive material. Contrary to this requirement, on February 3, 2009, a small strap was unconditionally released outside the RCA within the licensee's protected area but later that same day found to have low levels (10,000 disintegrations per minute) of fixed contamination. This incident was identified in the licensee's corrective action program as Issue Report 875773. This violation was determined to be of very low safety significance because the uncontrolled strap did not result in any dose to a member of the public in the restricted or controlled areas.

- The Licenses for Units 2 and 3 require that the licensee follow their fire protection program. Fire protection program requirements are identified in the Technical Requirement Manual. Technical Requirements Manual, Section 3.7.k, Condition B, states in part that the Auxiliary Electrical Equipment Room (AEER) halon system, including the extended discharge cylinders shall be operable and if not then a dedicated fire watch shall be established within 1 hour. Contrary to the above, on October 29, 2008, the licensee identified that all of the AEER halon fire suppression extended discharge cylinders were completely discharged. The licensee concluded that the cylinders had been in this condition since October 23, 2008, based on a trend recording of a downward spike in room temperature caused by the discharge.

  The inspectors and the NRC Region III senior reactor analyst (SRA) used IMC 0609 Appendix F, "Fire Protection Significance Determination Process" to evaluate the significance of the finding. The finding category of Fixed Fire Protection Systems was assigned because the function of the automatic Halon suppression system for the AEER was degraded. Since only the extended release portion of the system was affected and the initial release was fully functional, a moderate degradation rating for the finding was used. The NRC assumed that the design concentration could be achieved but could not be maintained for sufficient time to ensure fire extinguishment. The condition existed for 6 days. The Significance Determination Process (SDP) uses a generic duration factor of 0.1 if the duration of the degradation is between 3 and 30 days. Since the exact duration was known to be 6 days, the SRA calculated an actual duration factor of 0.016 (6/365 days). The generic fire area frequency for a cable spreading room with other electrical equipment is 6.0E-3/yr. The change in Core Damage Frequency (CDF) calculated in the phase 1 SDP is determined by multiplying the duration factor by the generic fire area frequency which is estimated to be 9.6E-5/yr. Since this value is greater than 1E-5, the screening criterion for moderate degradation findings, a phase 2 evaluation was required.

In step 2.1 of the SDP, the identified safe shutdown path for a fire in the AEER was evaluated. Because the finding does not affect the safe shutdown path, it can be credited. After reviewing the licensee's safe shutdown analysis, the SRA determined that safe shutdown after a fire in the AEER would require local operator actions. Therefore, a safe shutdown unavailability factor of 0.1 was applied. The delta CDF was recalculated by multiplying the duration factor, fire frequency, and safe shutdown path unavailability factor and was determined to be 9.6E-6/yr. This result was conservative because it does not include any credit for manual fire suppression using the installed carbon dioxide system which was unaffected by the finding. Since this delta CDF was less than the screening criterion of 1E-5, the finding was determined to be of very low safety significance and screened to Green in step 2.1.4. The inspectors had no issues with the licensee's corrective actions and determined that they were completed or had an acceptable time table for completion.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
T. Hanley, Site Vice President
S. Marik, Station Plant Manager
H. Bush, Radiation Protection Manager
B. Finely, Security Manager
D. Glick, Shipping Specialist
J. Griffin, Regulatory Assurance - NRC Coordinator
D. Gronek, Operations Director
J. Hansen, Corporate Licensing
L. Jordan, Training Director
R. Kalb, Chemistry
P. Karaba, Maintenance Director
M. Kluge, Design Engineer
D. Leggett, Nuclear Oversight Manager
R. Luburn, Radiation Protection
M. McDonald, Mechanical Maintenance
T. Mohr, Maintenance Planning
M. Overstreet, Lead Radiation Protection Supervisor
G. Petrovic, Maintenance
C. Podczerwinski, Maintenance Rule Coordinator
P. Quealy, Emergency Preparedness Manager
E. Rowley, Chemistry
R. Rybak, Regulatory Assurance
J. Sipek, Engineering Director
N. Starcevich, Radiation Protection Instrumentation Coordinator
J. Strmec, Chemistry, Environmental and Radwaste Manager
S. Taylor, Regulatory Assurance Manager
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R. Zuffa, Illinois Emergency Management Agency
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LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened:

05000237/2009002-01 05000249/2009002-01	NCV	Failure to Develop a Pre-Fire Plan for Fire Zone 18.6
05000237/2009002-02 05000249/2009002-02	NCV	Failure to Implement and Maintain in Effect All Provision of the Approved Fire Protection Program as Described in the UFSAR
05000237/2009002-03	NCV	Failure to Ensure the Control of the Design Basis Was Correctly Translated Into Station Procedures
05000237/2009002-04	NCV	Failure to Take Corrective Actions to Replace a Degraded Valve in a Timely Manner
05000237/2009002-05	FIN	Operator Performed an Incorrect Response to an Unexpected Alarm in the Control Room
05000249/2009002-06	NCV	Failure to Declare Primary Containment Isolation Valve Inoperable and Take Required Actions

Closed:

05000237/2009002-01 05000249/2009002-01	NCV	Failure to Develop a Pre-Fire Plan for Fire Zone 18.6
05000237/2009002-02 05000249/2009002-02	NCV	Failure to Implement and Maintain in Effect All Provision of the Approved Fire Protection Program as Described in the UFSAR
05000237/2009002-03	NCV	Failure to Ensure the Control of the Design Basis Was Correctly Translated Into Station Procedures
05000237/2009002-04	NCV	Failure to Take Corrective Actions to Replace a Degraded Valve in a Timely Manner
05000237/2009002-05	FIN	Operator Performed an Incorrect Response to an Unexpected Alarm in the Control Room
05000249/2009002-06	NCV	Failure to Declare Primary Containment Isolation Valve Inoperable and Take Required Actions
05000249/2008-001-00	LER	Unit 3 Drywell Floor Drain Sump Monitoring System Declared Inoperable
05000249/2008-002-00	LER	Unit 3 Primary Containment Isolation Valve Declared Inoperable

Discussed: None.

LIST OF DOCUMENTS REVIEWED

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

1R01 Adverse Weather Protection (71111.01)

- WO 01048083-01 & 02, "D2 6Y PM Functional Test of the CCSW Emergency Pump"
- WO 01122483, "D2 Semi-Annual PM Emergency Diesel Pump Operation"
- DRE03-0026, "Analysis of the Intake Canal, CCSW Heat Exchanger, and Temporary Pumps Following A Dam Failure and 1" LOCA"

1R04 Equipment Alignment (71111.04)

- DOP 1500-M1 "Unit 2 LPCI and Containment Cooling Valve Checklist," Revision 38
- DOP 6600-M1, "Unit 2 Standby Diesel Generator," Revision 27
- DOP 6600-E1, "Unit 2 Standby Diesel Generator," Revision 4
- DOP 6600-M2, "Unit 2/3 Standby Diesel Generator," Revision 24
- DOP 66600-E2, "Unit 2/3 Standby Diesel Generator," Revision 05
- IR 886403, "NRC Inspector Questions Valve Information on DGCW System"
- IR 886431, "NRC Inquiry Regarding U2 D/G System Valves"
- General Electric, Service Information Letter (SIL) 336, "Surveillance Testing Recommendations For HPCI and RCIC Systems," Revision 1

1R05 Fire Protection (71111.05)

- DRE97-0105, "Determination of Combustible Loading," Revision 8.
- Dresden Station Units 2 and 3 Fire Protection Reports Volume 1, "Updated Fire Hazards Analysis"
- IR 870982, "NRC Id: Fire Pre-Plans and Comb Load Updates – SBLC Rm"
- OP-AA-201-009, "Control of Transient Combustible Material," Revision 7
- NES-MS-5.1, "Combustible Loading Standard," Revision 2
- IR 873977, "Fire Pre-Plan Lists Incorrect Smoke Detector and Alarm Bell"
- IR 875688, "NRC Identified Fire Pre-Plan Error"
- IR 869302, "125V U2 Alt Batt Average Electrolyte Temperature Low"
- IR 750656, "NRC PI&R Question on Fire Pre-Plans"
- OP-AA-201-008, "Pre-Fire Plan Manual," Revision 2
- Dresden Station Individual Plant Examination of External Events (IPEEE) Submittal Report, Revision 1

1R06 Flooding (71111.06)

- IR 867770, "Heating Coil Failure Results in Water in Control RM Panels"
- IR 889472, "Fire Hazard Analysis (FHA) does not Match Field"
- IR 896456, "Main Control Penetrations not Shown on Drawings"
- IR 893448, "Leakage into Control Rm per IR 867770"
- IR 892096, IR "Support Bolts not Justified as Fire Barrier When Installed"

- Modification No. 12-2/3-86-6D, Upgrade Control Room Lighting and HVAC Systems to Correct Problems Identified by DCRDR
- Dresden Fire Hazard Analysis for Fire Zone 7.0.A
- EC 374653, “Fire Barrier Through Wall Bolt 86-10 Evaluation”

1R12 Maintenance Effectiveness (71111.12)

- NUREG/CR 5933, “High Pressure Coolant Injection (HPCI) System Risk-Based Inspection Guide for Dresden Nuclear Power Station, Units 2 and 3”
- IR 825731, “Maintenance Rule Clarification Questions”
- IR 869643, “IR654273 Reclassified As An MRFF [maintenance rule functional failure]”
- IR 654273, “U2 HPCI Inlet Drain Pot Piping Leak”
- Maintenance Rule Periodic Assessment #7, October 1, 2006 to September 30, 2008
- IR 815935, “Unexpected Alarm”

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

- CO 69979, “CCSW Loop II – Various PM’s and Repairs”
- CO 69910, “U2 LPCI Loop ‘B’ (2C & 2D LPCI Pumps) LCO Work”
- CO 70147, “3A Diesel Start Up Compressor”
- CO 70150, “Standby Diesel Generator”
- CO 70811, “D/G Fuel Oil Transfer Pmp Disch RV”
- CO 70824, “U2 D/G(Starting Air Compressors & Receivers)”
- CO 70822, “U2 D/G (Lube Oil & Coolant OOS)”
- CO 70854, “(U2 D/G) U2 D/G Replace Cooling Water Flex Hoses”

1R15 Operability Evaluations (71111.15)

- IR 845561, “NOS Ids Temperature Recorder Points not Working”
- DOS 1600-29, “Unit 2 and 3 Drywell Temperature Surveillance,” Revision 04
- Technical Specification 3.6.1.5, “Drywell Air Temperature”
- WO 1062071, “TR 3-5741-19 Point 10 TC Open. No Tech Spec Violation”
- IR 881159, “Lab Report on Unidentified Substance on Bottom of Bay 13”
- Material Safety Data Sheet for Spectrus CT 1300 (water-based microbial control agent)
- EC 343189, “Operability Determination: Torus Level Increase”
- Operability Determination 99-024, “Secondary Containment Operability”

1R18 Plant Modifications (71111.18)

- DOS 0010-20, “Cold Weather Operations For Unit 1 & Out Buildings,” Revision 11
- CC-AA-12, “Temporary Configuration Changes,” Revision 12

1R19 Post-Maintenance Testing (71111.19)

- DMP 1501-04, “Containment Cooling Service Water (CCSW) Pump Maintenance,” Revision 16
- MA-AA-716-011, “Work Execution & Close Out,” Revision 11
- WO 1113585-01, “Move Fuel Oil Flex Line – Piping Needs to be Separated – 2 EDG”
- WO 659478-01, “EM Replace U2 Emerg Diesel Gen Gov SD Solenoid EC 331152”
- WO 644121-01, “MM D2 6Y PM Standby Diesel Generator Inspection”
- WO 790548-02, “EM Determ/Reterm EDG 2 Turbo Oil Circ Motor 2-6660”

- WO 1202842, “D2 1M TS Unit Diesel Generator Operability”
- MA-DR-MM-5-66001, “Diesel Generator Post Maintenance Testing Run,” Revision 6
- DOS 6600-12, “Diesel Generator Tests Endurance and Margin/Full Load Rejection/ECCS/Hot Restart,” Revision 47
- IR 884180, “NRC Questions Post-Maintenance Testing on Unit 2 EDG”
- IR 864025, “Non-EQ HGA Relays Installed in EQ Applications of ASR Relays”
- IR 865482, “Non-EQ Relay Installed in EQ Application of ASR Relay”
- IR 865464, “Non-EQ HGA Relay Installed in EQ Application of ASR Relay”
- IR 884550, “Non-EQ Relay Reserved for a EQ Application”
- IR 882581, “NRC Resident Concern – Weld Document Issue in Work Package”

1R22 Surveillance Testing (71111.22)

- Appendix A, “Unit NSO Daily Surveillance Log,” Revision 114
- DOS 1400-09, “Core Spray System IST Comprehensive/Preservice Pump Test with Torus Available,” Revision 5
- DIP 0600-01, “Unit 3 Reactor Feedwater Flow Transmitter/Indicator/Recorder Calibration,” Revision 35
- IR 886508,”Issues Identified FW Flow Calibration”
- IR 886722, “U3 DW Entry Determined the Main Seat of 3C ERV is Leaking”

2PS2 Radioactive Material Processing and Transportation

- Shipment Manifest, Radiological Surveys and Associated Documentation for Shipment DW-08-011; LSA-II; Condensate Resin; Shipment dated March 4, 2008
- Shipment Manifest, Radiological Surveys and Associated Documentation for Shipment DM-07-0024; LSA-II; Contaminated Refueling Equipment; Shipment dated March 7, 2007
- Shipment Manifest, Radiological Surveys and Associated Documentation for Shipment DM-07-140; LSA-II; Control Rod Drives; Shipment date November 7, 2007
- Shipment Manifest, Radiological Surveys and Associated Documentation for Shipment DM-07-19; LSA-II; Contaminated Equipment; Shipment dated November 20, 2007
- Shipment Manifest, Radiological Surveys and Associated Documentation for Shipment DM-08-136; LSA-II; Contaminated Equipment; Shipment dated November 18, 2008
- Shipment Manifest, Radiological Surveys and Associated Documentation for Shipment DW-09-004; LSA-II; Concentrator Waste Sludge; Shipment dated January 27, 2009
- Shipment Manifest, Radiological Surveys and Associated Documentation for Shipment DW-07-003; Type B(U), Irradiated Hardware; shipment date February 22, 2007
- Shipment Manifest, Radiological Surveys and Associated Documentation for Shipment DW-08-005; Type B(U), Irradiated Hardware; Shipment dated February 13, 2008
- 10 CFR Part 61 Waste Stream Analysis Results for Condensate Resin (May 2007), Reactor Water Cleanup Resin (May 2007), Dry-Active Waste (May 2008), CRD/Torus Filters (November 2006), Concentrator Waste (November 2007), and Fuel Pool Resin (November 2007)
- RP-DR-605; 10 CFR Part 61 Waste Stream Sampling and Analysis; Revision 2
- AR 00828474; and Associated Apparent Cause Report/Human Performance Investigation; Contacted by Hittman Transport Driver of a Traffic Incident; dated October 8, 2008
- RP-AA-603; Inspection and Loading of Radioactive Material Shipments; Revision 4
- RP-AA-603-1001; Inspection and Loading of Radioactive Material/Waste Shipments; Revision 2
- Certificate of Compliance for Radioactive Material Packages;
- Model TN-RAM – Certificate Number 9233, Revision 7 and

- Model CNS 8-120B – Certificate Number 9168; Revision 16
- RP-AA-600-1001; Exclusive Use and Emergency Response Information; Revision 4
- RP-AA-601; Surveying Radioactive Material Shipments; Revision 10
- RP-AA-602; Packaging of Radioactive Material Shipments; Revision 12
- Focused Area Self-Assessment Report; Transportation and Radwaste; dated December 15, 2008
- Chemistry, Radwaste, Effluent and Environmental Monitoring Audit Report;
- Audit NOSA-DRE-08-04; dated April 30, 2008
- Audit Templates for Chemistry, Radwaste, Effluent, Environmental Monitoring, Handling, Storage and Shipping; Revision 2
- AR 00811979; Resin on Floor of Radwaste Basement; dated August 28, 2008
- AR 00656219; Spent Resin Tank/Sludge Tank Rooms with Sludge on Floor; dated August 1, 2007
- AR 00746403; Radwaste Floor Drain Collector/Waste Collector Room Inspection; dated March 7, 2008
- DOP 2000-53; Transfer of Resin from Spent Resin Tank to Contract Vendor Mobile Solidification/Dewatering Unit; Revision 09
- RW-AA-100; Process Control Program for Radioactive Wastes; Revision 7
- Check-In-Self-Assessment Report; Radwaste Onsite Long-Term Storage; dated December 10, 2007
- Hazardous Materials Transport Lesson Plan; Revision 006c
- Hazardous Materials Transport for Warehouse Personnel Lesson Plan; Revision 002
- Radioactive Materials Shipping for Laborers Lesson Plan; Revision 02
- Radioactive Material Shipments for Radiation Protection Technicians Lesson Plan; Revision 02
- AR 00704415; Poor CRD Sea-Land Condition; dated November 28, 2007
- AR 00707740; Inappropriate Material Loaded in Radwaste Sea-Van; dated December 3, 2007
- AR 00582895; Shipping Procedure Violation; dated January 24, 2007

4OA3 Event Follow-up (71153)

- IR 839009, "Unexpected Tech Spec Entry: RBCCW PCIV Declared Inoperable"
- OP-AA-108-115, "Operability Determinations (CM-1)," Revision 8
- OP-AA-108-115-1002, "Supplemental Consideration for On-Shift Immediate Operability Determinations (CM-1)," Revision 1
- OP-AA-103-102, "Watch-Standing Practices," Revision 8

4OA7 Licensee Identified Violations

- AR 875773 and Associated Quick Human Performance Investigation; Radioactive Material Event in the Unit 1 Corridor; dated February 3, 2009
- RP-AA-503; Unconditional Release Survey Method; Revision 1

LIST OF ACRONYMS USED

ACE	Apparent Cause Evaluation
AEER	Auxiliary Electrical Equipment Room
ASME	American Society of Mechanical Engineers
AOP	Abnormal Operating Procedure
ALT	Alternative
BLDG	Building
CAP	Corrective Action Program
CCSW	Containment Cooling Service Water
CFR	Code of Federal Regulations
CO	Clearance Order
CS	Core Spray
CRD	Control Rod Drive
DOT	Department of Transportation
DRP	Division of Reactor Projects
DRA	Dry Active Waste
EACE	Equipment Apparent Cause Evaluation
EC	Engineering Change
EDG	Emergency Diesel Generator
ERV	Electromagnetic Relief Valve or Emergency Relief Valve
FHA	Fire Hazard Analysis
4WSV	4-Way Solenoid Valves
HPCI	High Pressure Coolant Injection
HX	Heat Exchanger
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IR	Inspection Report
IR	Issue Report
IST	In-service Test
LER	Licensee Event Report
LCO	Limiting Condition for Operation
LOCA	Loss of Coolant Accident
LPCI	Low Pressure Coolant Injection
MCC	Motor Control Center
MR	Maintenance Rule
MRFF	Maintenance Rule Functional Failure
MOV	Motor Operated Valve
NCV	Non-Cited Violation
NDE	Nondestructive Examination
NEI	Nuclear Energy Institute
NER	Nuclear Event Report
NFPA	Nation Fire Protection Association
NOS	Nuclear Oversight
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NSO	Nuclear Safety Operator
PARS	Publicly Available Records
PCIV	Primary Containment Isolation Valve
PI	Performance Indicator or Problem Identification
PI&R	Problem Identification and Resolution

PM	Planned or Preventative Maintenance, or Post-Maintenance
PMT	Post-Maintenance Test
RCA	Radiologically Controlled Area
RB	Reactor Building
RBCCW	Reactor Building Closed Cooling Water
RX	Reactor
RCS	Reactor Coolant System
SBO	Station Blackout
SCIV	Secondary Containment Isolation Valve
SDP	Significance Determination Process
SBGT	Standby Gas Treatment
SSC	Structures, Systems, and Components
SR	Surveillance Requirement
SRA	Senior Reactor Analyst
SRO	Senior Reactor Operator
TCCP	Temporary Configuration Change Package
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
V	Volts
VDC	Volts Direct Current
WO	Work Order