UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
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LISLE, IL 60532-4352

April 23, 2012

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

SUBJECT: DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3 EVALUATION OF CHANGES, TESTS, OR EXPERIMENTS AND PERMANENT PLANT MODIFICATIONS BASELINE INSPECTION REPORT 05000237/2012008(DRS); 05000249/2012008(DRS)

Dear Mr. Pacilio:

On March 27, 2012, the U. S. Nuclear Regulatory Commission (NRC) completed an Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications inspection at your Dresden Nuclear Power Station, Units 2 and 3. The enclosed report documents the inspection findings, which were discussed on March 9, 2012, with Mr. Dave Czufin and on March 27, 2012, with Mr. Paul Wojtkiewicz 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 NRC identified two findings of very low safety significance and one traditional enforcement Severity Level IV violation. The issues involved violations of NRC requirements. However, because of their very low safety significance, and because the issues were entered into your corrective action program, the NRC is treating the issue as a Non-Cited Violations (NCVs) in accordance with Section 2.3.2 of the NRC Enforcement Policy.

If you contest the subject or severity of any 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 addition, if you disagree with the cross-cutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region III, and the NRC Resident Inspector at the Dresden Nuclear Power Station.

In accordance with Title 10, Code of Federal Regulations (CFR), Part 50, Section 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC Website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Hironori Peterson, Acting Chief
Engineering Branch 3
Division of Reactor Safety

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

Enclosure: Inspection Report 05000237/2012008; 05000249/2012008
w/Attachment: Supplemental Information

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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:	05000237/2012008; 05000249/2012008
Licensee:	Exelon Generation Company, LLC
Facility:	Dresden Nuclear Power Station, Units 2 and 3
Location:	Morris, IL
Dates:	February 21 through March 27, 2012
Inspectors:	Z. Falevits, Senior Reactor Inspector (Lead) D. Szwarc, Reactor Inspector J. Bozga, Reactor Inspector
Approved by:	H. Peterson, Acting Chief Engineering Branch 3 Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000237/2012008(DRS); 05000249/2012008(DRS); 02/21/2012 – 03/27/2012; Dresden Nuclear Power Station, Units 2 and 3; Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications.

This report covers a two-week announced baseline inspection on evaluations of changes, tests, or experiments and permanent plant modifications. The inspection was conducted by Region III based engineering inspectors. The NRC identified two findings of very low safety significance and one traditional enforcement Severity Level IV violation. The issues 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). Cross-cutting aspects were determined using IMC 0310, “Components Within the Cross-Cutting Areas.” 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 finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” for the licensee’s failure to check the adequacy of design for flammable hydrogen gas bottles installed in the reactor building and their impact on safety-related structures, systems, and components (SSCs). Specifically, the licensee failed to evaluate how a failure of the flammable hydrogen gas bottles and the resulting fire or explosion at the installed locations could impact nearby safety-related SSCs. The licensee entered this issue into their corrective action program to review the placement of the flammable hydrogen gas bottles.

  The inspectors determined that the finding was more than minor because the finding was associated with the Initiating Events cornerstone attribute of Protection against External Factors (Fire) and affected the cornerstone’s objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown, as well as power operations. The finding was of very low safety significance due to the low fire initiating frequency and the availability of remaining mitigating systems. This finding had a cross-cutting aspect in the area of problem identification and resolution, operating experience because the licensee did not properly evaluate relevant operating experience identified during the preparation of a focused area self assessment. [P.2(a)] (Section 1R17.2.b(1))

- Severity Level IV. The inspectors identified a Severity Level IV, Non-Cited Violation of 10 CFR 50.9(a), “Completeness and Accuracy of Information,” for the licensee’s failure to provide complete and accurate information to the NRC during a 2011 Triennial Fire Protection Inspection. Specifically, between July 7 and October 17, 2011, the licensee failed to inform the NRC that bottles containing 100 percent hydrogen were located in the plant in response to inspectors’ questions regarding flammable gas bottles. The licensee entered this issue into their corrective action program to document the incomplete response provided.

The inspectors determined that the performance deficiency was more than minor because it impacted the regulatory process. Specifically, had the NRC known during the 2011 Triennial Fire Protection Inspection that the hydrogen bottles contained 100 percent hydrogen the inspectors would likely have documented a finding associated with the hydrogen bottles. The issue was a Severity Level IV Non-Cited Violation because the inspectors documented a finding of very low safety significance associated with the flammable hydrogen bottles once they determined that bottles containing 100 percent hydrogen were located in the plant. (Section 1R17.2.b(3))

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50, Appendix B, Criterion X, "Inspection," for the licensee's failure to perform adequate post-installation and post-maintenance inspections on standby liquid control (SBLC) heat tracing and pumps. Specifically, the licensee failed to verify that heat tracing on the SBLC system components was properly installed and later failed to verify that thermal insulation was properly replaced following maintenance on the SBLC pumps, which led to thermal degradation of the explosive material in the squib valves. The licensee entered this issue into their corrective action program and replaced the 3B squib valve.

  The inspectors determined that the finding was more than minor because the finding was associated with the Mitigating Systems cornerstone attribute of equipment performance and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). The finding was of very low safety significance based on a Phase III Significance Determination Process Analysis. This finding had a cross-cutting aspect in the area of problem identification and resolution, operating experience because the licensee did not properly implement vendor operating experience. [P.2(b)] (Section 1R17.2.b(2))

B. Licensee-Identified Violations

No violations of significance were identified.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems, and Barrier Integrity

1R17 Evaluation of Changes, Tests, or Experiments and Permanent Plant Modifications (71111.17)

.1 Evaluation of Changes, Tests, or Experiments

a. Inspection Scope

From February 21, 2012 through March 27, 2012, the inspectors reviewed six safety evaluations performed pursuant to 10 CFR 50.59 to determine if the evaluations were adequate and that prior NRC approval was obtained as appropriate. The inspectors also reviewed 19 screenings where licensee personnel had determined that a 10 CFR 50.59 evaluation was not necessary. The inspectors reviewed these documents to determine if:

- the changes, tests, or experiments performed were evaluated in accordance with 10 CFR 50.59 and that sufficient documentation existed to confirm that a license amendment was not required;
- the safety issue requiring the change, tests or experiment was resolved;
- the licensee conclusions for evaluations of changes, tests, or experiments were correct and consistent with 10 CFR 50.59; and
- the design and licensing basis documentation was updated to reflect the change.

The inspectors used, in part, Nuclear Energy Institute (NEI) 96-07, "Guidelines for 10 CFR 50.59 Implementation," Revision 1, to determine acceptability of the completed evaluations, and screenings. The NEI document was endorsed by the NRC in Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments," dated November 2000. The inspectors also consulted Part 9900 of the NRC Inspection Manual, "10 CFR Guidance for 10 CFR 50.59, Changes, Tests, and Experiments."

This inspection constituted six samples of evaluations and 19 samples of changes as defined in IP 71111.17-04.

b. Findings

No findings of significance were identified.

.2 Permanent Plant Modifications

a. Inspection Scope

From February 21, 2012 through March 27, 2012, the inspectors reviewed 14 permanent plant modifications that had been installed in the plant during the last three years. This review included in-plant walkdowns for portions of the modified Units 2 and 3 standby liquid heat tracing and the high pressure coolant injection (HPCI) turbine high exhaust pressure switches 2-2368 A and B, portions of the modified flood seal penetration outside of the Unit 2 East and West low pressure core injection (LPCI) corner rooms, and Unit 2 MCC 28-1 and Unit 3 MCCs 38-1 thermal overloads and breakers. The modifications were selected based upon risk-significance, safety significance, and complexity. The inspectors reviewed the modifications selected to determine if:

- the supporting design and licensing basis documentation was updated;
- the changes were in accordance with the specified design requirements;
- the procedures and training plans affected by the modification have been adequately updated;
- the test documentation as required by the applicable test programs has been updated; and
- post-modification testing adequately verified system operability and/or functionality.

The inspectors also used applicable industry standards to evaluate acceptability of the modifications. The list of modifications and other documents reviewed by the inspectors is included as an Attachment to this report.

This inspection constituted 14 permanent plant modification samples as defined in IP 71111.17-04.

b. Findings

(1) Flammable Hydrogen Gas Bottles Installed in the Reactor Building

Introduction: The inspectors identified a finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” for the licensee’s failure to check the adequacy of design for flammable hydrogen gas bottles installed in the reactor building and their impact on safety-related SSCs. Specifically, the licensee failed to evaluate how a failure of the flammable hydrogen gas bottles and the resulting fire or explosion at the installed locations could impact nearby safety-related SSCs.

Description: In response to NUREG-0737, “Clarification of TMI [Three Mile Island] Action Plan Requirements,” Section II.F.1 Attachment 6, “Containment Hydrogen,” dated October 31, 1980, the licensee initiated a modification to their post accident containment air monitoring system. Modification M12-2(3)-81-24, “Containment Air Monitoring (CAM),” dated January 14, 1983, implemented changes to the plant’s existing hydrogen monitoring system. The licensee modified the existing hydrogen monitoring system to remove a sample of the containment atmosphere, identify the hydrogen content, and return the sample to the containment. The modification was performed in order to allow for the calibration of the system. Two compressed 100 percent hydrogen and two compressed 100 percent oxygen bottles were installed at each of two CAM panels in both Units 2 and 3, respectively for a total of four hydrogen and four oxygen bottles per unit. The compressed hydrogen and oxygen gas bottles installed were used as reagents for the CAM panels.

The CAM panels and the compressed hydrogen and oxygen gas bottles were located at the 517 foot elevation of the reactor building in Unit 2 and in Unit 3. One installation in Unit 2 was located within approximately 15 feet of safety-related Motor Control Center (MCC) 28-7, which controlled LPCI Loop 1 coolant injection inboard (2-1501-22A) and outboard (2-1501-21A) isolation valves. Another installation in Unit 3 was located less than 10 feet away from the Hydraulic Control Units (HCUs).

The safety evaluation (dated February 8, 1982), associated with the modification package did not consider the potential consequences of the failure of the compressed gas bottles at the installed locations and the resulting fire or explosion on nearby safety-related structures, systems, or components. The safety evaluation determined that the probability of an occurrence or the consequence of an accident, or malfunction of equipment important to safety as previously evaluated in the Final Safety Analysis Report (FSAR) was not increased nor was the possibility for an accident or malfunction of a different type than any previously evaluated in the FSAR created. Furthermore, the licensee determined that the margin of safety, as defined in the basis for any Technical Specification, was not reduced. The licensee did not specifically mention the installation of the hydrogen or oxygen bottles in the documentation for modification M12-2(3)-81-24 and did not address the impact of that modification on safety-related SSCs. Therefore, the inspectors determined that the licensee failed to check the adequacy of the design for the flammable hydrogen gas bottles installed and their impact on safety-related SSCs.

The flammable hydrogen gas bottles present a fire and an explosion hazard. According to Table 2-7.1 of the Society of Fire Protection Engineers (SFPE) Handbook of Fire Protection Engineering (Fourth Edition) hydrogen has a lower flammability limit of 4 percent and an upper flammability limit of 75 percent. This means that a hydrogen mixture of between 4 and 75 percent will burn. The hydrogen bottles contained 100 percent hydrogen. If the hydrogen were to escape from the bottle it would dilute with the surrounding atmosphere and fall into the flammable range of between 4 and 75 percent. Further, the oxygen located in the compressed oxygen bottles could enrich the fire and increase its severity.

The hydrogen gas bottles had a regulator attached to the discharge. However, if a piece of equipment or some object were to hit the regulator it could fail, cause a spark, and ignite the flammable gas. A fire from one or more of the flammable gas bottles could damage safety-related equipment and an explosion could additionally damage other nearby safety-related equipment.

In preparation for this inspection, the licensee prepared a focused area self-assessment (FASA). In that FASA the licensee listed violations identified during 2011 Evaluations of Changes, Tests, or Experiments and Permanent Plan Modifications inspections. One of those violations was a similar flammable gas bottle violation identified at another plant. However, the licensee did not initiate a followup corrective action item to evaluate and address this issue because the licensee erroneously concluded that it was not applicable to Dresden.

Subsequently, the licensee entered this issue into their corrective action program (CAP) as issue report (IR) 01337613, "Mod/50.59 – NRC Concern on Hydrogen and Oxygen Bottles," dated March 7, 2012, to review the placement of the flammable hydrogen gas bottles.

Analysis: The inspectors determined that the failure to evaluate the impact of the flammable hydrogen gas bottles' installed locations near safety-related structures, systems, and components was contrary to 10 CFR Part 50, Appendix B, Criterion III, "Design Control," and was a performance deficiency. The inspectors determined that the finding was more than minor because the finding was associated with the Initiating Events cornerstone attribute of Protection against External Factors (Fire) and affected the cornerstone's objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown, as well as power operations. Specifically, the installed locations of the flammable hydrogen gas bottles could have resulted in damage to safety-related SSCs if the hydrogen gas bottles were to ignite or explode.

In accordance with Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," Attachment 0609.04, "Phase I - Initial Screening and Characterization of Findings," Table 3b the inspectors determined the finding degraded the fire protection defense-in-depth strategies. Therefore, screening under IMC 0609, Appendix F, "Fire Protection Significance Determination Process," was required. The inspectors determined that the finding impacted the Fire Prevention and Administrative Controls category.

Based on review of IMC 0609, Appendix F, Attachment 2, "Degradation Rating Guidance Specific to Various Fire Protection Program Elements," the inspectors determined the degradation rating to be high because of the flammable gases being more flammable than low flashpoint combustibles and there being a significant fire hazard associated with release of the gas. The Duration Factor was 1.0 based on the duration of the degradation being greater than 30 days per Table 1.4.1, "Duration Factors." An overall fire frequency of 2.6E-3 per year was assigned for the four hydrogen gas bottles per unit based on information from IMC 0609, Appendix F, Attachment 4, "Fire Ignition Source Mapping Information: Fire Frequency, Counting Instructions, Applicable Fire Severity Characteristics, and Applicable Manual Fire Suppression Curves."

The Region III Senior Reactor Analyst used the Dresden Standardized Plant Analysis Risk (SPAR) Model, Version 8.15, and Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE), Version 8.0.7.18, to calculate a conditional core damage probability (CCDP) less than 1E-6 assuming a fire due to failure of the flammable gas bottles that resulted in a plant trip and damage to both trains of the LPCI system. Based on the above CCDP and frequency values, the risk associated with this finding is very low (Green).

This finding had a cross-cutting aspect in the area of problem identification and resolution, operating experience because the licensee did not properly evaluate relevant operating experience. Specifically, the licensee included a review of a similar issue identified at another plant last year in the FASA performed for this inspection and determined that it was not applicable to Dresden. [P.2(a)]

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program.

Contrary to the above, from February 8, 1982, through March 9, 2012, the licensee failed to check the adequacy of design for flammable hydrogen gas bottles installed within the reactor building and their impact on safety-related SSCs. Specifically, the licensee failed to evaluate how a failure of the flammable hydrogen gas bottles and a resulting fire or explosion at the installed and/or stored locations could impact nearby safety-related SSCs.

Because this violation was of very low safety significance and it was entered into the licensee's corrective action program as IR 01337613, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 05000237/2012008-01(DRS); 05000249/2012008-01(DRS), Flammable Hydrogen Gas Bottles Installed in the Reactor Building).

(2) Failure to Conduct Adequate Post Installation and Maintenance Inspections on Standby Liquid Control System Components

Introduction: The inspectors identified a finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50, Appendix B, Criterion X, "Inspection," for the licensee's failure to perform adequate post-installation and post-maintenance inspections on standby liquid control (SBLC) heat tracing and pumps. Specifically, the licensee failed to verify that heat tracing on the SBLC system was properly installed and later failed to verify that thermal insulation was properly replaced following maintenance on the SBLC pumps which led to degradation of the explosive material in the squib valves.

Description: The SBLC system is a safety-related system designed to shutdown the reactor by injecting sodium pentaborate. The sodium pentaborate solution in the SBLC system must be maintained above the design basis minimum temperature of 65 degrees Fahrenheit (°F) in order to maintain a liquid solution. The licensee's Technical Specifications (TS) surveillance requirement (SR) 3.1.7.3 requires that the SBLC pump suction piping temperature be greater than or equal to 83°F. The SBLC process piping was wrapped with heat tracing and insulation in order to maintain temperatures above 83°F.

The inspectors determined that excessive heat tracing on SBLC process system components caused thermal degradation on the squib valves in Units 2 and 3.

Unit 2

In August 2009 the licensee upgraded the Unit 2 SBLC heat tracing under modification EC 373699, "Upgrade U2 Standby Liquid Heat Tracing," Revision 0. The licensee replaced the existing heat tracing and insulation on the SBLC suction, discharge, and relief lines, and the injection pumps. The heat tracing was designed to maintain the sodium pentaborate at a temperature of 95.5°F (to provide margin above the TS minimum temperature of 83°F) when the Reactor Building is at a temperature of 65°F. The heat tracing installation was intended to stop at the spool piece before the squib valve. However, during the modification, the installers wrapped the Unit 2 squib valve 2A with heat tracing. The inspectors determined that the licensee did not perform an appropriate inspection upon completion of the modification that should have identified the inappropriate heat tracing installation. As a result, the trigger and primer of the squib valve were subjected to elevated temperatures.

During an injection test performed on October 28, 2011, the 2A SBLC squib valve in Unit 2 failed to function properly which resulted in no flow of demineralized water from the test tank to the reactor. The licensee entered the issue into their corrective action program as IR 01282544, "No Flow to Reactor During DOS 1100-03, SBLC Injection Test," dated October 28, 2011, and subsequently performed equipment apparent cause evaluation (EACE) 1282544-05 to determine the cause of the failure. The licensee determined that the apparent cause of the failure of the squib valve to fire properly was thermal degradation of the primer's explosive material. This conclusion was based on a root cause investigation performed by the squib valve vendor. The licensee documented that the primer experienced temperatures above 120°F, the upper storage and installed temperature limit of the primer. The licensee determined that excessive heating resulting from the heat tracing around the 2A squib valve contributed to the thermal degradation.

Unit 3

The Unit 3 squib valve 3B was installed in November 2008 and was also subjected to elevated temperatures above those specified by the vendor (greater than 120°F). During January 19 and 20, 2010, the licensee performed maintenance work (work order [WO] 0128253 01) on the Unit 3 3B SBLC pump in order to repair a leak. Step F.3 of WO 01282853 01 instructed the workers to "carefully remove blanket insulation, as not to damage the heat trace underneath, to access the cylinder head." Step F.6 instructed the workers, "after packing break-in, carefully reinstall the blanket insulation." These steps were signed off as having been completed in WO 01282853 01. However, the insulation material was not properly replaced and a gap remained in the insulation material of the Unit 3 SBLC injection pumps.

The licensee was also committed to the American Society of Mechanical Engineers (ASME) Operation and Maintenance (OM) Code 1998, 2000 Addenda. Section ISTC-5260(c) of that code stated that, "if a charge fails to fire, all charges with the same batch number shall be removed, discarded, and replaced with charges from a different batch." The inspectors verified that the 3B squib valve was from the same batch as the 2A squib valve that failed to fire properly in October 2011.

The inspectors reviewed EACE 1282544-05 and determined that squib valve 3B in Unit 3 had also been subjected to thermal degradation as a result of the excessive heat tracing. The inspectors discussed their concerns with the licensee who then issued IR 01337933, "NRC Concern: 3B SBLC Squib Valve Temperature," dated March 7, 2012, to document the concern and declared the 3B squib valve inoperable. The licensee documented the unplanned entry into TS 3.1.7 in IR 01337994, "Unplanned Entry into Tech Spec 3.1.7," dated March 7, 2012. Technical Specification 3.1.7 required the licensee to restore an inoperable standby liquid control system to operable status within seven days. The licensee replaced the 3B squib valve on March 9, 2012, and sent it to the vendor for testing.

The licensee documented the gap in the insulation material by the Unit 3 SBLC injection pumps in IR 01339952, “Generate WO Troubleshoot Cause of High Temperatures U3 SBLC,” dated March 12, 2012. The amount of heat tracing that is applied is determined by monitoring the temperatures at the suction and discharge lines of each SBLC injection pump. The licensee measured a temperature difference of 19°F between the suction and discharge of the 3B thermocouples. The normal expected temperature difference is less than 2°F. As a result, the controller had caused the heat tracing to remain energized for longer than necessary, thereby causing thermal degradation of the primer’s explosive material. The inspectors determined that the licensee failed to perform an adequate post-maintenance inspection during WO 0128253 01 that should have discovered the missing insulation material. Had an adequate inspection been performed and the missing insulation been replaced, the heat tracing would likely have not stayed energized as long, which would have resulted in lower temperatures of the SBLC components.

Analysis: The inspectors determined that the failure to perform adequate post-installation and post-maintenance inspections on SBLC heat tracing and pumps was contrary to 10 CFR Part 50, Appendix B, Criterion X, “Inspection,” and was a performance deficiency. Specifically, the licensee failed to verify that heat tracing on the SBLC system was properly installed and later failed to verify that thermal insulation was properly replaced following maintenance on the SBLC pumps, which led to degradation of the explosive material in the squib valves.

The inspectors determined that the finding was more than minor because the finding was associated with the Mitigating Systems cornerstone attribute of equipment performance and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, the thermal degradation of the 2A squib valve resulted in one train of SBLC on Unit 2 to be inoperable and the thermal degradation of the 3B squib valve could have resulted in one train of SBLC being inoperable on Unit 3.

In accordance with Inspection Manual Chapter (IMC) 0609, “Significance Determination Process,” Attachment 0609.04, “Phase I - Initial Screening and Characterization of Findings,” Table 4a the inspectors determined the finding represented an actual loss of a single Train of the Unit 2 SBLC for greater than its TS allowed outage time. The Region III Senior Reactor Analysts (SRAs) determined that a Phase III Significance Determination Process (SDP) evaluation was necessary.

The SRAs performed a Phase III internal events SDP evaluation of the finding using SAPHIRE Version 8.0.7.18 and the Dresden Standardized Plant Analysis Risk (SPAR) model (Version 8.15). The Dresden model was modified to not include manual shutdown sequences as potential anticipated transients without scram (ATWS) events, because operators perform a manual shutdown in a controlled manner, and transient initiating events already cover potential ATWS events in the SPAR model. This change was made after discussions with Idaho National Laboratory.

Using the SPAR model, the result was an estimated change in core damage frequency (ΔCDF) of 1.2E-7/yr for internal events. The dominant core damage sequence involved a transient initiating event with a failure of the reactor protection system (i.e., an ATWS event) and a failure of the standby liquid control system. Since the total estimated change in core damage frequency was greater than 1.0E-7/yr, IMC 0609, Appendix A, Attachment 3, "User Guidance for Screening of External Events Risk Contribution," was used to screen external event contributions.

The seismic risk contribution was screened since the SBLC system is not on the seismic safe shutdown equipment list provided in the licensee's Individual Plant Examination for External Events (IPEEE) evaluation. Flooding scenarios were screened using IMC 0609, Appendix A, Table 3.1, "Plant Specific Flood Scenarios." The guidance lists SSCs important to internal flooding and no risk-significant flooding scenarios were identified for Dresden. Fire risk contribution was screened out because the SBLC system is not included in the licensee's Appendix R fire safe shutdown analysis.

The potential risk contribution for this finding from large early release frequency (LERF) was evaluated using the guidance of IMC 0609 Appendix H, "Containment Integrity Significance Determination Process." Dresden is a boiling water reactor (BWR) with a Mark I containment. The dominant core damage sequences for this finding were anticipated transient without scram (ATWS) sequences. For these sequences there is a LERF factor of 0.3 for BWRs with Mark I containments. Multiplying the ΔCDF for the dominant core damage sequences (1.2E-7/yr) by the LERF factor of 0.3 yields a ΔLERF of 3.6E-8/yr (Green).

Based on the Phase III analysis, the inspectors determined that the finding was of very low safety-significance (Green).

This finding had a cross-cutting aspect in the area of problem identification and resolution, operating experience because the licensee did not properly implement operating experience. Specifically, the licensee performed an apparent cause evaluation that determined the failure mechanism of the 2B squib valve and failed to identify that the 3B squib valve was also subjected to thermal degradation conditions based on information supplied by the vendor. [P.2(b)]

Enforcement: Title 10 CFR Part 50, Appendix B, Criteria X, "Inspection," requires, in part, that a program for inspection of activities affecting quality shall be established and executed by or for the organization performing the activity to verify conformance with the documented instructions, procedures, and drawings for accomplishing the activity.

Contrary to the above, from August 2009 through March 7, 2012, the licensee failed to execute an appropriate inspection for work performed on the Units 2 and 3 SBLC pumps and associated equipment. Specifically, the licensee failed to determine via inspection that the heat tracing was properly installed on the Unit 2 2A SBLC components and that insulation material removed around the Unit 3 SBLC pumps was properly reinstalled post maintenance.

Because this violation was of very low safety significance and it was entered into the licensee's corrective action program as IRs 01337933 and 01337994, and the licensee replaced the 3B squib valve, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 05000237/2012008-02(DRS);

05000249/2012008-02(DRS), Failure to Conduct Adequate Post Installation and Maintenance Inspections on Standby Liquid Control System Components).

(3) Failure to Provide Complete and Accurate Information to the NRC

Introduction: The inspectors identified a Severity Level IV, Non-Cited Violation of 10 CFR 50.9(a), "Completeness and Accuracy of Information," for the licensee's failure to provide complete and accurate information to the NRC during a 2011 Triennial Fire Protection Inspection. Specifically, between July 7 and October 17, 2011, the licensee failed to inform the NRC that bottles containing 100 percent hydrogen were located in the plant in response to inspectors' questions regarding flammable gas bottles.

Description: During the 2011 Triennial Fire Protection Inspection at Dresden, the licensee documented in writing in their inspection question database, the inspectors' questions regarding the presence of hydrogen gas bottles in the plant as, "are there any compressed gas cylinders for either flammable gases, such as hydrogen or oxygen in the plant"? The licensee documented their response as, "yes, there are hydrogen and oxygen bottles used for calibrating the primary containment hydrogen/oxygen monitors," and that the "hydrogen bottles contain approximately 9.5 percent hydrogen." The licensee captured this question and response as request No. 18 in their inspection question database on July 7, 2011.

On August 30, 2011, the inspectors asked a follow-up question, which was documented in writing in request 18-1, for the licensee to, "identify which hydrogen/oxygen bottles have flammable concentrations of gas." The licensee documented their response that, "none of the bottles contain flammable concentration of gases." The licensee provided supporting information in writing that a hydrogen bottle containing 9.5 percent hydrogen and 90.5 percent nitrogen (an inert gas) was not flammable. The licensee's written responses were reviewed by several members of the licensee's staff.

During the current Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications Baseline Inspection the inspectors determined that bottles containing 100 percent hydrogen were located in the plant and documented an NCV of very low safety significance (Green) in Section 1R17.2b of this report. Had the inspectors been aware that 100 percent hydrogen bottles were located in the plant during the 2011 Triennial Fire Protection Inspection, the inspectors would likely have taken the same enforcement action.

The licensee documented the incomplete response in IR 01337800, "Mod/50.59: Incomplete Response to NRC Question during FP Inspection," dated March 7, 2012.

Analysis: The inspectors determined that the failure to provide complete and accurate information was contrary to 10 CFR 50.9 and was a performance deficiency because it resulted in the NRC not undertaking further inquiry when the original information was presented. The inspectors determined that the performance deficiency was more than minor because it impacted the regulatory process. Specifically, had the NRC known during the 2011 Triennial Fire Protection Inspection that the hydrogen bottles contained 100 percent hydrogen the inspectors would likely have documented a finding associated with the hydrogen bottles. An NCV of very low safety significance (Green) associated with the hydrogen bottles are documented in Section 1R17.2b of this report.

Violations of 10 CFR 50.9 are dispositioned using the traditional enforcement process instead of the significance determination process (SDP) because they are considered to be violations that potentially impede or impact the regulatory process.

Using the information provided in the Enforcement Policy, Section 6.9, the inspectors determined that this issue was a Severity Level IV NCV, as it did not meet the definition for a Severity Level I, II, or III violation.

Enforcement: Title 10 CFR Part 50.9(a), "Completeness and Accuracy of Information," requires, in part, that information provided to the NRC by a licensee to be complete and accurate in all material aspects.

Contrary to the above, from July 7, 2011, through October 17, 2011, the licensee failed to provide complete and accurate information to the NRC during the 2011 Triennial Fire Protection Inspection. Specifically, the licensee failed to inform the NRC in either its verbal or written responses to the inspectors' questions regarding flammable gas bottles that bottles containing 100 percent hydrogen were located in the plant. This information was material to the NRC's decision not to issue an NCV during the 2011 Triennial Fire Protection Inspection.

Because this violation was of very low safety significance and it was entered into the licensee's corrective action program as IR 01337800, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 05000237/2012008-03(DRS); 05000249/2012008-03(DRS), Failure to Provide Complete and Accurate Information to the NRC).

(4) Unit 2 East and West LPCI Corner Rooms Internal Flooding Event Issue

Introduction: The inspectors identified an issue related to the design basis internal flood barrier protection for the Unit 2 East and West LPCI Rooms. Specifically, flood seal No. 9 penetration for the Unit 2 East LPCI corner room and flood seal No. 5 and 10 penetrations for the Unit 2 West LPCI Corner Room were classified as non-safety-related and may potentially be susceptible to an internal flooding condition due to failure of non-seismic piping which was not designed and licensed to withstand a Class 1 earthquake event.

Description: The inspectors reviewed Engineering Change (EC) 386469, "Repair of Flood Seals for Unit 2 West LPCI Corner Room Penetration No. 5 and No. 10 on Flood Seal Drawing FL-37 and East LPCI Corner Room Penetration No. 9 on Flood Seal Drawing FL-41", Revision 3. Section 4.1.4.1 of this EC stated the function of the flood seal penetrations No. 5, 9 and 10 was to "prevent water from leaking from the Torus basement into the Reactor Building corner rooms where the LPCI and Core Spray pumps are located." The LPCI and Core Spray pumps are safety-related. The purpose of this EC was to repair the aforementioned existing flood seal penetrations. The repair to the existing flood seal penetrations was classified per the EC as non-safety-related. In Revision 1 of this EC, the classification of the modification was changed from safety-related to non-safety-related. The licensee made this classification change to the EC because they concluded that the flood seal penetrations do not perform a safety-related or accident mitigation function as described by their current license basis. Also, the licensee described the flood seal penetration as being conservatively classified as safety-related because the licensee had not assigned a classification to the flood seal penetrations. During a walkdown of the flood seal penetrations, the inspectors identified non-safety-related fire protection and service water piping in close proximity to the non-safety-related flood seal penetrations. The inspectors noted that the non-safety-related piping was not designed and licensed to withstand a Class 1 earthquake event and failure of the piping could result in an internal flood in the torus basement which could generate a flood height that could reach and bypass flood seal penetrations No. 5, 9 and 10. During this inspection, the licensee was unable to locate an evaluation of whether or not the non-safety-related piping could withstand a Class 1 earthquake event or an evaluation to determine the flood height generated by a failure of the non-safety-related piping that would flood the torus basement when subjected to a Class 1 earthquake event and determine whether the flood could reach and bypass the flood seal penetration.

In response to this concern, the licensee initiated Condition Report (CR) 01338733, "Mod/50.59: Add'l Info Needed for Non-Seismic Piping," dated March 8, 2012. The inspectors also discussed this issue with staff in the Office of Nuclear Reactor Regulation (NRR).

After the exit, the licensee provided the inspectors additional information on IPEEE relevant to the design basis and licensing basis of the flood seals for the Unit 2 East and West corner rooms, which will require additional NRC review. Therefore, this issue is considered unresolved pending additional inspector review of the information provided by the licensee and consultation with NRR to determine the design and licensing basis requirements of the flood seals at Dresden. (URI 05000237/2012008-04 (DRS), Unit 2 East and West LPCI Corner Rooms Internal Flooding Event Issue).

4. OTHER ACTIVITIES (OA)

4OA2 Identification and Resolution of Problems

.1 Routine Review of Condition Reports

a. Inspection Scope

From February 21, 2012 through March 27, 2012, the inspectors reviewed Corrective Action Process documents that identified or were related to 10 CFR 50.59 evaluations and permanent plant modifications. The inspectors reviewed these documents to evaluate the effectiveness of corrective actions related to permanent pant modifications and evaluations for changes, tests, or experiments issues. In addition, corrective action documents written on issues identified during the inspection were reviewed to verify adequate problem identification and incorporation of the problems into the corrective action system. The specific corrective action documents that were sampled and reviewed by the inspectors are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

4OA6 Meetings

.1 Exit Meeting Summary

The inspectors presented the inspection results to Mr. Dave Czufin, and to Mr. Paul Wojtkiewicz and to other members of the licensee staff on March 9, 2012, and on March 27, 2012, respectively. The licensee personnel acknowledged the inspection results presented. The inspectors confirmed that proprietary material was reviewed during the inspection and was either returned to the licensee staff or will be handled in accordance with NRC policy on proprietary information.

ATTACHMENT: SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
D. Czufin, Site Vice President
S. Marik, Plant Manager
G. Graff, Nuclear Oversight Manager
P. Wojtkiewicz, Manager Design Engineering
L. Jordan, Training Director
J. Knight, Engineering Director
T. Mohr, Engineering Program Manager
H. Bush, Radiation Protection Manager
P. Quealy, EP Manager
G. Storrick, Design Engineering
J. Patel, Design Engineering
D. Eaman, Design Engineering
D. Leggett, Regulatory Assurance Manager
R. Ruffin, Regulatory Assurance
T. Griffith, Corporate Regulatory Assurance

Nuclear Regulatory Commission
H. Peterson, Chief, Engineering Branch 3, DRS
T. Briley, Interim Resident Inspector, Dresden

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened and Closed

05000237/249/2012008-01	NCV	Flammable Hydrogen Gas Bottles Installed in the Reactor Building. (Section 1R17.2b)
05000237/249/2012008-02	NCV	Failure to Conduct Adequate Post Installation and Maintenance Inspections on Standby Liquid Control System Components. (Section 1R17.2b)
05000237/249/2012008-03	NCV	Failure to Provide Complete and Accurate Information to the NRC. (Section 1R17.2b)

Opened

05000237/2012008-04	URI	Unit 2 East and West LPCI Corner Rooms Internal Flooding Event Issue. (Section 1R17.2b)

Discussed

None

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.

APPARENT CAUSE EVALUATION REPORTS

Number	Description or Title	Date or Revision
EACE 1282544-05	SBLC Squib Valve Failure	03/07/12

CALCULATIONS

Number	Description or Title	Date or Revision
GENE-0000-0094-3260-R1	Piping Stress Analysis for Internal Core Spray Line with Lower Sectional Replacement – Dresden Unit 2	1
8188-123-D2	Flued Head Anchor X-123	Minor Revision 0A
8188-124-D2	Flued Head Anchor X-124	Minor Revision 2B
3C2-0181-001	Temperature Transient in the Main Steam Tunnel	Minor Revision 0B
DRE01-0041	Updated EQ Zone Parameter Tables Following Implementation of Extended Power Uprate	Minor Revision 2C
DRE05-0079	Evaluations of Bridges 8 and 9 for Steam Dryer Hauling Loads, Underground Utilities Along Path of Transporter	4
DRE08-0022	Load Drop Evaluation on the Turbine Floor EL 561'-6”	4
DRE11-0012	Evaluation of Turbine Building Floor Framing for the Loads Associated with the Turbine Retrofit Project	1
DRE11-0013	Unit 2 Trackway Sea Van Load Drop Evaluation	0
3C2-0181-001	Temperature Transient in the Main Steam Tunnel	000B
DRE09-0003	High Pressure Coolant Injection (HPCI) Turbine Exhaust High Pressure Switch Setpoint Error Analysis	0
10553-CALC-03	Dresden EDG Fuel Oil Storage Tanks Volume	0B
DRE02-0040	MOV Terminal Voltage AC Motor Terminal Voltage Calculation for Dresden System 1301, Unit 3	000A
DRE05-0081	Dresden U3 MCC Breaker Setting for Continuous Duty Motors and MOVs-Attach. A	001B

CORRECTIVE ACTION PROGRAM DOCUMENTS GENERATED DURING INSPECTION (AR-xx)

Number	Description or Title	Date or Revision
01338708	Typo On Drawing 12E-2460 Sh 3	03/09/12
01338917	SBLC Heat Tracing Line not Properly Installed After Maintenance	03/09/12
01338733	Mod/50.59: Add'l Info Needed For Non-Seismic Piping	03/08/12
01331002	Scaffold Installed Longer Than 90 Days Without 50.59	02/23/12
01331049	NRC Identified Issues With EC 380192	02/23/12
01331111	50.59 Applicability Form For Screening 2011-0225	02/23/12
01331443	Incorrect User Reference In DOP 5400-18 Procedure	02/22/12
01331539	NRC Observation on Responses to 50.59 Screening Question 4	02/24/12
01335421	Scaffold Removal Not Documented When Removed	03/02/2
01336242	ATI Closed Without Procedure Being Revised for EC 380192	03/05/12
01337200	Discrepancy In Voltage Values In Revised Calc.	03/05/12
01337277	Unit 1 DOP 6700-06 Procedure Enhancement	03/06/12
01337565	MOD/50.59 Inspection: UFSAR and TRM Surv Freq for Corner Room Door	02/29/12
01337613	MOD/50.59 - NRC Concern on Hydrogen and Oxygen Bottles	02/23/12
01337800	Mod/50.59: Incomplete Response to NRC Question During FP Inspection	03/07/12
01337933	NRC Concern: 3B SBLC Squib Valve Temperature	03/07/12
01337994	Unplanned Entry Into Tech Spec 3.1.7	03/07/12
01339952	Generate WO Troubleshoot Cause of High Temperatures U3 SBLC	03/12/12
01339954	Generate WO Troubleshoot Cause of High Temperatures U2 SBLC	03/12/12

CORRECTIVE ACTION PROGRAM DOCUMENTS REVIEWED

Number	Description or Title	Date or Revision
CR 01262422	Mounting Plate Does Not Match EC 384146 Drawing	09/13/11
CR 01262329	CEA's 4, 6, and 8 Exceeded the Embedded Depth After Torque	09/13/11
IR 00895342	Potential for H2 Addition System Trips on Units 2 and 3	03/20/09

CORRECTIVE ACTION PROGRAM DOCUMENTS REVIEWED

Number	Description or Title	Date or Revision
IR 01017020	MSLB Calculation 3C2-0181-001 Did Not Consider Opening to Reactor Building	08/18/09
IR 01282544	No Flow to Reactor During DOS 1100-03, SBLC Injection Test	10/28/11
IR 01285454	2B SBLC Squib Valve Has No Continuity Indication	11/03/11
IR 01305011	U2 SBLC Squib Valves and Heat Trace	12/20/11
IR 01305997	3B SBLC Squib Valve Replacement	12/22/11
CR 00987500	SDC Temp Element 2-260-13B Failed Open	11/02/12
CR 01320909	CREVS LCO Delayed Two Days Due to Parts Issue	01/31/12
CR 01021685	QV IDS That QV Hold Point Were not Placed in WP for SR Work	01/26/10
CR 01221132	NOS IDd QV Hold Points not Placed in WOs as Required	05/26/11

DRAWINGS

Number	Description or Title	Date or Revision
12E-3440, Sheet 3	Schematic Diagram – LPCI/Containment Cooling System MOVs	X
12E-2320	Key Diagram Reactor Building 480V Motor Control Centers 29-4, 28-7 & 29-7	AS
12E-2460	Standby Liquid Control Heat Tracing Layout Diagram (Unit 2) – Sheet 3	A
12E-2460	Standby Liquid Control Heat Tracing Layout Diagram (Unit 2) – Sheet 2	AD
12E-2460	Standby Liquid Control Heat Tracing Layout Diagram (Unit 2) – Sheet 2	AE
12E-2460	Standby Liquid Control Heat Tracing Layout Diagram (Unit 2) – Sheet 2	AF
12E-3460	Standby Liquid Control Heat Tracing Layout Diagram (Unit 3) – Sheet 3	A
M-33	Diagram of Standby Liquid Control Piping (Unit 2)	HZ
M-364	Diagram of Standby Liquid Control Piping (Unit 3)	AS
12E-2508	Schematic Diagram Primary Containment Isolation SDC System Isolation Logic	x
12E-2708	Wiring Diagram Panel 902-4 Part 4	CJ
12E-2711	Wiring Diagram Panel 902-4 Part 7	BW
12E-2901x	Cable Tabulation Cables 21050 to 21099	T

EQUIVILENCY EVALUATIONS

Number	Description or Title	Date or Revision
EC 386469	Repair of Flood Seals for Unit 2 West LPCI Corner Room Penetration No. 5 and No. 10 on Flood Seal Drawing FL-37 and East LPCI Corner Room Penetration No. 9 on Flood Seal Drawing FL-41	3

10 CFR 50.59 EVALUATIONS

Number	Description or Title	Date or Revision
2010-04-002	Cumulative Effects of Foreign Material on the Dresden Unit 3 Reactor Vessel and Connected Systems – D3R21	0
2011-04-001	D2R22 – Cumulative Effects of Foreign Material on the Dresden Unit 2 Reactor Vessel and Connected Systems	0
2009-07-001	Unit 2 Core Spray Lower Sectional Replacement	08/12/09
2009-09-002	EC 376856 / Gag PCIV 3-1599-61 Open	09/11/09
2010-01-001	Review of ATWS Requirements for Reactor Recirculation Pumps ASD Installation	0
2010-01-002	Rx Recirculation MG Set Replacement With ASD Units	0

10 CFR 50.59 SCREENINGS

Number	Description or Title	Date or Revision
2011-0225	Loss of Power to ESS Service System Bus or Instrument Bus	0
2009-0161	Unit 2(3) Monthly Station Battery Inspection	0
2009-0185	125VDC Electrical System	0
2009-0229	Crosstie Unit 1 480 Volt Power Center Feeds	0
2009-0231	Unit 2(3) SBO Ventilation	0
2010-01-002	RX Recirculation MG Set Replacement with ASD Units	0
2010-01-001	Review of ATWS Requirements for Reactor Recirculation Pumps ASD Installation	0
2009-0331	Minor Revision to Calculations 8188-123-D2 and 8188-124-D2 to Account for a Junction Box and Conduit Support Found Attached to Flued Head Anchors X-123 and X-124	11/24/09

10 CFR 50.59 SCREENINGS

Number	Description or Title	Date or Revision
2009-0232	Smoke, Noxious Fumes or Airborne Contaminants in the Control Room	08/17/09
2009-0279	Off Gas System Sample Conditioning System for H2 and O2 Analyzers and Hydrogen Analyzer Startup and Shutdown	12/18/09
2010-0006	Perform Line Stop on the RBCCW Line 3-3701B-18"-L While Valve 3-3701-B-500 is Repaired	01/15/10
2010-0074	Minor Revision to Calculation 041326(CMED)/EC 379343	04/15/ 10
2010-0098	Install Conduit Seals on Pressure Switches 2(3)-0263-111A, B, C, D	06/18/ 10
2010-0202	EC (DCR) to Revise Design Analysis 3C2-0181-001 and DRE01-0041 and UFSAR Change/EC 380908	09/30/10
2010-0279	Temporary Shielding on Lines 2/3-20432A-2", 2/3-20432B-2" and Temp Hoses for CW Transfer	10/19/10
2010-0303	Revise HPCI Instrument Calculations NED-I-EIC-0110 and NED-I-EIC-0111	11/15/10
2010-0329	Revise Setpoints for EDG Fuel Oil Storage Tank Level Switches	03/08/11
2011-0170	Structural Evaluations Associated with U2 Turb. Retrofit Project	08/17/11
2011-0179	Temporary Shielding Permits for D2R22	09//14/11

MODIFICATIONS

Number	Description or Title	Date or Revision
EC 377728	U2 Shuthdown Cooling Logic Change	000 & 003
PEEN 39628 (EC 362736)	Replacement of GE Relays 2-1530-205A	09/22/08
EC 380192	Thermal Overload Replacement for new Motor EPN 3-1301-1	000
EC 5771	Unit 2 Core Spray Lower Sectional Line Replacement	1
EC 362586	Piping Support Modification for SW Discharge Line – U3	0
EC 373804	Install Lateral Restraints on Unit 3 Reactor Building Exhaust Fan Discharge Duct	0
EC 376194	Install Carbon Fiber Wrap on Piping 2/3-3327-12", 2/3-3329-16", and 2/3-3346-24" near the 1A Condensate Storage Tank	4
EC 376423	Replace Shutdown Cooling (SDC) HX Relief	0

MODIFICATIONS

Number	Description or Title	Date or Revision
	Valve 3-1001-210B to be in Compliance with ASME Code - U3	
EC 383210	Perform Weld Overlay on Welds PS2/201-1, PS2-TEE/202-4B and PD1A-D14 on the Unit 2 Recirc Line	4
EC 374234	Replace HPCI Turbine High Exhaust Pressure Switches 2-2368 A and B	0
EC 372699	Upgrade U2 Standby Liquid Heat Tracing	0
EC 365618	Upgrade U3 Standby Liquid Heat Tracing	1
M12-2(3)-81-24	Containment Air Monitoring	01/14/83

OTHER DOCUMENTS

Number	Description or Title	Date or Revision
Design Specification No. 26A7705	Core Spray Line Lower Sectional Replacement	10/15/08
Procedure CC-AA-304	Component Classification	5
NUREG 0823	Integrated Plant Safety Assessment Report Dresden Unit 2, Section 4.7	02/83
01299519-03	Dresden Functional Area Self Assessment -- Configuration Management – NRC Triennial Modifications and 10CFR50.59 Inspection	02/29/12

PROCEDURES

Number	Description or Title	Date or Revision
Design Specification No. 26A7705	Core Spray Line Lower Sectional Replacement	10/15/2008
Procedure CC-AA-304	Component Classification	5
NUREG 0823	Integrated Plant Safety Assessment Report Dresden Unit 2, Section 4.7	02/83
DOP 3390-01	Unit 2 Hydrogen Addition System Operation	33
DOP 3390-01	Unit 2 Hydrogen Addition System Operation	39
DOA 5750-04	Smoke, Noxious Fumes or Airborne Contaminants in the Control Room	26
DOA 5750-09	Unplanned Breach in the Control Room Envelope Boundary	02
DOP 5400-18	Off Gas System Sample Conditioning System for H2 and O2 Analyzers and Hydrogen Analyzer Startup and Shutdown	36
ER-AA-2006	Lost Parts Evaluations	7

PROCEDURES

Number	Description or Title	Date or Revision
DOP 6700-06	Crosstie Unit 1 480 volt Power Center Feeds	9
LS-AA-104-1000	Exelon 50.59 Resource Manual	6

WORK ORDERS

Number	Description or Title	Date or Revision
01282853 01	Boron Liquid Leak on 3B SBLC Pump	01/20/10
01258155 01	3A SBLC Failed Quarterly IST Surveillance	08/10/09
01284082 03	Implement EC 377728bfor U2 Shutdown Cooling	02/17/11

LIST OF ACRONYMS USED

AC	Alternating Current
ADAMS	Agencywide Documents Access and Management System
AR	Action Request
ASME	American Society of Mechanical Engineers (ASME)
ATWS	Anticipated Transient without Scram
BWR	Boiling Water Reactor
CAM	Containment Accident Monitors
CAP	Corrective Action Program
CCDP	Conditional Core Damage Probability
CR	Condition Report
CFR	Code of Federal Regulations
DRP	Division of Reactor Project
DRS	Division of Reactor Safety
EACE	Equipment Apparent Cause Evaluation
EC	Engineering Change
EDG	Emergency Diesel Generator
ESS	Essential Service System
FASA	focused Area Self-Assessment
FP	Fire Protection
FSAR	Final Safety Analysis Report
HCU	Hydraulic Control Unit
HPCI	High Pressure Coolant Injection
IMC	Inspection Manual Chapter
IPEEE	Individual Plant Examination for External Events
IR	Issue Report
LERF	Large Early Release Frequency
LPCI	Low Pressure Coolant Injection
MCC	Motor Control Center
MOV	Motor-Operated Valve
MSLB	Main Steam Line Break
NCV	Non-Cited Violation
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
OM	Operation and Maintenance
PARS	Public Available Records System
SAPHIRE	Systems Analysis Programs for Hands-on Integrated Reliability Evaluations
SBLC	Standby Liquid Control
SDP	Significance Determination Process
SFPE	Society of Fire Protection Engineers
SPAR	Standardized Plant Analysis Risk
SR	Surveillance Requirement
SRA	Senior Rector Analyst
SSC	Structures Systems and Components
TRM	Technical Requirements Manual
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Issue
VDC	Volts Direct Current
WO	Work Order

In accordance with Title 10, Code of Federal Regulations (CFR), Part 50, Section 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC Website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Hironori Peterson, Chief
Engineering Branch 3
Division of Reactor Safety

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

Enclosure: Inspection Report 05000237/2012008; 05000249/2012008
w/Attachment: Supplemental Information

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