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

May 9, 2008

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

SUBJECT: DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3 PROBLEM IDENTIFICATION AND RESOLUTION INSPECTION 05000237/2008008; 05000249/2008008

Dear Mr. Pardee:

On March 28, 2008, the U.S. Nuclear Regulatory Commission (NRC) completed a Problem Identification and Resolution team inspection at your Dresden Nuclear Power Station, Units 2 and 3. The enclosed report documents the inspection findings, which were discussed on March 28, 2008, with Mr. David Wozniak and other members of your staff.

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

Based on the results of this inspection, the team concluded that in general, problems were properly identified, evaluated, and corrected. Two green NRC-identified findings of very low safety significance related to motor-operated valve (MOV) testing and the evaluation of a previously NRC-identified issue were identified by the inspection team. The first finding involved the failure to have adequate acceptance criteria in the MOV diagnostic test procedure. The second finding related to the failure to develop a pre-fire plan for a zone that contained safety-related equipment. Each finding involved a violation of NRC requirements. However, because of their very low safety significance, and because the issues were entered into your corrective action program, the NRC is treating each of the issues as Non-Cited Violation (NCV) in accordance with Section VI.A.1 of the NRC Enforcement Policy.

If you contest the subject or severity of an NCV, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Dresden Nuclear Power Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure 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), 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; 72-037
License Nos. DPR-19; DPR-25

Enclosure: Inspection Report 05000237/2008008; 05000249/2008008
w/Attachment: Supplemental Information

cc w/encl: Site Vice President - Dresden Nuclear Power Station
Plant Manager - Dresden Nuclear Power Station
Regulatory Assurance Manager – Dresden Nuclear Power Station
Chief Operating Officer and Senior Vice President
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
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State Liaison Officer
Illinois Emergency Management Agency
Chairman, Illinois Commerce Commission

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Vice President - Licensing and Regulatory Affairs
Director - Licensing and Regulatory Affairs
Manager Licensing - Clinton, Dresden, and Quad Cities
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Letter to C. Pardee from M. Ring dated May 9, 2008

SUBJECT: DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3 PROBLEM IDENTIFICATION AND RESOLUTION INSPECTION 05000237/2008008; 05000249/2008008

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-237; 50-249
License Nos:	DPR-19; DPR-25
Report No:	05000237/2008008; 05000249/2008008
Licensee:	Exelon Generation Company, LLC
Facility:	Dresden Nuclear Power Station, Units 2 and 3
Location:	Morris, IL
Dates:	March 10 – March 28, 2008
Inspectors:	A. Barker, Project Engineer, Team Lead D. Melendez, Resident Inspector, Dresden A. Dahbur, Senior Reactor Engineer D. Schrum, Reactor Engineer
Approved by:	Mark A. Ring, Chief Branch 1 Division of Reactor Projects

Enclosure

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SUMMARY OF FINDINGS

IR 05000237/2008008, 05000249/2008008; 03/10/2008 – 03/28/2008; Dresden Nuclear Power Station, Units 2 & 3; Problem Identification and Resolution.

This inspection was performed by three NRC regional inspectors and the Dresden resident inspector. Two findings of very low safety significance (Green) were identified during this inspection. Each of the findings was classified as a Non-Cited Violation (NCV). The significance of most findings is indicated by their color (Green, White, Yellow, Red) using NRC Inspection Manual Chapter (IMC) 0609, “Significance Determination Process” (SDP). 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.

Problem Identification and Resolution

On the basis of the sample selected for review, the team concluded that implementation of the corrective action program (CAP) at Dresden was generally good. The licensee had a low threshold for identifying problems and entering them in the CAP. Items entered into the CAP were screened and prioritized in a timely manner using established criteria; were properly evaluated commensurate with their safety significance; and corrective actions were generally implemented in a timely manner, commensurate with the safety significance. The team noted that the licensee reviewed operating experience for applicability to station activities. Audits and self-assessments were determined to be performed at an appropriate level to identify deficiencies. On the basis of interviews conducted during the inspection, workers at the site expressed freedom to enter safety concerns into the CAP. There were two Green findings identified by the team during the inspection. The first finding involved the failure to have adequate acceptance criteria in the motor-operated valve (MOV) diagnostic test procedure. The second finding related to the failure to develop a pre-fire plan for a zone that contained safety-related equipment. The second finding had a cross-cutting aspect in the area of Problem Identification and Resolution.

A. Inspector-Identified and Self-Revealed Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified an NCV of 10 CFR Part 50, Appendix B, Criterion XI, “Test Control,” for failure to include an uncertainty value, to account for test equipment accuracy and lubricant degradation, in the acceptance criteria for the stem factor in the diagnostic test of MOV 3-2301-3, “Unit 3 high pressure coolant injection (HPCI) steam admission valve.” The stem factor is used to calculate the coefficient of friction (COF) to determine the predicted stroke opening time of the MOV under design basis conditions. Corrective actions for this issue included a re-calculation of the stem factor to account for instrument accuracy and lubricant degradation and an evaluation for guidance to be added to the test procedure.

  This finding was more than minor because, if the finding was left uncorrected it would become a more significant safety concern. Specifically, the acceptance criteria specified in the diagnostic test for the stem factor did not assure that MOV 3-2301-3 would meet its design stroke time value to open in less than or equal to 30 seconds. The finding was of very low safety significance based on a Phase 1 screening in accordance with IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for

At-Power Situations," because the finding did not result in an actual loss of a safety function. (Section 4OA2.a).

- Green. The inspectors identified an NCV for the licensee's failure to develop a pre-fire plan for fire zone 18.6. The finding was a violation of Dresden Nuclear Power Station Renewed Operating License. License conditions 2.E and 3.G for Unit 2 and Unit 3, respectively, of the Dresden Nuclear Power Station Renewed Facility Operating Licenses state, 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 (UFSAR) or the facility...." Pre-fire plans are described in the UFSAR as "provided for all safety-related areas of the plant." Corrective actions by the licensee 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. 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. 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. The inspectors determined that this issue also affected the cross-cutting area of problem identification and resolution, CAP aspect P.1(c) because the licensee failed to thoroughly evaluate a problem previously identified by NCV 05000237/2006011-01; 05000249/2006011-01, "Licensee's failure to develop a pre-fire plan for fire zone 8.2.6.A, elevation 534'," such that the resolution did not fully address causes and extent of condition. (Section 4OA2.a).

B. Licensee-Identified Violations

No findings of significance were identified.

REPORT DETAILS

4. OTHER ACTIVITIES

4OA2 Problem Identification and Resolution (71152B)

This inspection constitutes one biennial sample of problem identification and resolution as defined in Inspection Procedure 71152.

a. Assessment of the Corrective Action Program Effectiveness

(1) Inspection Scope

The inspectors reviewed the licensee's corrective action program (CAP) implementing procedures and attended CAP meetings to assess the implementation of the CAP by station staff. The inspectors reviewed risk and safety significant issues in the licensee's CAP since the last NRC problem identification and resolution inspection conducted in January 2006. The selection of issues ensured an adequate review of issues across NRC cornerstones. The inspectors reviewed issues identified through NRC generic communications, self-assessments, licensee audits, operating experience reports, and NRC documented findings as sources to select issues. In addition, the inspectors reviewed issue reports (IRs) and a selection of completed CAP documents from the licensee's investigative methods, such as, root cause investigations, and apparent cause and equipment apparent cause evaluations. Specifically, the inspectors determined if the station staff was identifying plant issues at the proper threshold, entering the plant issues into the station's CAP in a timely manner, and assigning the appropriate prioritization for resolution of the issues. The inspectors also reviewed the effectiveness of corrective action for selected issue reports, completed investigations, and NRC findings, including NCVs.

The inspectors selected instrument air and high pressure coolant injection (HPCI) as the high risk systems to review in detail. The inspectors' review was to determine whether the station staff was properly monitoring and evaluating the performance of the systems through effective implementation of station monitoring programs. The 5-year review of the instrument air system involved a review to determine if the CAP was being implemented to identify and resolve issues relating to system degradation. The 5-year review of the HPCI system focused on issue reports that involved long-standing corrective action.

(2) Assessment

Identification of Issues

The inspectors concluded, in general, that the station identified issues and entered them into the CAP at the appropriate level. The inspectors' review of operating experience reports identified that the licensee was appropriately including the issues in the CAP. The licensee also used the CAP to document instances where previous corrective action was ineffective or inappropriately closed.

The inspectors conducted an expanded 5-year review of the instrument air system. The inspectors' review determined that the CAP was being implemented to identify and resolve issues through established station monitoring programs and corrective action. The review sample selected was approximately 110 IRs over the 5 year period. The IRs reviewed documented equipment issues, such as compressor and dryer failures, and also included adverse trends in equipment reliability from system engineering monitoring. Specifically, documented in the March 10, 2008, Management Review Committee (MRC) Meeting Agenda was IR 745080 that identified the reliability criteria for the maintenance rule function of the instrument air system had been exceeded. The Station Ownership Committee (SOC) determined two action tracking items from their review of IR 745080. The action tracking items directed plant engineering to prepare and present an (a)(1) classification determination of the instrument air system to the maintenance rule expert panel.

The inspectors also conducted an expanded 5-year review of the HPCI system. This review resulted in a finding that describes the failure to include an uncertainty value, to account for test equipment accuracy and lubricant degradation, in the acceptance criteria for the stem factor for motor-operated valve testing. This specific finding is listed in section (3) as NCV 05000237/2008008-01; 05000249/2008008-01.

Prioritization and Evaluation of Issues

The inspectors' observations of the SOC concluded that the committee consistently reviewed, when applicable, the initial operability basis that was concluded by on-shift operations staff for clarity and its appropriateness of content. For some issues, the committee had questions on the narrative detail of the operability basis. For these issues, the SOC returned the action request back to the on-shift operations staff with their questions. This consistent committee behavior was typically directed by the committee's chairing member. In addition, the SOC directed member follow-up of issues that required additional information so the committee could perform its function. The inspectors concluded that none of the issues that were assigned the additional follow-up resulted in an inappropriate prioritization of the issue based on significance. Examples of SOC action taken were to assign work requests, evaluations, and/or corrective action to specific departmental groups. The inspectors also observed the MRC function in an oversight role of the SOC. For example, the MRC changed the SOC recommended action of some issues based on committee dialogue and additional station awareness of the issue. Also, the MRC performed grading of investigative CAP products to provide feedback on product quality to the sponsoring manager.

The issue reports that were observed being reviewed by the SOC were also observed being reviewed by the MRC in their oversight role. The MRC member dialogue in the review of several CAP investigative documents was informative, and provided feedback to the staff on the appropriate implementation of the CAP. The inspectors concluded that issues were properly prioritized and generally evaluated well. However, the inspectors developed an observation regarding a narrowly focused equipment apparent cause evaluation (EACE). This degraded condition was also previously identified by Licensee Event Report (LER) 237/2006-003-00 in July 2006. The following paragraph provides the observation.

Evaluation Narrowly Focused

On February 16, 2007, the reactor steam dome pressure low permissive pressure switch 2-0263-52B, circuit 1 as-found value for the decreasing trip was out of tolerance low for its technical specification allowable value. This degraded condition was documented by IR 592439. This pressure switch had displayed historical out of tolerance conditions requiring LER 237/2006-003-00 to be submitted to document that the switch had been historically inoperable. Assignment #3 of IR 592439 was to perform an EACE. The inspectors' review of the EACE determined that the procedural guidance of LS-AA-125-1003, "Apparent Cause Evaluation Manual," Attachment 6 questions to address when completing an EACE was narrowly investigated. Such as, "are the operating procedures or practices for this component inappropriate or unacceptable?" The EACE documented, "No. Operations does not operate this pressure switch. Therefore, no operating procedures are involved with these events." In addition, the inspectors' analysis of the pressure switch failure history appeared to identify that during the colder months of the year switch performance as-found values degraded at a higher rate over this time period. This colder month analysis perspective was not identified in the EACE for, "has system/component monitoring been deficient in identifying normal or abnormal equipment degradation?" The EACE focused on the results determined from surveillance testing and the change in the predefined surveillance frequency. The licensee planned to conduct a root cause investigation in April 2008, to further expand the investigative effort on this pressure switch.

Effectiveness of Corrective Action

In general, the licensee corrective actions for the samples reviewed were appropriate, and appeared to have been effective. The inspectors determined that the licensee generated issues reports when a corrective action was identified which was either inadequate or inappropriate. The inspectors also reviewed selected NRC findings for the past two years. From the review of NCV 2006011-01, "Licensee's Failure to Develop a Pre-fire Plan for Fire Zone 8.2.6.A, Elevation 534'," a similar issue was identified for a different room location. The identification of this new room location was by the inspectors while performing an extent of condition review of the 2006 finding. The specific finding that describes the new room location is listed in section (3) as the failure to develop a pre-fire plan for fire zone 18.6, NCV 05000237/2008008-02; 05000249/2008008-02.

(3) Findings

The Acceptance Criteria for Stem Factor in MOV Testing Did Not Account for Uncertainty

Introduction: The inspectors identified a NCV of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," having very low safety significance (Green) involving the licensee's failure to have adequate acceptance criteria in the MOV diagnostic test procedure. Specifically, the licensee failed to include an uncertainty value, to account for test equipment accuracy and lubricant degradation, in the acceptance criteria for the stem factor for the diagnostic test of MOV 3-2301-3, "Unit 3 high pressure coolant injection (HPCI) steam admission valve."

Description: In response to industry concerns about direct current (DC) MOVs, a method to predict the stroke time and functional capability of DC motors was developed by the boiling water reactor owners group (BWROG) for use in the industry. The methodology, known as the DC motor methodology, was accepted by Exelon, and was designated an NRC commitment. A calculation was completed which showed that eleven MOVs, with valve designations of 2(3)-0220-2, 2(3)-1201-2, 3-2301-5, 2(3)-2301-3, 2(3)-2301-35 and 2(3)-2301-36 had stroke times that potentially exceeded their design basis. These deficiencies for the above MOVs were documented by IR 159922 that was issued on May 21, 2003.

As an example, the calculation showed for MOV 2-2301-3, Unit 2 HPCI Steam Admission Valve, the predicted stroke time to fully open was 34.9 seconds which exceeded its design basis value of 30 seconds. Similarly, for Unit 3 HPCI MOV 3-2301-3, the predicted stroke time to open was 34.1 seconds which also exceeded its design basis value of 30 seconds.

Upon discovery in 2003, the licensee evaluated the operability of the eleven MOVs using the lowest battery voltage for each unit, and the tested coefficient of friction plus the normal degradation margin. The licensee concluded that each valve would fully stroke within its design basis values.

The licensee indicated in action item 17/1 of IR 159922, the stroke opening time of the HPCI 2(3)2301-3 steam admission valves was dictated by the Dresden loss of coolant accident (LOCA) analysis which assumed that the HPCI turbine steam flow was delivered within 30 seconds. Because the HPCI valve had been demonstrated to deliver in excess of rated turbine steam flow when 50% open, the licensee determined that it was only critical that the 2(3)-2301-3 valves be at least 50 percent open in 30 seconds to satisfy the LOCA analysis. The action item recommended re-evaluation of the LOCA analysis if a further increase in stroke time was required. However, action item 17/4 was completed in February 2008, and concluded that for the HPCI steam admission valves 2(3)-2301-3, an increase in the allowable stroke open time from 30 seconds to 30 seconds at 50 percent open may affect other HPCI model parameters such as the time it takes for the turbine to reach rated speed and the time to establish sufficient pressure in the HPCI discharge line. Therefore, action item 17/5 was initiated on February 27, 2008, to develop an alternate plan to recover DC motor stroke time operating margin for HPCI MOVs 2(3)-2301-3.

As discussed above, the licensee indicated that the two HPCI MOVs 2(3)-2301-3 were operable, and that calculations showed that the open stroke time for MOV 2-2301-3 was 29.9 seconds at a stem friction coefficient (COF) value of 0.127, based on a value obtained during a 1998 test. Similarly, the open stroke time for MOV 3-2301-3 was 29.8 seconds at a COF value of 0.125, based on a value obtained during a 1999 test. The licensee used the test values for the COF instead of the industry bounding value recommended by Electrical Power Research Institute (EPRI) of 0.2 because prior calculations using the 0.2 value resulted in exceeding the stroke time design values for the MOVs. The practice of using MOV diagnostic testing to calculate a specific valve COF is an approved methodology in the Exelon Nuclear MOV program.

The inspectors' review of the latest diagnostic test for MOV 3-2301-3, which was performed in December 2007, resulted in questions on the acceptance criteria value for the stem factor of less than 0.0182 which corresponded to a calculated COF value of 0.13. The COF value of 0.13, when applied, determined that the open stoke time for the MOV was less than 30 seconds. The stem factor value of 0.0182 did not account for test equipment accuracy or lubricant degradation.

The licensee performed a preliminary calculation in response to the inspectors' concern. This calculation showed that the COF value for MOV 3-2301-3 remained less than 0.13 when they added the test equipment accuracy and lubricant degradation as uncertainty to the test value. The calculated COF value including the uncertainty was 0.128. Based on the original test value, the licensee had a margin of 0.02 (0.13 minus 0.11), however, this margin was significantly decreased to 0.002 when the inspectors' identified uncertainty was added to the test value (0.13 minus 0.128). A similar calculation was performed for MOV 2-2301-3 that showed similar results. The inspectors concluded that the acceptance criteria specified in the diagnostic test procedure did not assure that the stroke time of the MOVs would not exceed their design basis value. Upon discovery, the licensee generated IR 754896 and recommended actions to evaluate the need to provide additional guidance in procedure ER-AA-302-1008, "MOV Diagnostic Test Preparation Instructions," to ensure that the acceptance criteria for the COF (i.e. stem factor) value would include uncertainty.

Analysis: The inspectors determined that the licensee's failure to include uncertainty in the acceptance criteria for the stem factor in the MOV diagnostic test procedure, specifically for the HPCI MOV 3-2301-3, was a performance deficiency warranting a significance evaluation. The inspectors further determined that the issue was within the licensee's ability to foresee and correct, and that it could have been prevented because the licensee had performed the diagnostic test of MOV 3-2301-3 in December 2007.

The inspectors determined that the performance deficiency was more than minor in accordance with IMC 0612, Appendix B, "Issue Disposition Screening," issued on September 20, 2007, because if the finding was left uncorrected it would become a more significant safety concern. Specifically, the acceptance criteria specified in the diagnostic test for the stem factor did not assure that MOV 3-2301-3 would meet its design stroke time value for opening in less than or equal to 30 seconds. When the uncertainty value for test equipment accuracy and lubricant degradation was added to the COF value of 0.11, which was obtained during the last test performed in December 2007, the new calculated value was 0.128. The new calculated value substantially decreased the available margin to the acceptance limit of 0.13.

The inspectors screened the finding using IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," Phase 1 screening. The inspectors answered "No" to all the screening questions in the Mitigating Systems column because the finding did not result in an actual loss of a safety function. The new calculated COF value 0.128, which included the uncertainty value, did not exceed the maximum allowable value of 0.13. Therefore, the finding was determined to have very low safety significance (Green).

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," requires, in part, that a test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents.

Contrary to the above, on December 2007, the licensee's MOV diagnostic test for MOV 3-2301-3 failed to have adequate acceptance criteria for MOV stem factor. Specifically, the stem factor value of 0.0182 for MOV 3-2301-3 did not account for uncertainty for test equipment accuracy and lubricant degradation. This value did not assure that the predicted stoke opening time for the MOV would not exceed its design basis value of 30 seconds. However, because this violation was of very low safety significance and because the issue was entered into the licensee's CAP as IR 754896, this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy. **(NCV 05000237/2008008-01; 05000249/2008008-01)**

Failure to Develop a Pre-Fire Plan for Fire Zone 18.6

Introduction: The inspectors identified a 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 March 14, 2008, the inspectors reviewed the licensee's corrective action for NCV 05000237/2006011-01; 05000249/2006011-01, "Licensee's failure to develop a pre-fire plan for fire zone 8.2.6.A, elevation 534'." The inspectors could not locate pre-fire plans associated with each fire zone listed in the fire hazard analysis. The inspectors questioned the station fire marshal to determine if fire zone 18.6 had a pre-fire plan. The station fire marshal searched the pre-fire plan database and could not find one specifically addressing fire zone 18.6. 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 kept on a float charge.

As noted in the Enforcement section, pre-fire plans shall be developed for all safety-related areas and areas representing a hazard to safety-related equipment. The inspectors determined that the Unit 2 125V alternate batteries were safety-related equipment; therefore, fire zone 18.6 should have had a pre-fire plan. The licensee generated IR 750656 to address this issue.

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 IMC 0612, Appendix B, "Issue Screening," issued on September 20, 2007, 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, CAP aspect P.1(c) because the licensee failed to thoroughly evaluate a problem previously identified by NCV 05000237/2006011-01; 05000249/2006011-01, "Licensee's failure to develop a pre-fire plan for fire zone 8.2.6.A, elevation 534'," such that the resolution did not fully address causes and extent of condition.

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 state, 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 (UFSAR) 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 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, “Pre-Fire Plans,” Revision 2, 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. The licensee generated IR 750656 to address this issue. Corrective action 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 an NCV, consistent with Section VI.A.1 or the NRC Enforcement Policy. **(NCV 05000237/2008008-02; 05000249/2008008-02)**

b. Assessment of the Use of Operating Experience

(1) Inspection Scope

The inspectors reviewed the licensee's program for handling operating experience (OPEX). Specifically, the inspectors reviewed the implementing procedure and attended CAP meetings to observe the use of OPEX. The inspectors also reviewed selected OPEX documents, such as 10 CFR Part 21 reports, NRC information notices, NRC regulatory issue summaries, and other generic correspondence to determine if the program had adequately assessed issues for station applicability. The documents were reviewed to ensure that underlying problems associated with the OPEX issues were appropriately considered for corrective action. The team also reviewed how the licensee considered OPEX for applicability in CAP investigations.

(2) Assessment

No findings of significance were identified.

In general, OPEX information was being well utilized at the station. The inspectors observed that Exelon fleet internal OPEX and industry OPEX were discussed by licensee staff to support review activities and CAP investigations. During licensee staff interviews, the inspectors identified that the use of OPEX was being considered during daily activities.

Through IR 588143 the licensee evaluated the applicability of NRC Information Notice 2006-31, "Inadequate Fault Interrupting Rating of Breakers," for the station. The licensee determined that the issue did not apply to the station. The station currently uses the ETAP computer program to evaluate the AC auxiliary power system. However, the inspectors identified that IR 588143 was initiated in February 2007, but the action was not completed until October 2007. Procedure LS-AA-115, "Operating Experience Procedure," required a review to be completed within 120 days. The licensee communicated that this deficiency was identified in September 2007, and IR 674646 was generated to address the issue. The licensee's corrective action was to revise procedure LS-AA-115 to include a requirement that corporate NOS programs will assign the OPEX document within 14 days of receipt.

The inspectors developed a sample population of thirteen NRC regulatory issue summaries from 2005 through 2007, to verify if the licensee had reviewed the OPEX documents for station applicability. From the inspectors' review, an observation was determined regarding an OPEX document that had not been reviewed. The following paragraph provides this observation.

Regulatory Issue Summary 2006-23, "Post-Tornado Operability of Ventilating and Air-Conditioning Systems Housed in Emergency Diesel Generator Rooms"

In September 2007, IR 170146, assignment 33, revised procedure LS-AA-115, Attachment 4, "OPEX document list/classification," to require NRC regulatory issue summaries to have formal reviews conducted. The inspectors identified from the review of the developed sample that Regulatory Issue Summary 2006-23 was not reviewed for applicability by either the engineering staff or through the OPEX program. The licensee generated IR 758258 to address this issue. This issue report is a corporate CAP document that will also determine the OPEX value to multiple sites in conducting a historical review of past regulatory issue summaries.

c. Assessment of Self-Assessments and Audits

(1) Inspection Scope

The inspectors reviewed selected focused area self-assessments (FASA), check-in self-assessments, and Nuclear Oversight (NOS) audits of the corrective action program, materials management and procurement engineering, and the functional areas of operations and maintenance. The inspectors evaluated whether these audits and self-assessments were being effectively managed, were adequately covering the subject areas, and were properly capturing identified issues in the CAP. In addition, the inspectors also interviewed licensee staff regarding the implementation of the audit and self-assessment programs.

(2) Assessment

No findings of significance were identified.

The inspectors concluded that the self-assessments and NOS audits were generally critical and probing. Multi-discipline teams were utilized, when appropriate, to gain a broad perspective. The use of OPEX supported team preparations and scope development of the NOS audits. There were a number of deficiencies and recommendations identified across the spectrum of performance, including issues of improper CAP implementation. As appropriate, the self-assessment and NOS audit deficiencies were documented in the CAP.

The licensee completed FASA 700259 on the corrective action program in February 2008. The CAP manager from LaSalle Nuclear Power Station was included as a team member. The self-assessment was comprehensive in scope, resulting in 17 standards deficiencies being identified. The standards deficiencies, such as miscoding of work priorities, corrective action not designated or completed, and not documenting OPEX searches, were entered into the CAP, along with the other deficiencies that were identified.

The licensee completed NOS Audit NOSA-DRE-07-07 on the operations functional area in October 24, 2007. One audit deficiency identified that material deficiencies and potential operator challenges were not identified during the performance of non-licensed operator rounds. The potential operator challenges pertained to the need to perform manual actions, such as, operation of cooling towers, periodic cleaning of circulating water pump bowl drains due to zebra mussels and alum addition at the wastewater treatment facility. This deficiency was identified through the in-field observation of operator rounds and was entered in the CAP through IR 686417 and 685953.

d. Assessment of Safety Conscious Work Environment

(1) Inspection Scope

The inspectors interviewed selected members of the Dresden station staff to determine if there were any impediments to the establishment of a safety conscious work environment. In addition, the inspectors discussed the implementation of the Employee Concerns Program (ECP) with the ECP Coordinators, and reviewed their 2007 activities to identify any emergent issues or potential trends. Licensee programs to publicize the CAP and ECP programs were also reviewed.

(2) Assessment

No findings of significance were identified.

The inspectors determined that the conditions at the Dresden station were conducive to identifying issues. The staff was aware of and generally familiar with the CAP and other station processes, including the ECP, through which concerns could be raised. Staff interviews identified that issues could be freely communicated to supervision, and that several of the individuals interviewed had previously initiated issue reports. In addition, a review of the types of issues in the ECP indicated that site personnel were appropriately using the corrective action and employee concerns programs to identify issues. The inspectors interviewed the ECP Coordinators and concluded that the individuals were focused on ensuring all site individuals were aware of the program, comprehensive in their review of individual concerns, and used the corrective action and employee concerns programs to appropriately resolve issues.

4OA6 Management Meetings

Exit Meeting Summary

The inspectors presented the inspection results to Mr. Wozniak and other members of the Dresden staff at an exit meeting on March 28, 2008. Mr. Wozniak acknowledged the findings presented. The two proprietary informational documents that the team had reviewed were returned to the licensee on the day of the exit.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

D. Wozniak, Site Vice President
C. Barajas, Operations Director
H. Bush, Radiation Protection Manager
J. Ellis, Regulatory Assurance Manager
R. Fenili, Regulatory Assurance
D. Galanis, Design Engineering Manager
J. Griffin, Regulatory Assurance - NRC Coordinator
L. Jordan, Training Director
P. Karaba, Maintenance Director
D. Leggett, Nuclear Oversight Manager
P. Markoo, Chemistry
G. Papanic, Regulatory Assurance
M. Pavey, Radiation Protection
F. Pournia, System Engineer
R. Rybak, Regulatory Assurance
P. Salgado, Operations
J. Sipek, Engineering Director
J. Strmec, Chemistry, Environmental and Radwaste Manager
J. Walsh, Work Control

Nuclear Regulatory Commission

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

IEMA

R. Schulz, Illinois Emergency Management Agency

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000237/2008008-01; 05000249/2008008-01	NCV	Acceptance Criteria for Stem Factor in MOV Testing Did Not Account for Uncertainty
05000237/2008008-02; 05000249/2008008-02	NCV	Failure to Develop a Pre-Fire Plan for Fire Zone 18.6

Closed

05000237/2008008-01; 05000249/2008008-01	NCV	Acceptance Criteria for Stem Factor in MOV Testing Did Not Account for Uncertainty
05000237/2008008-02; 05000249/2008008-02	NCV	Failure to Develop a Pre-Fire Plan for Fire Zone 18.6

LIST OF DOCUMENTS REVIEWED

The following is a list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety, but rather, that selected sections of 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.

Previous NRC Findings

NCV 05000237/2007006-03; 05000249/2007006-03; Failure to Procedurally Control Regulatory Guide 1.97 Control Board Labeling
IR 629141; Reg Guide 1.97 Labels not Designated as such in Simulator; May 14, 2007
IR 628000; Labeling and Passport Discrepancies and Enhancements; May 10, 2007
IR 622767; Potential Enhancement to DAP 07-16 and Passport Data; April 27, 2007
DAP 07-16; Control of Plant Labeling and Annunciator Tiles and Unauthorized Operator Aids; Revision 17

NCV 05000237/2006011-01; 05000249/2006011-01; Licensee's Failure to Develop a Pre-fire Plan for Fire Zone 8.2.6.A, Elevation 534'
IR 559569; NRC Inspector Questioned Station Fire Pre-Plans; November 17, 2006
IR 750656; NRC PI&R Question on Fire Pre-Plans; March 17, 2008
IR 751140; NRC ID – FHA Fire Zone Description Requires Clarification; March 18, 2008
OP-AA-201-008; Pre-Fire Plan Manual; Revision 2

NCV 05000237/2006007-02; Unit 2 350 psig Reactor Low Pressure Emergency Core Cooling System Permissive Switch Out-of-Tolerance during Surveillance Testing
LER237/2006-003-00; Unit 2 Reactor Steam Dome Pressure-Low Permissive Switch Determined to Have Been Historically Inoperable
DIS 1500-01; Reactor Low Pressure (350 psig) ECCS Permissive; Revision 24
IR 218197; Rx Low Press ECCS Permissive Switch Found Out of Cal; May 1, 2004
IR 291533; PS 2-263-52B As Found Out of Tolerance – Tech Spec Violation; January 18, 2005
IR 354800; PS 2-0263-52B ECCS Permissive Tech Spec Violation; July 20, 2005
IR 354804; PS 2-0263-52B ECCS Perm. Indication Reading 20# High; July 20, 2005
IR 387131; 2-263-52B Found OOT Tech Spec; October 17, 2005
IR 443904; PS 2-263-52B Found Out of Tolerance, Violation of Tech Spec; January 19, 2006
IR 480479; ECCS Permissive PS 2-263-52B Out of Tech Spec Tolerance; April 19, 2006
IR 495327; 2-0263-52B Exceeds TS 6 of 9 Surveillances (Req LER); May 31, 2006
IR 496150; PS 2-0263-52B Circuit 1 Difficult to Adjust; June 2, 2006
IR 496177; Differential Pressure Switch Procured From Stores; June 2, 2006
IR 571094; PS Found Out of Tolerance (TS Violation); December 19, 2006
IR 592439; As Found Out of Tolerance; Tech Spec OOT; February 16, 2007
IR 712731; Document Shift Review of Power Labs Results; December 17, 2007
IR 713459; PS 2-0263-52B Found Out of Tech Spec during DIS 1500-01; December 19, 2007
IR 735961; Found Trip Point Out of As Found Tolerance for PS 2-263-52B; February 14, 2008
WO 885299; D2 QTR TS Reactor Low Pressure (350 psig) ECCS Permissive Cal
WO 913856; D2 QTR TS Reactor Low Pressure (350 psig) ECCS Permissive Cal
WO 927706; D2 QTR TS Reactor Low Pressure (350 psig) ECCS Permissive Cal
WO 939932; D2 1M TS Reactor Low Pressure (350 psig) ECCS Permissive Cal
WO 947076; D2 1M TS Reactor Low Pressure (350 psig) ECCS Permissive Cal
WO 959070; D2 1M TS Reactor Low Pressure (350 psig) ECCS Permissive Cal

WO 967340; D2 1M TS Reactor Low Pressure (350 psig) ECCS Permissive Cal
WO 975986; D2 1M TS Reactor Low Pressure (350 psig) ECCS Permissive Cal
WO 986314; D2 1M TS Reactor Low Pressure (350 psig) ECCS Permissive Cal
WO 992152; D2 1M TS Reactor Low Pressure (350 psig) ECCS Permissive Cal
WO 1002328; D2 1M TS Reactor Low Pressure (350 psig) ECCS Permissive Cal
WO 1012311; D2 1M TS Reactor Low Pressure (350 psig) ECCS Permissive Cal
WO 1021806; D2 1M TS Reactor Low Pressure (350 psig) ECCS Permissive Cal
WO 1030418; D2 1M TS Reactor Low Pressure (350 psig) ECCS Permissive Cal
WO 1040818; D2 1M TS Reactor Low Pressure (350 psig) ECCS Permissive Cal
WO 1047721; D2 1M TS Reactor Low Pressure (350 psig) ECCS Permissive Cal
WO 1063766; D2 1M TS Reactor Low Pressure (350 psig) ECCS Permissive Cal
WO 1071558; D2 1M TS Reactor Low Pressure (350 psig) ECCS Permissive Cal
WO 1083643; D2 1M TS Reactor Low Pressure (350 psig) ECCS Permissive Cal
WO 1091025; D2 1M TS Reactor Low Pressure (350 psig) ECCS Permissive Cal
WO 1097164; D2 1M TS Reactor Low Pressure (350 psig) ECCS Permissive Cal

FIN 05000237/2006002-01; 05000249/2006002-01; Identification of Electromatic Relief Valve Degradation
IR 443829; NRC/IEMA Inspectors Question No IR on ERV As-Found Condition; January 19, 2006
IR 561385; 3-0203-3C ERV; November 22, 2006
IR 561430; 3-0203-3B ERV Trending IR; November 22, 2006
MA-DR-ME-4-020046; Electromatic Relief Valve Replacement and Actuator Maintenance; Revision 8

Procedures

EI-AA-1; Safety Conscious Work Environment; Revision 2
EI-AA-101; Employee Concerns Program; Revision 7
EI-AA-100-1003; Employee Issues Advisory Committee Notification; Revision 0
EI-AA-101-1001; Employee Concerns Program Process; Revision 6
EI-AA-101-1002; Employee Concerns Program Trending Tool; Revision 4
LS-AA-1006; NRC Cross-Cutting Analysis and Trending; Revision 1
LS-AA-115; Operating Experience Procedure; Revision 10
LS-AA-120, Issue Identification and Screening Process, Revision 7
LS-AA-125, Corrective Action Program Procedure, Revision 11
LS-AA-125-1001; Root Cause Analysis Manual; Revision 6
LS-AA-125-1002; Common Cause Analysis Manual; Revision 5
LS-AA-125-1003; Apparent Cause Evaluation Manual; Revision 7
LS-AA-125-1004; Effectiveness Review Manual; Revision 2
LS-AA-126; Self-Assessment Program; Revision 5
LS-AA-126-1001; Focused Area Self-Assessments; Revision 4
LS-AA-126-1005; Check-In Self-Assessments; Revision 3
OP-AA-102-102; General Area Checks and Operator Field Rounds; Revision 5
OP-AA-108-115; Operability Determinations; Revision 5
OP-AA-106-101-1006; Operational and Technical Decision Making Process; Revision 5
LS-AA-125-1005; Coding and Analysis Manual; Revision 5
CC-AA-309-1012; 10 CFR Part 21 Technical Evaluations; Revision 1
WC-AA-101; On-Line Work Control Process; Revision 14
HU-AA-1212; Technical Task Risk/Rigor Assessment, Pre-Job Brief, Independent Third Party Review, and Post-Job Brief; Revision 2

HU-AA-101; Human Performance Tools and Verification Practices; Revision 4
HU-AA-102; Technical Human Performance Practices; Revision 1
HU-AA-104-101; Procedure Use and Adherence; Revision 3
ER-AA-302-101; MOV Rising Stem Motor Operated Valve Thrust and Torque Sizing and Set-up Window Determination Methodology; Revision 5
ER-AA-302-1008; MOV Diagnostic Test Preparation Instructions; Revision 5
LS-AA-104-1000; Exelon 50.59 Resource Manual; Revision 4
BSA-D-97-01; LPCI Room Thermal Response for a Loss of Room Cooler Event; October 14, 2007
Unit 2 DIS 1300-02; Unit 2 Isolation Condenser Steam/Condensate Line High Flow Calibration; Revision 27

Issue Reports

487346; Replace the Bus 24-1 Degraded Voltage Time Delay Relay; May 5, 2006
139819; TEC Breaker Setting; January 16, 2003
674646; OPEX Assignment Not Created in Timely Manner; September 24, 2007
588143; NRC IN 2006-31 Inadequate Breaker Interrupting Rating; February 6, 2007
456209; NRC IN 2006-03 Cutler Hammer Starter Mechanical-Interlock Binding; February 20, 2006
700836; 125 VDC Ground Alarm and Other Alarms with Undetermined Cause; November 17, 2007
611853; OPEX – NRC IN 2007-07 Needs Applicability Review; April 2, 2007
633468; IN 2007-07 Requires DSSP Enhancement; May 24, 2007
593555; IN 2007-06 Potential Common Cause Vulnerabilities in ESW; February 20, 2007
692661; Unit 2 HPCI Aux Oil Pump Trip after Releasing Hand Switch; October 31, 2007
626623; IN 2007-09; Equipment Operability under Degraded Voltage; May 8, 2007
694690; IN 2007-34; Electrical Circuit Breakers; November 5, 2007
703489; IN 2007-36; EDG Voltage Regulator Problems; November 26, 2007
568613; MSL HI Flow Switch Found Out of Tech Spec Tolerance; December 13, 2006
529693; DPIS 2-261-2N found out of TS 3.3.6.1 Allowable Value; September 11, 2006
540063; Main Steam HI Flow Switch out of Tolerance; October 4, 2006
540786; NRC Questions Conclusion of Safety Evaluation 2005-02-001; August 24, 2005
550111; NRC Questions HPCI Repairs and Appendix R Requirements; October26, 2006
742760; NRC Questions on Post EPU LPCI Room Heatup Calculations; February 29, 2008
180661; HPCI Room Heat-up Calculation Reanalysis and Validation; October 18, 2007
175568; Unresolved Safety Question Relating to HPCI HELB; September 12, 2003
164464; Review of Operability Evaluation 03-010; October 30, 2003
559176; U3 West LPCI Room Cooler Fails Air Leak Test; November 16, 2006
557050; U3 West LPCI Room Cooler Tubing Exceeds Wall Loss Plug Criteria; November 13, 2006
741858; 2A LPCI Room Cooler Eddy Current Test Found Significant Tube Degradation; February 27, 2008
272561; Unit 3 East LPCI Room Cooler Eddy Current Testing Results; November 11, 2004
640627; Received LPCI/CS Corner Room High Temperature During LPCI Pump Run; June 15, 2007
571613; Unit 3 LPCI/CS Pump Area Temperature High – Cooler OOS; December 20, 2006
586491; 3A LPCI Room Cooler OOS No Work Scheduled; February 1, 2007
445791; NOS IDS No Complex Troubleshooting on B LPCI Cooler; January 25, 2006
528027; 3B LPCI Room Cooler Leak; September 6, 2006
242676; 3A LPCI Room Cooler Leak; August 8, 2004

668240; LPCI Room Cooler DP is High; September 5, 2007
180661; Loss of HPCI Room Cooler Fan on Appendix K and EQ; September 4, 2003
455107; OPEX Effectiveness Review Contained Errors; February 16, 2006
502521; OPEXR to Plant Engineering to Review OE 22757; September 13, 2006
685829; Bus 24-1 Voltage Excursion Search of OPEX for Similar Event; December 14, 2007
707529; Review of OE25748 for Applicability at Dresden; December 5, 2007
488815; Quad Cities OPEX Applicable at Dresden; April 13, 2006
599546; NOS IDS Documentation Issues in CDBI FASA; March 5, 2007
650559; Clarification Needed – SOER Recommendations; July 16, 2007
738615; Degraded Unit 3 CCSW Piping Identified; February 19, 2008
451700; Part 21 – Tyco WCSF – 300 Tubing Use Over Bolted Splice; February 9, 2006
450059; OPEX – Access to GE SILs; January 29, 2006
588143; NRC Information Notice 2006-31 – Inadequate Breaker Interrupting Rating; February 6, 2007
464241; Cathodic Protection System Related to Service Water System; March 9, 2006

Root and Apparent Cause Reports

Root Cause Investigation Report 506230; Dresden Unit 2 Scrammed due to Main Steam Isolation Valve (MSIV) Closure Resulting From a Failed Tube Connection on the Pneumatic Supply to the 1A MSIV Solenoid Manifold Pilot Block; July 31, 2006
Root Cause Investigation Report 555432; during the D3R19 Refuel Outage, the Unit 3B Electromatic Relief Valve Pilot Failed to Fully Open during Testing due to Binding Between the Pilot Spring Cap and the Solenoid Mounting Bracket; January 4, 2007
Root Cause Investigation Report 606618; Personnel Safety Near Miss during the Removal of a 480 Volt Breaker Bucket From Unit 1 Power Center 4A Resulting in a Phase to Ground Fault due to an Obsolete Design Standard; May 11, 2007
Root Cause Investigation Report 672183; 2/3 A Standby Gas Treatment Inoperable due to Flow Oscillations; October 26, 2007
Apparent Cause Report 514789; Mispositioning of Control Rod during Single Notch Timing; October 5, 2006
Apparent Cause Report 583628; Plant Engineering System Manager Deficiencies for the Documentation of System Walkdowns Performed per the Requirements of ER-AA-2030, Conduct of Plant Engineering Manual; March 15, 2007
Apparent Cause Report 637281; Near Miss Configuration Control Event during DOS 1600-05 PMT; July 17, 2007
Equipment Apparent Cause Report 670697; High Pressure Coolant injection Primary Containment Isolation Motor-Operated Valve 3-2301-4 Failed to Stroke On-Demand due to Failed Actuator Motor; December 12, 2007

Self-Assessments and NOS Audits

Focused Area Self Assessment 700259; Preparation for the NRC Problem Identification and Resolution (PI&R) Inspection; February 4, 2008
Focused Area Self Assessment SART 442050; Dresden Engineering Programmatic Implementation of the Equipment Reliability and Configuration Control Series Procedures; May 19, 2006
Focused Area Self Assessment 568187; Reactivity Management; October 16, 2007
Check-in Self Assessment 501427; Self-Assessment Program Review; March 23, 2007
NOSA-DRE-06-01; NOSA Audit - Maintenance Functional Area; March 3, 2006
NOSA-DRE-07-01; NOSA Audit - Corrective Action Program Audit Report; April 13, 2007

NOSA-DRE-07-02; NOSA Audit – Materials Management and Procurement Engineering; February 21, 2007
NOSA-DRE-07-07; NOSA Audit – Operations Functional Area; October 24, 2007

Issue Reports Generated for Inspection

749105; Passport Headers/Footers Not Printing From CAP Toolbox; March 13, 2008
750656; NRC PI&R Question on Fire Pre-Plans; March 17, 2008
751140; NRC ID – FHA Fire Zone Description Requires Clarification; March 18, 2008
754896; MOV Testing-Development of COF Acceptance Criteria; March 26, 2008
755394; NRC Identifies Potential Improvement in CAP on this Issue; March 27, 2008

Other

WO 1019346; MOV 3-2301-3 Diagnostic Test Instructions
WO 99225212; Unit 3 – Replace B Electromatic Relief Valve
NTS 237-251-92-0062; Evaluation of Basis for HPCI Room Normal Service and LOCA Temperatures and Impact of 150 Degrees Fahrenheit on the EQ Equipment Qualification; March 15, 1993
DRE01-0041; Updated EQ Zone Parameter Tables Following Implementation of Extended Power Uprate; September 24, 2007
RIS 2005-07; Compensatory Measures to Satisfy the Fire Protection Program Requirements; April 19, 2005
IN 97-48; Inadequate or Inappropriate Interim Fire Protection Compensatory Measures; July 9, 1997
EC351392; Reroute of 24 Inch HPCI Suction Line; Revision 2
EC351751; Operating Considerations for Unavailability of CST Suction to HPCI; Revision 1
50.59 Review; Crediting the Isolation Condenser and Control Rod Drive Systems for Mitigating a HPCI HELB; March 11, 2008
257HA353AB; High Pressure Injection System – Data Sheets; Revision 2
Maintenance Rule Performance Evaluation; January 29, 2008
Dresden Generated List of ARs (1/01/06-3/10/08) Pertaining to OPEX; March 10, 2008
HSBO-02; Isolation Condenser Area Average Temperature Following SBO; Revision 1
Corporate Generated OPEX List With Dresden Review Assigned; March 10, 2008
Instrument Air Piping Diagram M-367 for 3A and 3B Compressors (sheet numbers 1 through 7)
Unit 3 Service Air Piping Diagrams M-368 and M-368A
Station Ownership Committee Meeting Agenda for March 11, 2008
Station Ownership Committee Meeting Agenda for March 12, 2008
Station Ownership Committee Meeting Agenda for March 13, 2008
Station Ownership Committee Meeting Agenda for March 14, 2008
Station Ownership Committee Meeting Agenda for March 24, 2008
Management Review Committee Meeting Agenda for March 11, 2008
Management Review Committee Meeting Agenda for March 12, 2008
Management Review Committee Meeting Agenda for March 13, 2008
Management Review Committee Meeting Agenda for March 14, 2008
Management Review Committee Meeting Agenda for March 24, 2008
Management Review Committee Meeting Agenda for March 25, 2008
Management Review Committee Meeting Agenda for March 26, 2008

LIST OF ACRONYMS USED

AC	Alternating Current
BWROG	Boiling Water Reactor Owner Group
CAP	Corrective Action Program
CFR	Code of Federal Regulations
COF	Coefficient of Friction
DC	Direct Current
DRP	Division of Reactor Projects
EACE	Equipment Apparent Cause Evaluation
ECP	Employee Concerns Program
EPRI	Electric Power Research Institute
FASA	Focused Area Self-Assessment
HPCI	High Pressure Core Injection
IMC	Inspection Manual Chapter
IR	Issue Report
LOCA	Loss of Coolant Accident
MOV	Motor-Operated Valve
MRC	Management Review Committee
NCV	Non-Cited Violation
NFPA	National Fire Protection Association
NOS	Nuclear Oversight
NRC	U.S. Nuclear Regulatory Commission
OPEX	Operating Experience
PI&R	Problem Identification and Resolution
SDP	Significance Determination Process
SOC	Station Ownership Committee
UFSAR	Updated Final Safety Analysis Report
V	Volt