July 30, 2007

Mr. Christopher M. Crane
President and Chief Nuclear Officer
Exelon Nuclear
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3
NRC INTEGRATED INSPECTION REPORT 05000237/2007003;
05000249/2007003; 07200037/2007-001

Dear Mr. Crane:

On June 30, 2007, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Dresden Nuclear Power Station, Units 2 and 3. The enclosed integrated inspection report documents the inspection findings, which were discussed on July 17, 2007, with Mr. D. Bost 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, four NRC identified findings of very low safety significance (Green) were identified. All of these issues involved violations of NRC requirements. However, because of the very low safety significance and because they were entered into your corrective action program, the NRC is treating these violations as Non-Cited Violations consistent with Section VI.A.1. of the NRC Enforcement Policy.

If you contest the subject or severity of a Non-Cited Violation, 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, D.C. 20555-0001; with copies to the Regional Administrator, Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Dresden Nuclear Power Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be 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 05000237/2007003; 05000249/2007003
and 07200037/2007-001
w/Attachment: Supplemental Information

cc w/encl:
Site Vice President - Dresden Nuclear Power Station
Dresden Nuclear Power Station Plant Manager
Regulatory Assurance Manager - Dresden
Chief Operating Officer
Senior Vice President - Nuclear Services
Senior Vice President - Mid-West Regional Operating Group
Vice President - Mid-West Operations Support
Vice President - Licensing and Regulatory Affairs
Director Licensing - Mid-West Regional Operating Group
Manager Licensing - Dresden and Quad Cities
Senior Counsel, Nuclear, Mid-West Regional Operating Group
Document Control Desk - Licensing
Assistant Attorney General
Illinois Emergency Management Agency
State Liaison Officer
Chairman, Illinois Commerce Commission

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Dresden Nuclear Power Station Plant Manager
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Chief Operating Officer
Senior Vice President - Nuclear Services
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Letter to Christopher M. Crane from Mark A. Ring dated July 30, 2007

SUBJECT: DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3
NRC INTEGRATED INSPECTION REPORT 05000237/2007003;
05000249/2007003; 07200037/2007-001

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-237; 50-249; 72-037
License Nos:	DPR-19; DPR-25
Report No:	05000237/2007003; 05000249/2007003; 07200037/2007-001
Licensee:	Exelon Generation Company
Facility:	Dresden Nuclear Power Station, Units 2 and 3
Location:	Morris, IL 60450
Dates:	April 1 through June 30, 2007
Inspectors:	C. Phillips, Senior Resident Inspector M. Sheikh, Resident Inspector J. McGhee, Reactor Engineer A. Koonce, Reactor Engineer M. Bielby, Senior Operations Engineer M. Gryglak, Reactor Inspector, Region III S. Bakhsh, Health Physicist, Region III J. Pearson, Transportation and Storage Safety Inspector, Spent Fuel Storage and Transportation (SFST), Office of Nuclear Material Safety and Safeguards (NMSS) A. Dahbur, Reactor Engineer C. Acosta Acevedo, Reactor Engineer R. Schulz, Illinois Emergency Management Agency
Approved by:	M. Ring, Chief Branch 1 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000237/2007003; 05000249/2007003; 07200037/2007-001; 04/01/2007 - 06/30/2007; Exelon Generation Company, Dresden Nuclear Power Station, Units 2 and 3, Evaluations of Changes, Tests, or Experiments, Fire Protection, Corrective Action, and Event Followup.

This report covers a 3-month period of baseline resident inspection, routine inspections by regional inspectors, and a routine inspection by regional and headquarters inspectors of pre-operational and operational activities associated with an Independent Spent Fuel Storage Installation. The inspection was conducted by Region III inspectors, headquarters inspectors, and the resident inspectors. Four Green findings, involving Non-Cited Violations, were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. <u>NRC-Identified and Self-Revealing Findings</u>

Cornerstone: Initiating Events

Green. A performance deficiency involving a Non-Cited Violation of 10 CFR 50.65(a)(4) was self-revealed after the Unit 2 reactor scram on May 4, 2007. The licensee failed to adequately assess and manage the risk associated with emergent work. The licensee's root cause report determined that the cause of the scram was a loss of feedwater due to closure of the condensate pre-filter isolation and bypass valves. The condensate pre-filter programmable logic controller central processing unit (CPU) had failed and work was in-progress to replace the CPU. The performance of this work caused the valve closure. Corrective actions included: 1) the CPU card was replaced with vendor support and the programmable logic controller (PLC) was restarted; 2) a standing order on risk management treatment of condensate pre-filter PLC hardware activities was initiated; 3) actions for design engineering to develop a design specification for the condensate pre-filter valve CPU software were assigned; and 4) actions to proceduralize the requirements associated with performing condensate pre-filter troubleshooting, maintenance activities and system restoration were initiated.

The finding was greater than minor because the maintenance that was performed resulted in an initiating event. The finding was of very low safety significance because all the equipment necessary to mitigate the transient worked as expected. The primary cause of this finding was related to the cross-cutting issue of human performance (work practices) because the human performance prevention techniques provided to the staff were not adequately followed. Personnel proceeded in the face of uncertainty and unexpected circumstances by not following a conservative decision making process (H.4.(a)). (Section 4OA3.2)

Cornerstone: Mitigation Systems

Green. The inspectors identified a finding having very low safety significance and an associated Severity Level IV Non-Cited Violation (NCV) of 10 CFR 50.59 for the licensee's failure to perform an adequate safety evaluation review for changes made to the facility per safety evaluation 2005-02-001. Specifically, the licensee failed to provide an adequate basis as to why changes that credited the isolation condenser for decay heat removal in lieu of the automatic depressurization system and low pressure coolant injection (LPCI)/containment cooling service water and credited the control rod drive system for control of reactor coolant inventory in lieu of LPCI during a postulated high pressure coolant injection (HPCI) room high energy line break (HELB) did not require a license amendment. The licensee entered this issue into its corrective action program.

Because the issue potentially impacted the NRC's ability to perform its regulatory function, this finding was evaluated using the traditional enforcement process. The finding was determined to be more than minor because the inspectors could not reasonably determine that the activity implemented per 10 CFR 50.59 safety evaluation 2005-02-001, which adversely affected systems important to safety, would not have ultimately required NRC approval. The inspectors completed a significance determination of the underlying technical issue using NRC's Inspection Manual Chapter (IMC) 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," and answered "no" to all of the questions under the Mitigating Systems cornerstone. Based upon this Phase 1 screening, specifically this issue did not represent a loss of function, did not result in exceeding a Technical Specification (TS) allowed outage time, and did not affect external event mitigation, the inspectors concluded that the issue was of very low safety significance (Green). The inspectors determined there was not a cross-cutting aspect to this finding. In accordance with the Enforcement Policy, the violation was therefore classified as a Severity Level IV violation. (Section 1R02)

Green. The inspectors identified a Non-Cited Violation (NCV) of the Dresden Nuclear Power Station Renewed Facility Operating License having very low safety significance (Green) for the licensee's failure to restore fire brigade equipment to a ready status in a timely manner after the performance of a fire drill on June 12, 2007. Corrective actions by the licensee included the restoration of the fire brigade equipment to a ready status at 8:22 p.m. on June 13, 2007.

The inspectors determined that this finding was more than minor because the failure to restore the fire brigade equipment to a ready status, if left uncorrected, would become a more significant safety concern. The inspectors completed a Phase 1 significance determination of this issue using IMC 0609, "Significance Determination Process," Appendix A, Attachment 1, dated March 23, 2007. The inspectors determined that the finding affected the Mitigating Systems cornerstone and the fire protection defense-in-depth strategies. However, as discussed by IMC 0609, Appendix A, Attachment 1, issues related to performance of the fire brigade are not included in IMC 0609, Appendix F, "Fire Protection Significance Determination Process," 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 the condition existed for slightly more than 24 hours, the delay in getting additional self-contained breathing apparatus would be a maximum of about 10 minutes, and the majority of the safety significant equipment in the turbine building is protected by an automatic fire suppression system. The primary cause of this finding was related to the cross-cutting issue of human performance (work practices) because the licensee failed to ensure adequate supervisory and management oversight of work activities (restoration of the fire brigade equipment), such that nuclear safety was supported (H.4(c)). (Section 1R05)

Green. On September 7, 2006, the inspectors identified a performance deficiency involving a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” for the failure to promptly identify and adequately correct issues with the operation and testing of the isolation condenser emergency make-up pump until prompted by the inspectors. The pump is designed to ensure an adequate supply of make-up water to the isolation condenser during flood conditions to prevent core damage. Specifically, the licensee failed to ensure adequate corrective actions were taken to test the pump to its design limits, and failed to identify deficiencies with the suction hose to eliminate friction losses in the suction line. The licensee missed an opportunity to re-evaluate the losses in the suction line when the required flowrate was increased to 350 gpm in 2003 as a result of extended power uprate and consequently, failed to ensure that equipment associated with the pump was sized adequately for execution of the pump’s safety function. The licensee’s corrective action for this issue included increasing the margin in net positive suction head (NPSH) by increasing the size of the pump suction line to a 30-foot length of 4-inch diameter hose, eliminating restrictions in the flowpath to the isolation condenser, and planning to review DOA 0010-04 and revise the procedure as appropriate to address the height above the water to which the pump could be elevated to maintain NPSH.

This finding was more than minor because it affected the equipment performance and procedure quality attributes of the Mitigating Systems cornerstone, and affected the cornerstone objective of ensuring the reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The issue was of very low safety significance based on the low initiating event probability and, because of the slow onset of the flooding and the reduced decay heat in the reactor core at the time recovery actions would be necessary, the licensee would be able to reasonably perform recovery actions that would prevent core damage. The primary cause of this finding was related to the cross-cutting issue of problem identification and resolution (corrective action program) because the licensee failed to take appropriate corrective actions to address safety issues and adverse trends in a timely manner, commensurate with their safety significance and complexity (P.1(d)). (Section 4OA3.3)

B. Licensee-Identified Violations

No findings of significance were identified.

REPORT DETAILS

Summary of Plant Status

Unit 2 began the inspection period at 912 MWe (95 percent thermal power and 100 percent of rated electrical capacity).

- On May 4, 2007, the unit was manually scrammed following a loss of feedwater flow event due to condensate demineralizer pre-filter bypass valves going closed. The unit returned to full power on May 10, 2007.
- On May 19, 2007, power was reduced to approximately 87 percent to perform a control rod pattern adjustment and turbine valve testing, and returned to full power the same day.
- On June 16, 2007, power was reduced to approximately 87 percent to maintain the unit within main condenser vacuum limits due to operating with two circulating water pumps. A circulating water pump had to be secured due to a trip of the 2/3 A lift pump on low lube water flowrate. The unit returned to full power on the same day.
- On June 30, 2007, power was reduced to approximately 87 percent to perform a control rod pattern adjustment, and returned to full power on the same day.

Unit 3 began the inspection period at 912 MWe (95 percent thermal power and 100 percent of rated electrical capacity).

- On April 9, 2007, power was reduced to approximately 97 percent to perform control rod drive time testing, and returned to full power on the same day.
- On May 17, 2007, power was reduced to approximately 97 percent to perform control rod drive time testing, and returned to full power on the same day.
- On May 18, 2007, power was reduced to approximately 97 percent due to an unplanned scram of control rod drive F-2. The unit returned to full power on the same day.
- On May 27, 2007, power was reduced to approximately 72 percent to perform turbine valve testing, control rod drive testing, a control rod pattern adjustment, and various other activities. The unit returned to full power on the same day.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather (71111.01)

a. Inspection Scope

The inspectors conducted inspections on the following equipment and systems:

- Unit 2 and 3 Station Blackout Diesels
- Unit 2 125/250 Volts-Direct Current Batteries

The inspectors selected the inspection samples listed above to ensure the equipment could perform design functions during summer conditions. The inspection reviewed the licensee's operating experience, corrective action program, projected work schedule, summer readiness procedures, and Updated Final Safety Analysis Report (UFSAR) to verify facility readiness for summer high temperatures and high winds generated by summer storms. In addition to system walkdowns, a site walkdown was performed to evaluate potential vulnerabilities for missile generation during high winds or tornados. Communications protocol between the control room and the transmission system operator was also reviewed during the inspection and examples of the quality of communication were observed due to switchyard voltage issues that occurred during the inspection. Weather related conditions identified during the hot months of 2006 were verified to have been appropriately addressed through the corrective action program.

This review represented one inspection sample.

b. Findings

No findings of significance were identified.

1R02 Evaluations of Changes, Tests, or Experiments (71111.02)

.1 Review of 10 CFR 50.59 Evaluations and Screenings

a. Inspection Scope

This inspection was performed to follow-up URI 05000237/2006012-02; 05000249/2006012-02 that was opened in the Modifications/50.59 inspection report (ML063200532) in 2006 and was related to a UFSAR change that credited non-safety systems to mitigate a high pressure coolant injection pump high energy line break when the licensee discovered credited safety systems (ADS/LPCI) would not be available.

b. Findings

Change of Systems Credited to Mitigate a High Pressure Coolant Injection (HPCI) Pump Room High Energy Line Break (HELB)

Introduction: The inspectors identified a Severity Level IV Non-Cited Violation (NCV) of 10 CFR 50.59 "Changes, Tests, and Experiments," having very low safety significance (Green) involving the adequacy of a 10 CFR 50.59 safety evaluation for UFSAR changes that the licensee had implemented. Specifically, the inspectors questioned the adequacy of the licensee's basis for determining that changes to UFSAR Section 3.6.1.5, "Use of Isolation Condenser and Control Rod Drive System for Safe Shutdown Following a HELB," did not require a license amendment.

Description: Unresolved Item (URI 05000237/2006012-02; 05000249/2006012-02) was opened during the 2006 NRC inspection for Evaluation of Changes, Tests, or Experiments and Permanent Plant Modifications, concerning the adequacy of the licensee's basis, described in the 10 CFR 50.59 safety evaluation, for determining that changes to UFSAR Section 3.6.1.5 did not require a license amendment.

On August 10, 2005, the licensee had completed 10 CFR 50.59 safety evaluation 2005-02-001 to support a UFSAR change to the systems credited with safe shutdown following a HPCI pump room HELB from strictly safety-related systems to a combination of safety-related and non-safety-related systems. Specifically, the licensee credited the isolation condenser for decay heat removal in lieu of the Automatic Depressurization System (ADS) and Low Pressure Coolant Injection (LPCI)/Containment Cooling Service Water (CCSW); and credited the control rod drive system for control of reactor coolant inventory in lieu of LPCI. The reason for this change as stated by the licensee, was that the original HELB analysis did not evaluate the affects of a HPCI steam line break in the HPCI room on plant equipment in other areas and that the Equipment Qualification (EQ) program only evaluated the equipment in the HPCI room that was needed to isolate the break. The licensee determined that the use of non-safety-related equipment for safe shutdown was made necessary because the postulated break in the HPCI room resulted in a harsh environment in the Unit 2/3 Emergency Diesel Generator (EDG) Room. Since the 2/3 EDG was not environmentally qualified for this environment, it was assumed to fail as a consequence of the HELB. To meet single failure requirements, the Unit 2 EDG was assumed to also fail. This scenario resulted in loss of all AC power required to power the LPCI or CCSW systems. Therefore, the original HELB analysis did not reflect the possibility that the ADS and LPCI would not be available to mitigate the HPCI room HELB.

Rationale given in the safety evaluation for why a license amendment was not required was that the NRC addressed the issue of using the isolation condenser and control rod drive systems in lieu of safety-related systems for safe shutdown following a HELB for other areas during the Systematic Evaluation Program (SEP). Specifically, the licensee took credit for a safety evaluation generated by the NRC as part of the SEP which approved use of the isolation condenser for decay heat removal and control rod drive for reactor coolant inventory control in response to a feedwater system HELB and concluded that NRC acceptance implied that the HPCI room HELB change did not result in more than a minimal increase in the likelihood of a malfunction of an SSC important to safety. There was not rationale or explanation provided to support the jump to this conclusion.

The inspectors noted that although similar non-safety equipment was relied upon for mitigating the two scenarios, the licensee had not evaluated whether the HPCI room HELB scenario was bounded by the feedwater system HELB scenario with respect to challenges to this equipment and the ability to mitigate. Specifically, the licensee's basis for concluding a license amendment was not required did not compare the two scenarios with respect to any of the criteria in 10 CFR 50.59(c)(2). In particular, the feedwater HELB is a different accident scenario when compared to the HPCI HELB in that it involves a loss of inventory.

After discussions with Office of Nuclear Reactor Regulation staff, the inspectors concluded that accepting the use of the isolation condenser and control rod drive system to respond to the event was a possible viable approach assuming that the licensee provided an adequate basis in the safety evaluation and application was restricted to within the confines of HELB evaluations. However, in this case, the licensee had failed to provide an adequate basis.

Analysis: The inspectors determined that the failure to provide an adequate basis for changes made to the facility in accordance with 10 CFR 50.59 safety evaluation was a performance deficiency. The finding was determined to be more than minor because the inspectors could not reasonably determine that the activity to change the systems credited with safe shutdown following a HPCI pump room HELB from strictly safety-related systems to a combination of safety-related and non-safety-related systems would not have ultimately required NRC prior approval. The licensee needed to address each of the criteria in 10 CFR 50.59(c)(2) to ensure the HPCI HELB event was entirely bounded by the feedwater system HELB, and the end result was not obvious to the inspectors without this more in-depth evaluation.

Because violations of 10 CFR 50.59 are considered to be violations that potentially impede or impact the regulatory process, they are dispositioned using the traditional enforcement process instead of the Significance Determination Process (SDP). However, if possible, the underlying technical issue is evaluated under the SDP to determine the severity of the violation. In this case, the inspectors completed a significance determination of the underlying technical issue using NRC's IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." The inspectors determined that the finding should be evaluated under the Mitigating Systems cornerstone as it affected the availability and reliability of mitigating equipment (LPCI and ADS). Based upon this Phase 1 screening, and the fact that this issue did not represent a loss of function, did not result in exceeding a Technical Specification allowed outage time, and did not affect external event mitigation, the inspectors concluded that the issue was of very low safety significance (Green). In accordance with the Enforcement Policy, the violation was therefore classified as a Severity Level IV violation.

The inspectors determined there was not a cross-cutting aspect to this finding.

Enforcement: Title 10 CFR 50.59(d)(1) states, in part, that the licensee shall maintain records of changes in the facility, of changes in procedures, and of tests and experiments as described in the UFSAR. These records must include a written evaluation which provides a basis for the determination that the change, test, or experiment does not require a license amendment.

Contrary to the above, the licensee, in August 2005, in safety evaluation 2005-02-001, failed to provide an adequate basis for why changes that credited the isolation condenser for decay heat removal in lieu of ADS and LPCI/CCSW and credited the control rod drive system for control of reactor coolant inventory in lieu of LPCI during a postulated HPCI room HELB did not require a license amendment. In accordance with the Enforcement Policy, this violation of the requirements of 10 CFR 50.59 was classified as a Severity Level IV Violation because the underlying technical issue was of very low safety significance. Because this non-willful violation was non-repetitive, and was captured in the licensee's corrective action program (CAP 00534275), it is considered an NCV consistent with VI.A.1 of the NRC Enforcement Policy.
(NCV 05000237/2007003-01; 05000249/2007003-01)

1R04 Equipment Alignment (71111.04Q)

a. Inspection Scope

The inspectors selected a redundant or backup system to an out-of-service or degraded train to determine that the system met the design of the UFSAR. Piping and instrumentation diagrams were used to determine correct system lineup and critical portions of the system configuration were verified. Instrumentation, valve configurations, and appropriate meter indications were also observed. The inspectors observed various support system parameters to determine the operational status of systems. Control room switch positions for the systems were observed. Other conditions, such as adequacy of housekeeping, the absence of ignition sources, and proper labeling were also evaluated.

The inspectors performed partial equipment alignment walkdowns of the following systems:

- Unit 2 and Unit 3 reactor building containment cooling water piping following the failure of the 2/3 reactor building containment cooling water pump.

This represented one inspection sample.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05Q and A)

.1 Routine Quarterly Inspections

a. Inspection Scope

The inspector conducted a tour of the areas listed below to assess the material condition and operational status of fire protection features. The inspectors verified that combustibles and ignition sources were controlled in accordance with the licensee's administrative procedures; that fire detection and suppression equipment was available for use and access was not obstructed; that passive fire barriers were maintained in good material condition; that procedures were maintained and adequate to support fire fighting activities; and that compensatory measures for out-of-service, degraded, or inoperable fire protection equipment were implemented in accordance with the licensee's fire plan. Minor deficiencies noted during this inspection were verified to be included in the licensee's corrective action program. Documents reviewed are listed in the Attachment. The following areas were walked down:

- Unit 3 turbine building, 517' elevation, reactor feed pumps, Fire Zone 8.2.5.E;
- Station blackout diesel generator building, North and South Ends, Non-Power Block, Fire Zones: N/A;
- Unit 2 battery and charger turbine building, 549' elevation, Fire Zones 7.0.A.1, 7.0.A.2, 7.0.A.3 and 8.2.7;
- Unit 2 reactor building, 476' 6" elevation, high pressure coolant injection room, Fire Zone 11.2.3;

- Unit 2 turbine building, 517' elevation, reactor feed pumps, Fire Zone 8.2.5.A; and
- Unit 3 turbine building, 517' elevation, diesel generator, Fire Zone 9.0B.

This represented six inspection samples.

b. Findings

No findings of significance were identified.

.2 Drill Evaluation

a. Inspection Scope

The inspectors observed a fire drill conducted on June 12, 2007, to assess the material condition and operational status of fire brigade equipment, and to assess the readiness and ability of the on-shift fire brigade to combat a simulated fire. The inspectors verified that the correct number of fire brigade members participated in the drill; that they were qualified in accordance with the licensee's administrative procedures; that the fire brigade members arrived at the scene of the simulated fire in a timely manner; that each member of the fire brigade correctly donned protective clothing and self-contained breathing apparatus; that the fire brigade leader demonstrated adequate control, communications, and utilized a copy of the licensee's pre-fire plan for the drill area; that the fire hose lines and other equipment were adequate for the situation; that the licensee conducted a post-drill critique; and that at the conclusion of the drill the fire fighting equipment was returned to a condition of readiness. Minor deficiencies noted during this inspection were verified to be included in the licensee's corrective action program. This represented one inspection sample.

b. Findings

Introduction: The inspectors identified a Non-Cited Violation (NCV) of the Dresden Nuclear Power Station Renewed Facility Operating License having very low safety significance (Green) for the licensee's failure to restore fire brigade equipment to a ready status in a timely manner after the performance of a fire drill.

Description: The inspectors observed a fire drill conducted between 4:00 p.m. and 5:18 p.m. on June 12, 2007. During the course of the drill the fire brigade members donned and used the four of the six available Scott Air Packs (self-contained breathing apparatus) during the course of the drill. Inspection Procedure 71111.05, Section 02.02.r, has the inspectors check that, "at the conclusion of the drill, all fire fighting equipment is returned to a condition of readiness to respond to an actual fire." The inspectors checked the fire brigade equipment storage location in the turbine building at about 2:15 p.m. on June 13, 2007, with the fire marshal present, and observed that the four Scott Air Packs used during the drill had not been cleaned, recharged with air, and returned to the fire brigade equipment storage area.

The fire marshal wrote an issue report and prompted radiation protection department management who had been tasked with performing the above activity. The shift logs documented that the fire brigade equipment was returned to a ready status at 8:22 p.m. on June 13, 2007.

Analysis: The inspectors determined that the failure to restore fire brigade equipment to a ready status after the performance of a fire drill, was a performance deficiency warranting a significance evaluation. Using IMC 0612, Appendix B, "Issue Screening," issued on November 2, 2006, the inspectors determined that this finding was more than minor because the failure to restore the fire brigade equipment to a ready status, if left uncorrected, would become a more significant safety concern.

The inspectors completed a Phase 1 significance determination of this issue using IMC 0609, "Significance Determination Process," Appendix A, Attachment 1, dated March 23, 2007. The inspectors determined that the finding affected the Mitigating Systems cornerstone and the fire protection defense-in-depth strategies. However, as discussed by IMC 0609, Appendix A, Attachment 1, issues related to performance of the fire brigade are not included in IMC 0609, Appendix F, "Fire Protection Significance Determination Process," 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 the condition existed for slightly more than 24 hours, the delay in getting additional self-contained breathing apparatus would be a maximum of about 10 minutes, and the majority of the safety significant equipment in the turbine building is protected by an automatic fire suppression system. The primary cause of this finding was related to the cross-cutting issue of human performance (work practices) because the licensee failed to ensure adequate supervisory and management oversight of work activities (restoration of the fire brigade equipment), such that nuclear safety was supported (H.4(c)).

Enforcement: The inspectors determined that the licensee's failure to restore fire brigade equipment to a ready status after the drill performance, was a violation of the Dresden Nuclear Power Station Renewed Operating License. License conditions 2.E 23 and 3.G of the Dresden Nuclear Power Station Renewed Facility Operating Licenses for Unit 2 and Unit 3, respectively, 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 for the facility...." Section 9.5.1, "Fire Protection System," of the Dresden UFSAR states that "The design bases, system descriptions, safety evaluations, inspection and testing requirements, 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 2.5.3, "Equipment," of Dresden Station Units 2 and 3 Fire Protection Reports, Volume 1, "Updated Fire Hazards Analysis," specifies that "The fire brigade is provided with sufficient equipment to perform manual fire suppression operations as required.

Full personnel protective gear, including self-contained breathing apparatus with reserve breathing air is provided."

Procedure OP-AA-201-003, Revision 8, "Fire Drill Performance," Paragraph 4.4.5, stated, (at the completion of the drill), "Return all Fire Brigade equipment back to a ready condition."

Contrary to the above, from about 5:18 p.m. on June 12, 2007, until about 8:22 p.m. on June 13, 2007, the fire brigade equipment had not been returned to a ready status at the completion of a drill which ended at about 5:18 p.m. on June 12, 2007, because the four Scott Air-Packs (self-contained breathing apparatus) for the four fire brigade members had been discharged and were not returned to the designated fire brigade equipment storage location. This failure could have adversely impacted the fire brigade's ability to fight a fire in a timely manner. This issue was entered in the licensee's corrective action program as IR 639984. Corrective actions by the licensee included the restoration of the fire brigade equipment to a ready status at 8:22 p.m. on June 13, 2007. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program, this violation is being treated as a Non-Cited Violation, consistent with Section VI.A.1 or the NRC Enforcement Policy. **(NCV 05000237/2007003-02; 05000249/2007003-02)**

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

Unit 2 and Unit 3 containment cooling service water system vaults were inspected. The inspectors verified that flooding mitigation plans and equipment were consistent with the design requirements and risk analysis assumptions. The inspectors reviewed UFSAR Section 3.4.1.2 for internal flooding protection measures, reviewed the licensee's flooding mitigation procedures, and reviewed issue reports related to possible flood protection issues. Additionally, plant walkdowns were performed to verify design barriers were properly maintained. Penetrations between rooms, watertight doors, electrical conduit seals and covers, and room drains were inspected to verify material condition met design assumptions. The inspectors performed a review of the station's maintenance data base to verify preventative maintenance was current and equipment deficiencies were being appropriately reported and resolved. Additionally, the inspectors reviewed the maintenance rule scoping and performance criteria and determined that the function was being tracked appropriately. The corrective action program was also reviewed for the past 12 months for issues related to internal flood protection. The inspectors completed one inspection sample by completing the internal flooding review.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11Q & B)

.1 Quarterly Evaluation

a. Inspection Scope

The inspectors observed a training crew during two evaluated simulator scenarios and reviewed licensed operator performance in mitigating the consequences of events. The scenarios included multiple equipment and instrumentation failures, and the transients resulted in complex accidents that yielded the declaration of several emergency classifications. Areas observed by the inspectors included: clarity and formality of communications, timeliness of actions, prioritization of activities, procedural adequacy and implementation, control board manipulations, managerial oversight, emergency plan execution, and group dynamics.

This represented one inspection sample.

b. Findings

No findings of significance were identified.

.2 Annual Operating Test Results and Biennial Written Examination Results

a. Inspection Scope

The inspectors reviewed the overall pass/fail results of Job Performance Measure (JPM) operating tests, simulator operating tests, and the biennial written examination (required to be given per 10 CFR 55.59(a)(2)) administered by the licensee from May through June 2007. The overall results were compared with the significance determination process in accordance with NRC Manual Chapter 0609, Appendix I, “Operator Requalification Human Performance Significance Determination Process (SDP).”

This represented one inspection sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12Q)

a. Inspection Scope

The inspectors assessed the implementation of the licensee’s maintenance rule program to evaluate maintenance effectiveness for the selected systems in accordance with 10 CFR 50.65, “Maintenance Rule.” The following systems were selected based on being designated as risk significant under the Maintenance Rule, being in increased monitoring (Maintenance Rule Category a(1) group), or due to an inspector’s identified issue or problem that potentially impacted system work practices, reliability, or common cause failures:

- Unit 2 station blackout diesel generator; and
- Unit 2 isolation condenser.

The inspectors verified the licensee's categorization of specific issues, including evaluation of the performance criteria, appropriate work practices, identification of common cause errors, extent of condition, and trending of key parameters. Additionally, the inspectors reviewed the licensee's implementation of the Maintenance Rule requirements, including a review of scoping, goal-setting, performance monitoring, short-term and long-term corrective actions, functional failure determinations associated with the condition and issue report reviews, and current equipment performance status.

This represented two inspection samples.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors evaluated the implementation of the licensee’s maintenance risk program with respect to the effectiveness of the risk assessments performed before maintenance activities were conducted on structures, systems, and components and verified that the licensee managed the risk in accordance with 10 CFR 50.65, “Maintenance Rule.” The inspectors evaluated whether the licensee had taken the necessary steps to plan and control emergent work activities. The inspectors also verified that equipment necessary to complete planned contingency actions was staged and available. The inspectors completed evaluations of maintenance activities on the:

- 3B reactor protection system motor generator preventative maintenance;
- 2A core spray pump;
- Failure of central processing unit card caused the Unit 2 scram;
- Unit 3 high pressure coolant injection system out of service due to emergent work on the #3 steam supply valve due to leak; and
- Unit 2 isolation condenser unavailable due to surveillance testing.

This represented five inspection samples.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed operability evaluations and issue reports (IR) to ensure that operability was properly justified and the component or system remained available, such that any non-conforming conditions were in compliance with Generic Letter 91-18, "Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability." The review included issues involving the operability of:

- Issue Report 555432, "Root Cause Investigation Report for 3B ERV Pilot Valve";
- Issue Report 198817, "Unit 3 battery room cold";
- Issue Report 301184, "Wires in U2 EDG Local Control Panel with Defective Insulation"'
- Issue Report 638430, "Unit 2 and 3 Emergency Diesel Generator Circuit Breaker Closing Coil";
- Issue Report 555410, "MS Snubber Fails Test"; and
- Issue Report 637041, "Abnormal Sump Pumping U2 DWEDS at 20:00."

This represented six inspection samples.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17A)

Unit 3 125 volt main station battery replacement

a. Inspection Scope

The inspectors reviewed a permanent plant modification associated with the Unit 3 125 VDC replacement. The inspectors reviewed Work Order 99053808-01, "Unit 3 17 year PM Replace 125 Volt Station Main Battery," to verify that the completed activity was in accordance with the specified design requirements, and the licensing bases, and to confirm that the changes did not adversely affect any system's safety function. Design and post-modification testing aspects were verified to ensure the functionality of the modification, its associated system, and any support systems. The inspectors also verified that the modifications performed did not place the plant in an increased risk configuration.

The inspectors also used applicable industry standards to evaluate acceptability of the modifications. This represented one inspection sample.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed post-maintenance tests associated with the activities listed below to verify that procedures and test activities ensured system operability and functional capability. The inspectors reviewed the licensee's procedures to verify that the procedure adequately tested the safety function(s) that may have been affected by the maintenance activity, that the acceptance criteria in the procedure were consistent with information in the applicable licensing basis and/or design basis documents, and that the procedure had been properly reviewed and approved. The inspectors reviewed the work packages, monitored the test performance, and reviewed the test data to verify that test results adequately demonstrated restoration of the affected safety function(s).

- Unit 3 125vdc load test after battery replacement;
- Work order 9906879-01, Perform Preventive Maintenance Inspection on 480v Breaker MCC 28-1, Cubicle H2, 2A Core Spray Pump Downstream Inspection Valve 2-1402-25A; and WO 99086879-03, Unit 2, 6 Year Preventive Maintenance Inspection 480V Motor Control Center Motor Operator Valve 2-1402-25A;
- Work order 689909, D3 40M EQ REPL ASCO SOL VALVE 3-1601-20A [Replace the U3 ASCO solenoid valve on 3-1601-20A];
- 2A core spray pump post maintenance testing; and
- Work order 99053796-01, 12 Year Preventive Maintenance Inspection, 480V MCC Breaker Reactor Water Cleanup Reject to Condenser 3-1201-11.

This represented five inspection samples.

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities (71111.20)

.1 Unit 2 manual scram resulting from loss of feedwater

a. Inspection scope

On May 4, 2007, Unit 2 was manually scrammed following a loss of feedwater flow event due to condensate pre-filter bypass valves going closed. High Pressure Coolant Injection was used to restore water level. Minimum water level was -67 inches medium range indication. Main steam isolation valves closed and the Unit 2 and Unit 2/3 emergency diesel generators auto started on low low level. The isolation condenser was used to control decay heat following the event. The inspectors responded to the site, verified that equipment expected to mitigate the event worked properly, interviewed members of the operations crew, and reviewed emergency and abnormal operating procedures.

b. Findings

The findings associated with this inspection are found in Section 4OA3.3 of this report.

1R22 Surveillance Testing (71111.22)

.1 Routine Inspections

a. Inspection Scope

The inspectors observed surveillance testing on risk-significant equipment and reviewed test results. The inspectors assessed whether the selected plant equipment could perform its intended safety function and satisfy the requirements contained in TSs. Following the completion of each test, the inspectors determined that the test equipment was removed and the equipment returned to a condition in which it could perform its intended safety function.

The inspectors witnessed one reactor coolant system (RCS) leakage detection surveillance test to assess whether the structures, systems, and components met the requirements of the TSs, and the Updated Final Safety Analysis Report. The inspectors also evaluated whether the testing effectively quantified RCS leakage and demonstrated that the structures, systems, and components were operationally ready and capable of performing their intended safety functions.

The inspectors observed surveillance testing activities and/or reviewed completed packages for the tests listed below, related to systems in the Initiating Events, Mitigating Systems, and Barrier Integrity cornerstones:

- DOS 1500-17, "Containment Cooling Service Water IST [in-service test] Comprehensive/Preservice Pump Test," Revision 02;
- DIS 1500-31, "Division II Low Pressure Coolant Injection/Containment Spray, Torus Cooling, and 2/3 Core Coverage Logic System Functional Test," Revision 4;
- Issue Report 598179, "ASME Code Test Interval For Class 1 Safety Relief Valves"; and
- DOP 2000-24, "Drywell Sump Operations," Revision 13.

This represented a total of four inspection samples, of which one was in-service testing, one was reactor coolant system leakage detection, and two were routine surveillance tests.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The resident inspectors reviewed a simulator-based training evolution to evaluate drill conduct and the adequacy of the licensee's critique of performance to identify weaknesses and deficiencies. The inspectors selected a simulator scenario that the licensee had scheduled as providing input to the Drill/Exercise Performance Indicator. The inspectors observed the classification of events, notifications to off-site agencies, protective action recommendation development, and drill critiques. Observations were compared to the licensee's observations and corrective action program entries. The inspectors verified that there were no discrepancies between observed performance and performance indicator reported statistics. The simulator scenario observed resulted in a site area emergency classification and declaration.

This simulator emergency preparedness drill observation constituted a single inspection sample.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Initiating Events, Mitigating Systems, and Barrier Integrity Performance Indicator Verification

a. Inspection Scope

The inspectors reviewed, at a minimum, the most recent 24 months of licensee event reports, licensee data reported to the NRC, plant logs, issue reports, and NRC inspection reports to verify the following performance indicators reported by the licensee for the 1st Quarter of 2007:

- Unplanned scrams per 7000 critical hours, Units 2 and 3;
- Scrams with loss of normal heat removal, Unit 3; and
- Unplanned power changes per 7000 critical hours, Units 2 and 3.

The inspectors verified that the licensee accurately reported performance as defined by the applicable revision of Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline."

These performance indicator reviews constituted five inspection samples.

b. Findings

No findings of significance were identified.

.2 Data Submission

a. Inspection Scope

The inspectors performed a review of the data submitted by the licensee for the 1st Quarter 2007 performance indicators for any obvious inconsistencies prior to its public release in accordance with IMC 0608, "Performance Indicator Program."

This review was performed as part of the inspectors' normal plant status activities and, as such, did not constitute a separate inspection sample.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Routine Quarterly Review

As discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify that they were being entered into the licensee's corrective action system at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. In addition, in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's corrective action program. This review was accomplished by reviewing daily issue reports and attending daily issue report review meetings.

This represents one routine quarterly review.

.2 In-depth Review (Operator Workaround)

a. Inspection Scope

As required by Inspection Procedure (IP) 71152, Identification and Resolution of Problems, the inspectors performed a review of the licensee's corrective action program (CAP) and associated documents to evaluate operator workarounds for mitigating systems. This review was to determine if the mitigating system function was affected or if the operator's ability to implement abnormal and emergency operating procedures was affected.

b. Findings

There were no findings of significance identified. The inspectors reviewed procedure OP-AA-102-103, "Operator Work-Around Program," and the issues being tracked for program resolution. As of June 18, 2007, there were three operator challenges that were being tracked for resolution. The inspectors' review determined that the issues were appropriately characterized as operator challenges.

The inspectors also reviewed selected operations department concerns, out of tolerance items that were identified in operations narrative logs and performed control room walkdowns to identify potential operator workarounds that were not in the program. In addition, annunciator procedures DAN 902(3)-4 G-2 and G-6 were reviewed for a potential workaround that may have been formalized as corrective action for a degraded condition. The potential workaround review applied to a caution in the annunciator procedures. The caution stated that the reactor recirculation motor generator set lube oil pumps should not operate concurrently for more than a few minutes (5-10 minutes) as the oil will overheat if both pumps simultaneously run. The inspectors concluded that there were no additional workarounds identified from the sources reviewed, and that the caution was due to system design, positive displacement pumps with discharge relief valves that on actuation, return flow to the pump suction.

.3 Semi-annual Trending

a. Inspection Scope

As required by IP 71152, “Identification and Resolution of Problems,” the inspectors performed a review of the licensee’s CAP and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspector’s review was focused on AC electrical and air systems, and consisted of a six month period from January 2007 through June 2007. The inspectors reviewed multiple issue reports (IRs) generated during the time period, in an attempt to identify potential trends. The screening was accomplished as follows:

- IRs dealing with company policies, administrative issues, and other minor issues were eliminated as being outside the scope of this inspection;
- The IRs were sorted into categories involving same equipment problems, repetitive issues, reoccurring departmental problem/challenges and repeated entries into TSs. The IRs were then screened for potential common cause issues and considered for potential trends;
- The inspectors removed a group of IRs that discussed strictly programmatic problems because the inspection requirement was primarily for equipment problems and human performance issues;
- The inspectors also removed a group of IRs where their review indicated that duplicate IRs had been written for the same event or failure;
- The remaining groups, considered potential unidentified trends, were provided to the licensee for discussion in case there was extenuating information that the inspectors were not aware of; and
- Groups of IRs remaining after all of the above screening were considered trends which the licensee had failed to identify.

The inspectors then were able to make an assessment by comparing the trends identified by the licensee to those trends identified by the NRC. In addition, the inspectors reviewed a CAP nuclear oversight assessment and audit conducted during January 2007 to June 2007.

This represented one inspection sample as a semiannual review for trends.

b. Findings

There were no findings of significance identified. The inspectors determined that, within the areas reviewed, the licensee staff initiated IRs at an appropriate threshold. The IRs reviewed also identified if any repeat or similar conditions were known to have occurred. IR 594373, dated February 21, 2007, documented that transformer 22 east bank of cooling fans were not running, and that IR 538828, dated October 2, 2006, was initiated on a similar condition. The assigned action for IR 594373 was for plant engineering to determine how many fans may be degraded and evaluate the issue for impact on summer readiness. The disposition of IR 594373 documented that in the spring of 2005, several auxiliary transformer cooling fan motors failed. Because of these failures, system engineering initiated three work orders to replace all fan motors on TR 21, TR 31 and TR 22. Only the east bank of TR 22 remained to be completed.

The inspectors also identified that IR 620611 was initiated on the determination that a trip of the 3A instrument air compressor was a maintenance rule functional failure. This IR directed the appropriate evaluations and communication of information, such as, an EPIX (Equipment Performance and Information System) report to be completed. The EPIX report will result in data available for industry trends. In addition, IRs were generated by the CAP manager on increased maintenance and decreased reliability of station equipment. These IRs identified equipment trends through the use of the CAP.

.4 Assessment of the Corrective Action Program Effectiveness

a. Inspection Scope

As required by Inspection Procedure 71152, "Identification and Resolution of Problems," the inspectors performed a review of the licensee's CAP and associated documents to evaluate licensee corrective action in response to NCV 05000237/2006007-06; 05000249/2006007-06. This violation of 10 CFR 50.62 was associated with a licensee-identified material condition. The review of licensee corrective action will specifically consider the extent of condition and any generic implications.

b. Assessment

Prioritization and Evaluation of Issues

There were no findings of significance identified.

The licensee identified that the inputs to a design analysis, titled, DRE01-0066, "Dresden Unit 2 & 3 Standby Liquid Control System Discharge Piping Pressure Drop," Revision 1 were non-conservative. Some of the valves actually installed in the plant were not the same type of valves assumed to be installed in the design analysis. This ultimately resulted in a change in a design calculation that demonstrated that standby liquid control system relief valves could lift upon system initiation during an anticipated transient without scram (ATWS) event.

The impact of having some of the valves actually installed in the plant that were not the same type of valves assumed to be installed in the ATWS Analysis was that it resulted in a higher pressure drop between the pump discharge and the reactor than calculated in DRE01-0066, Revision 1. This higher pressure drop meant that pump discharge pressure had to be higher to get flow to the reactor and the SBLC system relief valves could lift during an ATWS event. The lifting of the relief valves would cause SBLC system flow to be recirculated to the system storage tank rather than injected into the reactor vessel. Due to the inability to provide continuous SBLC system design flow into the reactor vessel as required by 10 CFR 50.62.c.(4), the licensee would fail to comply with 10 CFR 50.62.c.(4) if the relief valve lifted.

The inspectors reviewed IR 487350 that linked the incorrect modeling of the check valves in the SBLC system to calculation 097/098-M-001, which was originally prepared in December 1985, by an engineering contractor. The SBLC lift check valves were incorrectly modeled as swing check valves. The cause of the incorrect modeling could not be conclusively determined by the licensee due to the length of time, and the personnel involved were not available for interview. The IR continued, by stating, several process improvements have occurred since the error that would reduce the probability of a similar error, such as, implementation of pressure drop standard NES-MS-01.1, “Fluid Piping Pressure Loss.” Also, all calculations completed by engineering contractors undergo an owners acceptance review in accordance with CC-AA-309, “Control of Design Analyses.”

The extent of condition, documented by IR 487350, was to review other pressure drop calculations performed by the same engineering contractor in the same time frame. This corrective action resulted in two additional pressure drop calculations being identified and reviewed. There were no errors identified for these additional calculations.

The inspectors reviewed IR 527285 that documented results from a self-assessment in the area of design configuration, and was initiated approximately four months after IR 487350. IR 527285 identified that there have been recent discoveries where the physical plant configuration for some systems important to safety did not match the design input used in design analyses. The assigned action for IR 527285 was to establish a cross-functional team to develop a plan to address the issues identified by the self-assessment.

A current design engineering initiative titled, “Calculation Improvement Project,” is defined by the problem statement that some calculations prepared by multiple architect engineering firms in the 1980s and 1990s lack sufficient quality. Technical and administrative issues have been identified during self-assessments, audits and NRC inspections. Factors identified by the project that support the problem statement are: 1) calculations were not properly updated and/or superseded when plant changes were implemented, 2) assumptions in existing calculations have not always been verified, 3) there have been discoveries where the physical plant configuration did not match the design input used in calculations and 4) no programmatic effort made to reconcile current vs. historical calculations. The action plan for this project is to fix known calculation issues, such as, control room and auxiliary electrical equipment room HVAC calculations, by September 2007. The project will review existing calculations to define the project scope scheduled for August 2007.

4OA3 Event Followup (71153)

.1 (Closed) Licensee Event Report (LER) 237; 249/2006-005-00: "Units 2 and 3 Control Room Emergency Ventilation Air Conditioning System Inoperable Due To Leaking Fittings"

a. Inspection Scope

Inspectors reviewed one licensee event report (LER) to ensure that the issue documented in the report was adequately addressed in the licensee's corrective actions program. The inspectors reviewed plant Technical Specifications, Action Requests (IR 00555484), operating logs, and interviewed plant personnel to verify the statements contained in the LER.

b. Findings

On November 8, 2006, the Control Room Emergency Air Conditioning System (CREVS) was unable to maintain the control room temperature within the required Technical Specification range of 70 to 80 degrees F. Following investigation, the licensee determine that CREVS was inoperable due to a loose fitting on the Refrigeration and Condensing Unit (RCU) that resulted in refrigerant leaking from the unit. The fitting was determined to be loose due to vibrations caused by the unit's operation. The licensee entered the appropriate Technical Specification action statements and made the correct notifications required by 10 CFR 50.72 and 10 CFR 50.73. The RCU failure was corrected on November 9, 2006, and CREVS was returned to service. The licensee's apparent cause evaluation determined that the event was maintenance preventable and developed corrective action to implement a new annual maintenance activity to monitor for refrigerant leaks.

This LER is closed with no associated violation.

This represents one inspection sample.

.2 (Closed) Licensee Event Report (LER) 237/2007-002-00, "Unit 2 Reactor Scram Due to Loss of Feedwater"

a. Inspection Scope

The inspectors responded to the site after being notified of the above event. The inspectors interviewed operators and licensee management after the event. The inspectors reviewed the licensee's root cause report of the event associated with issue report 625692.

This LER is closed with the associated violation.

This represents one inspection sample.

b. Findings

Introduction: A Green finding involving a Non-Cited Violation of 10 CFR Part 50.65(a)(4), was self revealed after the Unit 2 reactor scram on May 4, 2007. The licensee performed an inadequate risk assessment of emergent maintenance on the condensate pre-filter flow controller central processing unit on May 4, 2007. The maintenance performed resulted in a complete loss of feedwater and a reactor scram.

Description: On May 4, 2007, Unit 2 was manually scrammed following a loss of feedwater flow. High pressure coolant injection was used to restore water level. Main steam isolation valves closed and the Unit 2 and Unit 2/3 emergency diesel generators auto started on low low level signal. The isolation condenser was used to control decay heat and no relief valves lifted. All control rods inserted and all automatic actions took place as expected.

The licensee's initial investigation identified that at the time of the scram, work was in-progress to replace the condensate pre-filter flow controller central processing unit (CPU). The CPU had failed earlier in the day leaving the pre-filter inlet and outlet valves closed and the bypass valves open. A work activity was initiated to replace the CPU to restore operation of the pre-filter flow controller. The CPU was replaced and, after re-initiation of the CPU, the pre-filter inlet, outlet, and bypass valves all closed, shutting off flow to the feedwater system by dead heading the condensate pumps and removing suction from the condensate booster pumps.

The inspectors reviewed the root cause report and noted that the licensee identified that the staff lacked understanding regarding the risk associated with the CPU replacement. Personnel failed to adequately apply the work control process. The emergent work was not screened as a production risk evolution. The electrical maintenance planner and operations staff failed to associate the condensate pre-filter flow controller with the condensate system, which was on the station production risk system matrix in WC-AA-108, "Review and Screening for Reactivity Risk," Attachment 2, Revision 1, as a procedurally directed risk significant system. As a result, contingencies were not established.

The root cause report stated that the licensee's staff failed to require air isolation to the condensate pre-filter bypass valves. The condensate pre-filter bypass valves fail in the open position on a loss of air. Work planning personnel assumed the pre-filter programmable logic controller (PLC) software was fail-safe and did not perceive the CPU replacement as a production risk. This assumption was based on previous experience associated with PLC work activities. The station black-out (SBO) diesel PLC CPU, which was a similar model, had been previously replaced without incident. However, the staff failed to recognize that the condensate pre-filter system CPU replacement was a first-time evolution. This was the first PLC CPU replacement on the condensate pre-filter flow controller during power operation; unlike the SBO diesel PLC CPU replacement which was performed while the system was in stand-by operation.

The licensee's root cause report concluded that the overconfidence of the involved individuals in the previous performance of PLC CPU maintenance, combined with the belief that the software was "fail safe," resulted in not implementing the precautionary action to fail open the pre-filter bypass valves.

Procedure WC-AA-104, "Review and screening of production risk," step 2.4, stated, in part, that a production risk activity is any activity that has the potential to derate the plant. The staff's failure to screen this work as a production risk activity resulted in the lack of additional review requirements for production risk significant systems. Had the staff exercised a more rigorous challenge the transient may have been prevented.

The inspectors determined that although the licensee's risk assessment was not adequate during this event, mitigating systems operated as designed and the reactor was successfully shut down.

Analysis: The inspectors concluded that the failure to adequately perform risk assessment and management of emergent work was a performance deficiency that effected the Initiating Events cornerstone. Using IMC 0612, Appendix B, "Issue Screening," dated November 2, 2006, the inspectors determined that this finding was greater than minor because the maintenance that was performed resulted in an initiating event. Using IMC 0609, Appendix K, "Maintenance Risk Assessment and Risk Management Significance Determination Process," dated May 19, 2005, the finding was determined to be of very low safety significance (Green) since the change in incremental core damage probability (ICDP) and incremental large early release probability (ILERP) were less then 1E-6 and 1E-7, respectively. The primary cause of this finding was related to the cross-cutting issue of human performance (work practices) because the human performance prevention techniques provided to the staff (the station production risk system matrix) were not adequately followed. Personnel proceeded in the face of uncertainty and unexpected circumstances by not implementing a conservative decision making process (H.4.(a)).

Enforcement: 10 CFR 50.65(a)(4) states, in part, that before performing maintenance activities, the licensee shall assess and manage the risk that may result from the proposed maintenance activities. Contrary to the above, on May 4, 2007, the licensee failed to adequately assess the risk associated with emergent maintenance activity related to the Unit 2 condensate pre-filter PLC CPU failure and replacement. Specifically, the licensee's overconfidence in the historic performance, combined with the assumption that the PLC software was "fail safe," resulted in not implementing the precautionary action to fail open the pre-filter bypass valves. The licensee underestimated the risk associated with this work and failed to screen the work as a production risk activity which resulted in the lack of additional review requirements for work on production risk significant systems.

Corrective actions included, 1) the CPU card was replaced with the vendor support and the PLC was restarted; 2) a standing order on risk management treatment of condensate pre-filter PLC hardware activities was initiated; 3) actions for design engineering to develop a design specification for the condensate pre-filter valve CPU software were assigned; and 4) actions to proceduralize the requirements associated with performing condensate pre-filter troubleshooting, maintenance activities and system restoration were initiated. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program as IR 625692, this violation is being treated as a Non-Cited Violation (NCV), consistent with Section VI.A of the NRC Enforcement Policy. **(NCV 0500237/2007003-03).**

.3 <u>(Closed) Unresolved Item (URI) 05000237/2006010-04; 05000249/2006010-04, "Full Flow Testing of the Diesel Driven Flood Pump at Design Conditions"</u>

a. <u>Inspection Scope</u>

The inspectors reviewed this URI to ensure that the issues documented in the report were adequately addressed in the licensee's corrective action program. The inspectors interviewed plant personnel and reviewed test data, calculation results, and other documents.

b. <u>Findings</u>

<u>Introduction</u>: The inspectors identified a performance deficiency involving a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," for the failure to promptly identify and adequately correct issues with the operation and testing of the isolation condenser emergency make-up pump.

<u>Description</u>: On June 5, 2004, the inspectors identified that the licensee's abnormal operating procedure for response to external flooding, and the surveillance test procedure for the diesel driven pump necessary to provide make-up to the isolation condenser for response to external flooding, were not adequate for the circumstances (NCV 05000237/2004010-02; 05000249/2004010-02). As a corrective action the licensee planned to change the surveillance test procedure and perform a full flow test of the pump. This was documented in IR 246038. Subsequently, the inspectors performed a detailed review of IR 246038 to determine if the corrective actions were appropriate and were adequately completed.

The inspectors noted that the isolation condenser emergency make-up pump was full flow tested on November 21, 2005. On September 7, 2006, the inspectors reviewed the details of the pump test and the licensee's conclusions. The inspectors noted that the pump capacity at the maximum pump speed was measured at 298 gpm at a discharge head of 114 psig. This was only 80.5 percent of the expected capacity of 370 gpm based on the manufacturers' pump curve. The inspectors were concerned that the lower than expected flowrate may be an indication of pump degradation. The licensee tested the pump at only one point and assumed that the actual pump curve would follow the generic pump curve established by the manufacturer. The inspectors questioned the method of testing and the method used to extrapolate the flowrate to generate a pump curve.

The inspectors questioned licensee personnel to determine if any actions had been taken to address the cause of the 19.5 percent degraded pump test results, and to ensure the pump would not degrade further over time. The inspectors learned that no actions had been taken. The inspectors concluded that the lack of additional points tested on the pump curve did not ensure the pump would provide adequate flow at design conditions.

The inspectors were concerned that the licensee did not perform an adequate test to validate the performance of the emergency flood pump. An unresolved item was created to track the issue while further review was conducted by the inspectors (URI 05000237/2006010-04; 05000249/2006010-04). To address the inspectors' concerns about this issue, the licensee sent the pump to an offsite facility for additional testing.

Subsequently, in January, 2007, the inspectors reviewed the test results. The inspectors noted that the test results indicated that the emergency flood pump could not achieve the flowrate and discharge head as calculated in calculation DRE99–0035, Rev. 03 "Capacity and Discharge Head For Portable Isolation Condenser Make-up Pump To Be Used During Flood Conditions," due to inadequate Net Positive Suction Head (NPSH). The licensee stated that this was caused by friction losses in the suction hose combined with how high the pump was elevated above the flood water. Also, the licensee's procedure did not clearly state how high the pump could be lifted above the flood water before it would start to lose NPSH.

The licensee concluded that the cause of this deficiency was attributed to original sizing of the pump's suction hose. The size of the suction hose was originally based on a manufacturer recommendation when the pump was purchased in 2000. The required flowrate and discharge head at that time was 336 gpm and 247.2 feet of water. A 3-inch diameter suction line was used for this flow. However, the losses in the suction line were not re-evaluated when the required flowrate was increased to 350 gpm in 2003 as a result of Extended Power Uprate (EPU). Contributing to this error was the lack of vendor information on NPSH requirements for the pump and size of the suction line used to generate the published pump curve.

The inspectors then questioned what the impact of a higher elevation lift or longer suction hose length would have been on the ability to remove reactor decay heat, and what would have happened using Dresden Operating Abnormal Procedure DOA 0010-04, Revision 24, which was not precise regarding the elevation lift and suction hose length. To address the inspectors concerns, the licensee performed an Engineering Change (EC) 364149, Revision 00. The licensee determined that in a worst case scenario which would be the effect of combined worst case elevation and worst case hose length of 3-inch diameter suction hose, the maximum flowrate would have been 269 gpm. This flowrate was determined to be sufficient to balance reactor decay heat and provide make-up to the spent fuel pools at 30.6 hours following reactor shutdown.

Page 9 of Calculation DRE 99-0035 stated that the timetable for the licensing basis flood was given in Technical Evaluation Report (TER) C5257-421, "Hydrological Considerations," dated May 7, 1982. The report concludes that the flood waters will rise from 509 feet to 517 feet in 7 hours assuming dam gates open to 16 feet. The Technical Requirements Manual and Procedure DOA 0010-04, "Floods," both require that both units be shutdown when the river elevation reaches 509 feet. Therefore, the amount of water needed for the isolation condensers is based on decay heat 6 hours after shutdown.

It was determined that the increase in time to balance reactor decay heat from 6 hours to 30.6 hours would not have significantly impacted plant safety. DOA 0010-04 requires a unit shutdown if river levels are predicted to exceed elevation 509 feet withing 3 days. Assuming the procedure is executed as written, the reactor could have been shut down as much as 72 hours prior to the flood water reaching elevation 509 feet.

Analysis: The inspectors determined that the failure to promptly identify and adequately correct issues with the operation and testing of the isolation condenser emergency make-up pump, until prompted by the inspectors, to ensure an adequate supply of make-up water to the isolation condenser during flood conditions to prevent core damage was a performance deficiency warranting a significance evaluation in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," issued on November 2, 2006. The inspectors determined that the finding was more than minor because it (1) involved the equipment performance and procedure quality attributes of the Mitigating Systems cornerstone and (2) affected the cornerstone objective of ensuring the reliability, and capability of systems that respond to initiating events to prevent undesirable consequences.

The inspectors determined that the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Appendix A, dated March 23, 2007, because the finding was associated with the reliability of a mitigating system. The inspectors concluded that the diesel driven make-up pump would be a mitigating system in the case of the probable maximum flood (PMF). For the Phase 1 screening, the inspectors answered "No" to the first four questions under the mitigating systems column. The inspectors then went to the Phase 1 worksheet for Seismic, Fire, Flooding, and Severe Weather Criteria. Question 1 was answered "Yes." Question 2.c was answered "No." As a result, the issue was screened to be of very low safety significance, "Green." The primary cause of this finding was related to the cross-cutting issue of problem identification and resolution (corrective action program) because the licensee failed to take appropriate corrective actions to address safety issues and adverse trends in a timely manner, commensurate with their safety significance and complexity (P.1(d)).

Enforcement: Title 10 CFR Part 50, Appendix B, "Introduction," requires, in part that nuclear power plants and fuel reprocessing plants include structures, systems, and components that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public.

Title 10 CFR Part 50, Appendix B, Criterion II, "Quality Assurance Program," requires, in part, that the applicant shall identify the structures, systems, and components to be covered by the quality assurance program.

The licensee's Quality Assurance Topical Report (QATR), Revision 79, Appendix F, Section 2.2, "Quality Classification," stated, in part, that the scope of the Company's QATR includes, but is not limited to, items and activities related to safe nuclear plant operation,...this process relies on the use of the terms "Safety Related," "Augmented

Quality," and "QATR Scope." Section 2.2.1.1 of Appendix F, stated, in part, that items within the scope of the QATR are designated as "Nuclear Safety Related" or "Augmented Quality."

The isolation condenser emergency make up pump is designated "Augmented Quality," and is a mitigating system in the PMF postulated scenario in the UFSAR.

Title 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that measures be established to assure that conditions adverse to quality, such as deficiencies, defective material and equipment, and nonconformances, are promptly identified and corrected.

Contrary to this requirement, from June 5, 2004, to January 3, 2007, the licensee failed to ensure that measures be established to assure that conditions adverse to quality, such as inadequate testing, insufficiently sized suction hoses, and inadequate operational procedures, were promptly identified and adequately corrected after they were previously identified by the inspectors. Specifically, the licensee failed to ensure adequate corrective actions were taken to:

- test the isolation condenser make-up pump to its design limits;
- identify operational deficiencies such as the size and length of the suction hose to eliminate friction losses in the suction line; and
- identify and correct operational issues regarding the height to which the pump could be raised to maintain NPSH.

The licensee's corrective actions for this issue included increasing the margin in NPSH by increasing the size of the pump suction line to a 30-foot length of 4-inch diameter hose, eliminating restrictions in the flowpath to the isolation condenser, and planning to review DOA 0010-04 and revise the procedure as appropriate to address the height above the water to which the pump could be elevated to maintain NPSH. Because this issue is of very low safety significance and has been entered into the licensee's corrective action program (Issue Report 574887), this violation is being treated as a Non-Cited Violation, consistent with Section VI.A., of the NRC Enforcement Policy **(NCV 05000237/2007003-04; 05000249/2007003-04)**. Unresolved Item (URI 05000237;249/2006010-04) is considered closed.

This represents one inspection sample.

4OA5 Other Activities

.1 <u>Preoperational Testing of an Independent Spent Fuel Storage Installation (ISFSI)</u> (60854)

a. <u>Inspection Scope</u>

<u>Training</u>

The inspectors reviewed the licensee's Systematic Approach to Training program which consisted of classroom and on-the-job training to ensure all staff involved in the use of the HI-STORM cask were trained. The inspectors reviewed the training material, including the content of the manuals, visual aids, and techniques used to perform on-the-job training. The inspectors independently verified satisfactory completion of training by applicable staff by comparing training documentation to the licensee's Personnel Qualification Database. The inspectors interviewed training instructors and select individuals who were responsible for performance of specific tasks during loading to evaluate their knowledge regarding the campaign activities, the cask loading process, and use of the equipment. The inspectors reviewed training records of welders and other personnel who the licensee authorized to perform the non-destructive examination inspections to ensure that these individuals' training was current.

b. Findings

No findings of significance were identified.

.2 Preoperational Testing of an Independent Spent Fuel Storage Installation at Operating Plants (60854.1)

a. Inspection Scope

10 CFR 72.212 Evaluation

The inspectors reviewed the licensee's 10 CFR Part 72.212 evaluation to determine its acceptability and compliance with conditions set forth in the Certificate of Compliance (CoC), the Final Safety Analysis Report (FSAR) and 10 CFR Part 72 requirements in regard to the Holtec MPC-68 cask system. The inspectors interviewed personnel to ensure the Dresden Nuclear Power Station (DNPS) West ISFSI 72.212 Evaluation adequately addressed the requirements for 72.212(b)(8)(iii) for cask sales, lease or loan. The inspectors discussed with the licensee staff the need to clearly identify the two 72.212 evaluations for the West and East ISFSI pads to eliminate the possibility of confusion between the two documents.

The inspectors reviewed select documents referenced in the DNPS West ISFSI 72.212 Evaluation. The documents reviewed included a sample of operational procedures, evaluations and design of the west pad, modifications to the reactor building and the transfer route, evaluation of the dose rate limits, the quality assurance program, procurement and fabrication packages, and the emergency plan.

The inspectors reviewed records to determine that a graded approach was being applied at DNPS under the application of the Exelon Generating Company Quality Assurance Program Topical Report NO-AA-10, Revision 77. The inspectors also verified that the licensee maintained a records management system and adequately controlled a copy of the CoC and the FSAR as specified in the 72.212 Evaluation.

Demonstrations

The inspectors reviewed the loading and unloading procedures to determine if they contained commitments and requirements specified in the license, the Technical Specifications, the FSAR, and Title 10 Code of Federal Regulations, Part 72. The inspectors also reviewed the licensee's contingency procedures to examine if the necessary failure scenarios were addressed and adequate initial recovery actions were proposed to place equipment in a safe configuration.

The inspectors observed licensee personnel perform select activities associated with dry fuel storage to demonstrate their readiness to safely load fuel from the spent fuel pool into the dry cask storage system. The inspectors attended a High Level Awareness meeting and various pre-job briefs to assess the licensee's ability to identify critical steps of the evolution, potential failure scenarios and tools to prevent errors. The inspectors observed the licensee perform a number of critical activities including the transfer of the Multi-Purpose Canister (MPC) from the transfer cask (TC) to the storage cask, the installation of the storage cask lid, and transfer of the storage cask to the pad. The inspectors observed the licensee respond to an unplanned condition where the TC pool lid became bound inside the mating device, thus preventing the mating device to open fully. This condition prevented the pool lid from being removed, resulting in inability to insert the MPC into the storage cask. The inspectors evaluated the adequacy of the licensee's proposed actions to recover from the condition by first securing the heavy load, the suspended MPC, and then disassembly of the mating device, evaluation of the mating device and the lid, engineering evaluation to repair, and finally repair activities. During the demonstrations, inspectors evaluated the adequacy of the operational procedures and verified the staff's familiarity with procedures and the equipment. During the observations, the inspectors verified that adequate supervisory oversight was provided, good communication and coordination between the groups was established, and the procedures were adhered to.

Fuel Selection

The inspectors reviewed the licensee's procedure describing the process to verify that the selected fuel was appropriately characterized and selected. The inspectors reviewed a completed fuel selection package for MPC No. 83 which was to be loaded first during the campaign to verify that the licensee used criteria specified in the CoC, the Technical Specifications, and the FSAR to verify the acceptability of assemblies. The inspectors reviewed the certification guides for an individual and the associated training material used for the training to ensure that the individual was adequately trained to perform fuel selection and its verification.

Heavy Loads

The inspectors reviewed 26 Condition Reports associated with the Unit 2/3 reactor building crane generated since July 2006. The inspectors interviewed personnel and reviewed work requests to ensure the crane main hoist upper travel limit switch will not adversely impact the loading of fuel and that the licensee has taken measures to repair the switch. The inspectors also reviewed documentation for the 110 ton single failure proof crane to ensure the loads to be handled are within the crane and the transfer cask trunnion load capacity limits. The inspectors performed a detailed review of the licensee's heavy load program including maintenance, surveillance, and testing of the crane, the special lifting device and the slings.

Radiation Protection

The inspectors evaluated the licensee's radiation protection program as it related to the operation of the ISFSI including a review of the licensee's As-Low-As-Reasonably-Achievable (ALARA) Plan. The inspectors compared estimated exposure to the actual doses (in person-rem) received during the past several campaigns and the licensee's plan to ensure the doses remain ALARA during the upcoming campaign. The inspectors also reviewed the licensee's effluent control program which included review of the annual ISFSI report. The inspectors reviewed the procedures associated with radiation protection safety to verify that all dose rate limits and surveillance requirements contained in the Technical Specifications were incorporated into procedures. The inspectors also reviewed the analyses for establishing administrative dose limits for the transfer and storage cask radiation surveys, the ISFSI pad radiation surveys, and storage cask inspection. The inspectors reviewed methods of personnel monitoring. The inspectors interviewed the licensee's personnel to verify their knowledge regarding the scope of the work and the radiological hazards associated with transfer and storage of dry fuel.

b. Findings

No findings of significance were identified.

.3 Operation of an Independent Spent Fuel Storage Installation (60855.1)

a. Inspection Scope

The inspectors observed and evaluated the licensee's loading of the first canister during the campaign and the transfer of the first storage cask to the west ISFSI pad to verify compliance with the applicable CoC conditions, the associated Technical Specifications, and procedures. Specifically, the inspectors observed independent verification of the fuel assemblies, lifting of the transfer cask from the spent fuel pool, decontamination and surveying, welding of the lid, draining of water, vacuum drying, down loading of the transfer cask and transfer of the MPC to the storage cask and transfer of the storage cask to the ISFSI pad. The inspectors reviewed daily radiation dose records for the staff who performed work to verify that the radiation doses received were below the licensee's administrative limits. The inspectors also verified that the contamination and radiation levels from the transfer cask and the loaded MPC as well as the storage cask were well below the regulatory limits and the licensee's administrative limits. The inspectors reviewed select documents, in part, after the licensee completed certain loading activities. Specifically, the inspectors reviewed the welding travelers and the visual and dye penetrant testing records for MPC No. 84 to ensure that acceptance criteria were followed in accordance with codes and that the documents were complete. In addition, the inspectors reviewed a number of condition reports and the associated follow up actions that were generated in response to some unexpected conditions encountered during the loading campaign. The inspectors verified that the licensee took adequate corrective actions to correct an issue associated with loading MPC No. 84 with fuel which was designated to be initially loaded into MPC No. 83. The inspectors verified that this event did not adversely affect the MPC design or its capability to safely store fuel.

b. Findings

No findings of significance were identified.

.4 (Closed) Unresolved Item (URI) 05000237/2006012-02; 05000249/2006012-02: Change of Systems Credited to Mitigate a High Pressure Coolant Injection (HPCI) Pump Room High Energy Line Break (HELB)

An URI was opened during the 2006 NRC Evaluation of Changes, Tests, or Experiments and Permanent Plant modification Baseline Inspection concerning the adequacy of the licensee's basis, described in the 10 CFR 50.59 safety evaluation, for determining that changes to UFSAR Section 3.6.1.5 did not require a license amendment. Based on the information discussed in Section 1R02.1.b.1 of this report, an NCV of 10 CFR 50.59 was identified. Therefore, this URI is closed.

4OA6 Meetings

.1 Exit Meeting

The inspectors presented the inspection results to the Site Vice President, Mr. D. Bost, and other members of licensee management on July 17, 2007. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was discussed.

.2 Interim Exit Meetings

An interim exit meeting was conducted for the Licensed Operator Requalification Inspection 71111.11B with Mr. P. O'Connor, Licensed Operator Requalification Training Lead, on June 26, 2007, via telephone.

Closure of URI 05000237/2006012-02; 05000249/2006012-02 as an NCV with Mr. James Ellis and other members of licensee management on July 5, 2007, via telephone.

.3 Interim Exit Meetings

An interim exit meeting was conducted on April 23, 2007, at the completion of the pre-operational inspection of the dry fuel storage activities. A second interim exit meeting was held on May 11, 2007, which concluded the inspection of the loading activities and transfer of the first storage cask to the west ISFSI pad. The individuals present included:

**J. Ellis, Regulatory Assurance Manager
***J. Griffin, NRC Coordinator
*K. Hunter, Reactor Services
***D. Legget, Nuclear Oversight Manager
***M. Mikota, Reactor Services
*B. Rybak, Regulatory Assurance

*P. Salgado, Operations
**J. Sipek, Assistant Engineering Director
*C. Symonds, Training Director
***D. Wozniak, Plant Manager

* indicates individuals present at the April 23, 2007 meeting.
** indicates individuals present at the May 11, 2007 meeting.
*** indicates individuals present at the April 23 and May 11, 2007 meetings.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

D. Bost, Site Vice President
D. Wozniak, Plant Manager
C. Barajas, Senior Operations Supervisor
H. Bush, Radiation Protection Manager
J. Ellis, Regulatory Assurance Manager
R. Gadbois, Operations Director
D. Galanis, Design Engineering Manager
D. Glick, Shipping Specialist
G. Graff, Operations Training Manager
J. Griffin, Regulatory Assurance - NRC Coordinator
T. Hanley, Engineering Director
D. Hartung, Test Engineer
K. Hunter, Dry Fuel Storage Engineer
M. Kluge, Senior Staff Engineer
D. Leggett, Nuclear Oversight Manager
M. Mikota, Project Manager
P. O'Connor, Licensed Operator Requalification Training Lead
M. Overstreet, Radiation Protection
C. Podczerwinski, Maintenance Rule Coordinator
K. Purdy, Projects
E. Rowley, Chemistry
R. Rybak, Regulatory Assurance
J. Schrage, Licensing Engineer
P. Simpson, Manager of Licensing (Cantera)
J. Sipek, Assistant Engineering Director
D. Smith, Projects
J. Strmec, Chemistry Manager
C. Symonds, Training Director
B. Zanc, Training

NRC personnel

M. Ring, Chief, Division of Reactor Projects, Branch 1
D. Hills, Chief, Engineering Branch 1

IEMA personnel

R. Schulz, Illinois Emergency Management Agency

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000237/2007003-01 05000249/2007003-01	NCV	Change of Systems Credited to Mitigate a High Pressure Coolant Injection Pump Room High Energy Line Break (1R02)
05000237/2007003-02 05000249/2007003-02	NCV	Failure to Restore Fire Brigade Equipment to a Ready Status in a Timely Manner after the Performance of a Fire Drill (1R05)
05000237/2007003-03	NCV	Inadequate Risk Assessment That Led to Reactor Scram (4OA3.2)
05000237/2007003-04 05000249/2007003-04	NCV	Failure To Identify And Correct Issues With The Operation And Testing Of The Diesel Driven Pump Used To Respond To External Flooding (4OA3.3)

Closed

05000237/2007003-01 05000249/2007003-01	NCV	Change of Systems Credited to Mitigate a High Pressure Coolant Injection Pump Room High Energy Line Break (1R02)
05000237/2007003-02 05000249/2007003-02	NCV	Failure to Restore Fire Brigade Equipment to a Ready Status in a Timely Manner after the Performance of a Fire Drill (1R05)
05000237/2007003-03	NCV	Inadequate Risk Assessment That Led to Reactor Scram (4OA3.2)
05000237/2007003-04 05000249/2007003-04	NCV	Failure To Identify And Correct Issues With The Operation And Testing Of The Diesel Driven Pump Used To Respond To External Flooding (4OA3.3)
05000237/2006010-04 05000249/2006010-04	URI	Full Flow Testing of the Diesel Driven Flood Pump at Design Conditions (4OA3.3)
05000237/2006012-02 05000249/2006012-02	URI	Change of Systems Credited to Mitigate a High Pressure Coolant Injection Pump (4OA5.4)
05000237;249/2006-005-00	LER	Units 2 and 3 Control Room Emergency Ventilation Air Conditioning System Inoperable Due To Leaking Fittings (4OA3.1)
05000237/2007-002-00	LER	Unit 2 Reactor Scram Due to Loss of Feedwater (4OA3.2)

Discussed

05000237/2004010-02 05000249/2004010-02	NCV	Source of Make-up Water (4OA3.3)
05000237/2006007-06 05000249/2006007-06	NCV	Standby Liquid Control Valves Installed in the Plant Different than Those Assumed in Design Calculation (4OA2.4)

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 or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

1R01 Adverse Weather Protection

- OP-AA-108-111-1001, Revision 2; Severe Weather and Natural Disaster Guidelines
- OP-AA-102-102, Revision 5; General Area Checks and Operator Field Rounds
- WC-AA-107, Revision 3; Seasonal Readiness
- DOA 0010-02, Revision 10; Tornado Warning/Severe Winds
- IR 494561, Unexpected Tech Spec Entry Due to Storm, issued May 29, 2006
- IR 498693, Cold Cooling Tower Pump Trip, issued June, 10, 2006
- IR 514085, Failure of U3 Service Water Rad Monitor, issued July 27, 2006
- IR 627691, NER NC-07-011 Red - Summer Readiness Concerns, dated May 10, 2007
- IR 627808, NER NC-07-011 Red - Summer Readiness Concerns, dated May 10, 2007
- Dresden UFSAR Section 3.3, Wind and Tornado Loadings, Revision 4
- Dresden UFSAR Section 6.2, Containment Systems, Revision 5
- Dresden UFSAR Section 8.3, Onsite Power Systems, Revision 6

1R02 Evaluations of Changes, Tests, or Experiments

- 2005-02-001; Crediting the Isolation Condenser and Control Rod Drive Systems for Safe Shutdown; Revision 0
- IR 00534275; 1982 NRC Safety Evaluation on HELB Inconsistent with Submittals; dated September 21, 2006
- IR 00540786; NRC Questions Conclusion of Safety Evaluation 2005-02-001; dated October 06, 2006
- IR 173612; HPCI/HELB 10 CFR 50.59 Documentation; dated August 28, 2003

1R05 Fire Protection

- Dresden Unit 2 Fire Pre-Plan, Dresden Station Units 2 & 3 Updated Fire Hazards Analysis
- Fire Pre-Plan, Unit 1/2/3 SBO North End, Non-Power Block, Revision 6; Fire Zone - N/A
- Fire Pre-Plan, Unit 1/2/3 SBO South End, Non-Power Block, Revision 6; Fire Zone - N/A
- Fire Pre-Plan, Unit 2 Turbine Building, 549' Elevation, Battery Rooms; Fire Zones - 7.0.A.1, 7.0.A.1, 7.0.A.2, 7.0.A.3 8.2.7
- Fire Pre-Plan, Unit 2 Turbine Building, 517' Elevation, Reactor Feed Pumps; Fire Zone 8.2.5.A
- Fire Pre-Plan, Unit 3 Turbine Building, 517' Elevation, Diesel Generator; Fire Zone 9.0.B

1R06 Flood Protection Measures

- Operating Experience Smart sample (OpESS) FY 2007-02, Flooding Vulnerabilities Due to Inadequate Design and Conduit/ Hydrostatic Seal Barrier Concerns
- IN 98-31, “Fire Protection System Design deficiencies and Common-Mode Flooding of Emergency core Cooling Rooms at Washington Nuclear Project Unit 2,” August 18, 1998

- IN 05-30, "Safe Shutdown Potentially Challenged by Unanalyzed Internal Flooding Events and Inadequate Design," November7, 2005
- Dresden UFSAR Section 3.4, Water Level (Flood) Design, updated June 2007
- Dresden UFSAR Section 9.2.1, Containment Cooling Service Water Systems, updated June 2007
- Dresden Drawing FL-1, Flood Barriers - Basement Floor; Revision A
- IR 634651, DOS 4400-01 Acceptance Criteria Appears to be Conservative; May 29, 2007
- IR 633578, CCSW Vault Draining At a Slow Rate After Leakage into Vault; May 24, 2007
- WO 831146-0, Perform DOS 1500-2, CCSW Pump Vault Watertight Door Leak Test; May 25, 2007
- DOS 1500-2, CCSW Pump Vault Watertight Door Leak Test; Revision 01
- WO 792769-01, Perform DOS 4400-01 for U3; May 25, 2007
- DOS 4400-01, Containment Cooling Service Water Vault Floor Drain Surveillance; Revision 9
- DOA 0040-02, Localized Flooding in Plant; Revision 20
- System Z15 Maintenance Rule Failure Report
- MA-AA-716-026, Housekeeping and Material Condition; Revision 5

1R11 Licensed Operator Requalification Program

Simulator Exercise Guides:
- OPEX-Y; Revision 4
- OPEX-AE; Revision 6

1R12 Maintenance Effectiveness

- Dresden System ZZ103 Equipment Failure Report for January 2005 to January 2007
- System ZZ103; Station Blackout (SBO) Diesel Generator Performance Detail Evaluation for 01 March 2005 to 31 March 2007 dated April 26, 2007
- WO 0810021, D2 2Y PM SBO DG Mechanical Maintenance Inspection; April 30,2007
- DMS 6600-04, Revision 9; SBO Diesel Generator Mechanical Inspection and Preventative Maintenance Procedure
- VETIP Binder D2028, Inland Diesel Operating and Maintenance Manual
- IR 577267, U2 & U3 SBO DGs Maintenance Rule Unavailability Time Review Needed; January 10, 2007
- IR 519051, Follow-up to IR 483591 (Potential SBO Low Airflow); August 11, 2006
- IR 544735, SBO DG Hydramotor PM Discrepancies; October 16, 2006
- IR 585997, Hydromotor Did Not Stroke; January 31, 2007
- IR 526080, Unit 2 SBO Secondary Ventilation Fan Failed to Start; August 31, 2006
- EC 359733, Re-orient the U2 SBO Engine AC Continuous Lube Oil Pump/Motor (2-6620-8A and 8b); approved February 21, 2007
- IR 448800, DP Switch Can Not be Valved In; February 1, 2006
- IR 455731, ISO Bldg Heater Fan Froze Up; February 19, 2006
- IR 503273, ISOL Condr Demin Wtr M-U Trouble Alarm; July 4, 2006
- IR 574887, Inadequate NPSH for Emergency Flood Pump; January 3, 2007
- IR 575918, NOD IDs No IR When Flood Pump Did Not Meet Calc Flow; January 5, 2007
- IR 168251, 2/3B ISOL Condenser M/U Pump Fail to Start; July 18, 2003
- IR 506273. ISOL Cndr Demin Wtr M/U Trouble Alarm; July 4, 2006

1R13 Maintenance Risk Assessments and Emergent Work Control

- IR 609137, "DOS 6600-14 EDG Availability May Need Revision"

1R15 Operability Evaluations

- WO 99225212-03, D3 2RFL TS IST RV - Replace B Electromatic RV, MM-3B Replace Electromatic Pilot Valve Assembly
- WO 99225212-02, D3 2RFL TS IST RV - Replace B Electromatic RV, EM RFL PM Maint/Surv on 3-203-3B Electromatic
- Vendor Equipment Technical Information Program D1991, Consolidated Electromatic Relief Valve (ERV), Revision 2
- IR 568293, "Failure Analysis report" and EACE under IR 555410-02, IR 582712, "Snubber serial # discrepancy"
- ER-AA-330-010, Snubber Functional Testing, Revision 3
- DTS 0020-02, Snubber Functional Performance Criteria, Revision 15
- 10CFR21 Reportable Defect, Diesel Generator Units Applied as Standby Power Suppliers for Nuclear Power Generating Stations, Morrison-Knudson Company, dated July 10, 1989
- DES 8300-19, Unit 3 125 Volt Main Station Battery Modified Performance Test, Revision 10
- IR 617925, Seismic Restraint of 125V DC Batteries in D3 Battery Room, April 16, 2007
- CC-AA-309-101, Revision 8, EC 00365706 Evaluation of the time period for taking batery cell (specific gravity) readings after a recharge
- IR 622756, Eval Needed for U3 125 V Battery Repl Cell from Quad, April 27, 2007
- OP-AA-108-115, Revision 1, Attachment 1, Operability Evaluation 07-005, Rev. 000; EC OPEV 365926 Revision 000
- IR 629308, CDBI - 4 Kv Breaker Closing Coil Voltage Acceptance Criteria, May 14, 2007

1R17 Permanent Plant Modification

- WO 99053808-01, D3 17Y PM Replace 125V Sta Main Batt
- WO 99053808-10, D3 17Y PM Replace 125V Sta Main Batt
- WO 99053808-11, D3 17Y PM Replace 125V Sta Main Batt

1R19 Post-Maintenance Testing

- WO 689909, D3 40M EQ REPL ASCO SOL VALVE 3-1601-20A; June 4, 2007
- DOS 1600-28, Revision 14; Air Operated Valve Fail Safe and Accumulator Integrity Test
- DOS 1600-05, Revision 36; Unit 3 Quarterly Valve Timing
- WO 817781, D2 2Y EQ 2A CS PMP Motor E.Q. Surv; May 25, 2007
- Dresden UFSAR, Revision 5, Section 6.3.2

1R22 Surveillance Testing

- 1998 OM Code, through 2000 Omb Addenda Section I-1330(a)
- Technical Specification 5.5.6 Inservice Testing Program
- DOS 1500-17, Revision 02; Containment Cooling Service Water IST Comprehensive/ Preservice Pump Test
- WO 802340-01, D2 2Y TS CCSW Pump Comprehensive Operating Test and IST Surveillance
- WO 957624-01, D3 QTR TS CCSW Pump Operability Test and IST Surveillance

- Calculation DRE97-0252, Revision 3; Sizing of CCSW Pipe & Flow Limiting Orifices for ECCS Room Coolers
- IST-DRE-BDOC-V-27, Revised April 2, 2007; Dresden Inservice Testing Bases Document for 2-3999-252 (Unit 2 Service Water Supply to ECCS Room Coolers Check Valve)
- MA-AA-716-040, Revision 4; Control of Portable Measurement and Test Equipment Program
- IR 608929, issued March 26, 2007; NOS Identified Operations Procedure Deficiencies
- IR 612389, issued April 3, 2007; NRC Questions Use of M&TE and Testing Acceptance Criteria

1EP6 Drill Evaluation
Procedures:
- EP-MW-114-100-F-01; Nuclear Accident Reporting System (NARS) Form; Revision 1
- EP-AA-111; Emergency Classification and Protective Action Recommendations; Revision 11
- EP-AA-1004; Radiological Emergency Plan Annex for Dresden Station; Revision 22
- LS-AA-1020; Reportability Tables and Decision Trees; Revision 11

4OA1 Performance Indicator Verification

Dresden Unit 2 Performance Indicator Data Packages:
- Safety System Functional Failure Data Index; 2005 – 2007
- Unplanned Power Changes per 7,000 Critical Hours; April 2005 – March 2007

Dresden Unit 3 Performance Indicator Data Packages:
- Safety System Functional Failure Data Index; 2005 – 2007
- Unplanned Power Changes per 7,000 Critical Hours; April 2005 – March 2007
- Unplanned Scrams with Loss of Normal Heat Removal; April 2005 – March 2007

- Dresden Policy 86; Safety System Functional Failures Performance Indicator Data Collection; Revision 3

4OA2 Identification and Resolution of Problems

- IR 488251; Non-Conservative Inputs Used in SBLC Pressure Drop Calc
- IR 487350; Standby Liquid Control Valves Do Not Match Calculation
- IR 527285; Dresden Mid-Cycle Assessment - Design Configuration
- OP-AA-102-103; Operator Work-Around Program; Revision 1
- DAN 902(3)-4 G-2; A1/A2 AC Oil PP Auto Start; Revision 7
- DAN 902(3)-4 G-6; B1/B2 AC Oil PP Auto Start; Revision 6
- OWA/OC Report #45; Unit 2 EHC Pressure Regulator Continues to Drift; March 4, 2005
- OWA/OC Report #50; Unit 2 CRDs; August 3, 2005
- OWA/OC Report #52; Unit 2 Fuel Pool Filter Frequent Backwashing; June 28, 2006
- GE BWR Services Information Letter (SIL) No. 523; Motor Oil Cooling Coil Corrosion; August 29, 1990
- IR 638911; Loss of U3 RPS Bus A
- IR 638117; Alarm: Main TR 2 Trouble
- IR 634669; Received Alarm: 903-8 C-1 Title: Aux TR 31 Trouble
- CR 620001; U2 Bus Duct Temperature High
- IR 636375; 4KV Bus Overvoltage for Bus 24-1
- IR 628654; 2B SBO DC Lube Oil Pump Failed to Start

- IR 632450; Elevated Vibration Levels on the 3B RPS MG Set
- IR 632739; TR-22 Bank 2 Fans Breaker Found Tripped
- IR 627504; Bus Cooling Fan Alarm Relay Chattering
- IR 627930; TR 22 High Tank Pressure
- IR 623288; Received Alarm: 902-8 G-2 Title: Bus 21/22 BRKR Cooling
- IR 616813; TR 12 (1-6412) Abnormal Cooling Pattern on Cooling Fins
- IR 617286; Issues While Transferring TR 12 Loads to TR 13
- CR 592900; TR 32 Load Tap Changer Oil Level Offscale High
- IR 598486; DOA 6500-12 Entered
- IR 593280; Damaged Cooling Vanes on TR 39
- IR 594373; TR 22 East Bank Cooling Fan Not Running
- IR 593271; Large Delta Temp Across Cooling Fins
- IR 593393; Main Power Transformer 2, Bank 9 Top Fan Not Working
- IR 574627; TR 3 Winding Temperature 2 High Temp At 77 Degrees
- IR 577574; Entry into DOA 6500-12 For Low Switchyard Voltage
- IR 578294; Insulation Resistance Low Unit 2 Generator
- IR 572775; 2C Circ Water Pump Trip
- IR 572762; 902-8 E-8 Alarm ESS UPS Low Rectifier AC Voltage
- IR 564639; TR 31 Trouble Alarm
- IR 568598; U3 Voltage Regulator Minor Trouble
- IR 620611; 3A IAC Trip Determined to be a MR Functional Failure
- IR 598822; 2A IAC Tripped
- IR 629996; Instrument Air Pressure Regulator Has Internal Leak
- IR 629998; Instrument Air Pressure Regulator Has Internal Leak
- IR 628160; 3A Instrument Air Comp Outlet Temp Indication High
- IR 622710; Air Separators Leaking at Gasketed Joints
- IR 622769; 3A Instrument Air Dryer Switching Failure
- IR 618303; 2A IAC Oil Hose Assemblies Showing Cracks
- IR 617659; 3A IAC Tripped
- IR 617662; DOA 4700-01 Entry
- IR 619770; 2A IAC Leaking Oil From Compressor Casing Shaft Seal
- IR 580246; Instrument Air Line Are Vibrating and Wearing Thin
- IR 613757; Instrument Air TV to 2B3 FW HTR Extraction Nonreturn CK Valve Failure
- IR 592761; U1 Instrument Air Dryer Trouble
- IR 593303; Air Leak Found in Hose
- IR 589751; 2B IAC Dryer Standby Tower Does Not Depressurize
- IR 581363; 2B Instrument Air Dryer Moisture Indicator Has Air Leak
- IR 573089; Bottle Pressure Drops 2600# in Less Than 24 Hours
- IR 564759; 2B Instrument Air Compressor at 130 AMPS
- IR 565640; Control Room Compressor Found With Failed Control Relay
- IR 589182; Instrument Air Leak in Supply Tubing For DPT 3-5703-14A
- IR 590162; U1 Instrument Air Dryer Trouble
- IR 639859; 2/3 RBV and Chimney Spings Showing Increased Maintenance; June 13, 2007
- IR 639848; 2A and 2B LPCI Temp Recorder Shows Decreased Reliability; June 13, 2007
- IR 639851; Fourteen WRs for RFP Rollomatic Repair Over 16 Months; June 13, 2007
- Dresden Learning Program 1Q07 Performance Summary
- NOS Quarterly Report NOSPA-DR-07-1Q; January 1- March 31, 2007
- NOS Audit NOSA-DRE-07-01; Corrective Action Program; April 18, 2007
- Self-Assessment ASSA #567557-03; Corrective Action Program Review; June 5, 2007

4OA5 Other Activities

(60854) Preoperational Testing of an Independent Spent Fuel Storage Installation (ISFSI)

Training
- Manual; Dresden Cask Site Transportation; dated February 1, 2005
- Manual; HI-TRAC Movement & MPC Transfer; dated November 23, 2004
- Performance Training and Evaluation Form; Prepare the Trackmobile and LPT for use; dated February 2, 2005
- Performance Training and Evaluation Form; Trackmobile Operation; dated February 2, 2005
- Performance Training and Evaluation Form; Move Cask in/out of the Reactor Building; dated January 18, 2005
- Non-Station Procedure; WCP-3; Weld Material Control; Revision 8
- Non-Station Procedure; WCP-5; Weld and Base Metal Repair; Revision 4
- Non-Station Procedure; 8MN-GTAW/SMAW; Welding Procedure Specification; Revision 15
- Non-Station Procedure; 8MN-GTAW; Welding Procedure Specification; Revision 10
- Non-Station Procedure; PI-900343-04; Closure Welding of Multi-Purpose Canister; Revision 8
- Non-Station Procedure; High Temperature Liquid Penetrant Examination and Acceptance Standards for Welds, Base Materials and Cladding; Revision 02
- Non-Station Procedure; GQP-9.6; Visual Examination of Welds; Revision 8
- RXS Qual Report, qualification database

10 CFR Part 72.212
- DDP 23; Unit 1 Safety Classification for Dry Cask Storage (DCS) and Independent Spent Fuel Storage Installation (ISFSI); Revision 02
- DNPS Units 1, 2, and 3 West ISFSI 10 CFR 72.212; Revision 0; dated March, 2007.
- DNPS 10 CFR 72.212 Evaluation Report (East ISFSI); Revision 2; Section 5
- DNPS Units 2 & 3 UFSR; Section 11.3.3.3
- DRE-06-0038; HI-STORM CoC Radiation Protection Program Dose Rate Limits for Dresden; Calculation; Revision 0
- DRE-06-007; Design of Independent Spent Fuel Storage Installation (ISFSI) Reinforced Pad for Dry Cask Storage Project; Calculation; Revision 1
- DRE-97-0200; Evaluation of On-Site Transportation Route associated with Unit -1 Dry Cask Storage Project; Calculation; Revision 1B
- EGC Quality Assurance Program Topical Report NO-AA-10; Revision 77
- Engineering Change 358290; Installation of ISFSI Pad - West, 50.59 Review; Revision 00
- Engineering Change 358291; Reactor Building Modifications for Spent Fuel Cask Indoor Transfers, 50.59 Reviews; Revision 00
- EP-AA-1004; Exelon Nuclear Radiological Emergency Plan Annex for Dresden Station; Revision 22
- Holtec International; Final Safety Analysis Report for the HI-STORM 100 Cask System; NRC Docket 72-1014; Revision 4
- HU-AA-104-101; Exelon Nuclear Procedure Use and Adherence; Revision1
- MA-AA-716-011; Exelon Nuclear Work Execution and Close Out; Revision 8
- MPC Cask 68-009; procurement, fabrication, including receipt inspection package 53496 records
- MPC Cask 68-035; procurement, fabrication, including receipt inspection package 65343 records

- RM-AA-101, Exelon Nuclear Records Management Program; Revision 7

Demonstrations
- DAP 07-54; Administrative Controls for the Independent Spent Fuel Storage Installations; Revision 04
- DFP 0800-64; Point Transporter Operations; Revision 04
- DFP 0800-65; Spent Fuel Cask Site Transportation; Revision 05
- DFP 0800-68; Hi-Track Preparation; Revision 12
- DFP 0800-69; Hi-Track Movement Within the Unit 2/3 Reactor Building; Revision 08
- DFP 0800-70; Hi-Track Loading Operations; Revision 17
- DFP 0800-71; MPC Processing; Revision 19
- DFP 0800-72; Hi-Storm Processing; Revision 19
- DFP 0800-75; MPC Inspection; Revision 03
- DFP 0800-76; 4-Point Transporter Undocumented Visual Inspection; Revision 01
- DFP 0800-78; Vacuum Drying System Operation; Revision 11
- DFP 0800-82; Hi-Track/MPC Unloading Operations in the Unit 2/3 Reactor Building; Revision 02
- DOA 0800-01; Spent Fuel Cask Abnormal Conditions; Revision 00
- DFP 0850-02; New/Irradiated Fuel Damage; Revision 04
- Work Order No. 01018978-01; DSC Hi-Storm/Hi-Track Mating Device Repair/Modification; dated April 16, 2007
- 72.48 Screening/Evaluation No. 842; Grind 0.25 inches of metal off of sides of pool lid to allow it to fit through the Mating Device; dated April 12, 2007
- 72.48 Screening/Evaluation No. 812; Change in the torque requirements for the TC pool lid and MPC cleats; dated September 6, 2006
- AR 559386; Hoist Ring Socket Shears While Unloading Hi-Storm Lid; dated November 17, 2006
- AR 612497; Contingency WO for U2/3 Rx Bldg OH Crane for DSC Projects; dated April 3, 2007

Fuel Selection
- DTP 67; Fuel Selection and Documentation For Fuel Cask Loading; Revision 6
- DTP-67; Selection Package; MPC-68-083; Revision 6
- Certification Guide; Dry Cask Storage-Selection/Documentation, dated March 10, 2005
- Certification Guide; Review Requirements; dated August 17, 2005
- ESPT/Position Specific Training; Dry Cask Storage Fuel Selection, Revision 01, dated October 2006
- Issue No. 00615939, MPC Contacts Outside of the Hi-Storm Overpack During Transfer; dated April 11, 2007

Heavy Loads
- Action Request (AR) 509389; 2/3 Reactor Building Bridge Crane Remote Would Not Work; dated July 14, 2006
- AR 524715; Rx Bldg Aux Hook Will Not Move Up or Down; dated August 28, 2006
- AR 533568; Potential Platform Interference With RX Bldg Overhead Crane; dated September 20, 2006
- AR 528039; Aux Hook on Rx Bldg Crane is Cutting out Periodically; dated September 6, 2006
- AR 541891; Reactor Building Overhead Crane Trolley Mode Overheating; October 9, 2006

- AR 542550; Reactor Building Overhead Crane Deficiencies/Work Arounds, dated October 11, 2006
- AR 545526; Reactor Building Overhead Crane Auxiliary Hoist Trips with Increased Frequency, dated October 18, 2006
- AR 547819; Electrical Work Not Performed Per EC: dated October 23, 2006
- AR 552681; RBOC Limits Not Wired Per Current Design Dwgs; dated November 2, 2006
- AR 552724; Crane Cab Came in Contact With Aux Platform Jib Hoist; dated November 2, 2006
- AR 553864; Incorrect Limit Switch Installation on Crane; dated November 5, 2006
- AR 553918; Wrong Part Number ID'D For Reactor Building Crane Switch; dated November 6, 2006
- AR 555973; RX Building Crane Main Hoist Upper Limit Still an Issue; dated November 9, 2006
- AR 556319; NOS ID Reactor Building Crane Safety Concern; dated November 10, 2006
- AR 556500; Limit Switch Trouble Shooting Outside Scope of Package; dated November 9, 2006
- AR 574883; Need to Upgrade or Obtain Spare Parts For Cranes; dated January 3, 2007
- AR 608620; Rx Bldg Crane Aux Hook will not go up or down; dated March 25, 2007

Radiation Protection
- Calculation DRE06-0038; Hi-Storm CoC Radiation Protection Program Dose Rate Limits for Dresden; dated February 21, 2007
- Procedure DRS 6021-33; Independent Spent Fuel Storage Installation Radiation Survey; Revision 5
- Procedure DRP 6021-22; Hi-Storm Radiation Survey; Revision 2
- Procedure DRP 6021-21; Hi-Track Radiation Survey; Revision 3
- Procedure DFP 0800-63; Hi-Storm Inspection; Revision 1
- RWP No. 10004822; ALARA Plan; U-2/3 Dry Cask Storage Project 2005 (3 casks)
- RWP No. 10007365; ALARA Plan; U-2/3 Dry Cask Storage Project 2007 (5 casks)

Loading
- AR 623408; Foreign Material Found on Bundle LJU342 and Removed; dated April 30, 2007
- AR 623708; RX Building Crane Tripped out When in Restricted Mode; dated April 30, 2007
- AR 625578; Dry Cask MPC Serial No. Does Not Match Fuel Move Sheets; dated May 5, 2007
- AR 626185; Individual Contaminated In Rx Bldg Cab; dated May 7, 2007
- AR 627366; HI-Track Inner Pool Lid Found Contaminated After Down Load; dated May 9, 2007
- Work Order 00898315-02; 2007 Dry Cask Campaign Cask 2; PCI Perform MPC-Cask Welding Per PCI PI-900343-04

LIST OF ACRONYMS USED

2Y	2 Year
AC	Alternate Current
ADAMS	Agencywide Documents Access and Management System
ADS	Automatic Depressurization System
ALARA	As-Low-As-Reasonably-Achievable
ASME	American Society of Mechanical Engineers
CAP	Corrective Action program
CFR	Code of Federal Regulation
CoC	Certificate of Compliance
CPU	Central Processing Unit
CCSW	Containment Cooling Service Water
D2	Dresden Unit 2
DG	Diesel Generator
DNPS	Dresden Nuclear Power Station
DOP	Dresden Operating Procedure
DOS	Dresden Operating Surveillance
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EPU	Extended Power Uprate
EQ	Equipment Qualification
FSAR	Final Safety Analysis Report
FY	Fiscal Year
HELB	High Energy Line Break
HPCI	High Pressure Coolant Injection
IEMA	Illinois Emergency Management Agency
IMC	Inspection Manual Chapter
IR	Inspection / Issue Report
ISFSI	Independent Spent Fuel Storage Installation
IST	In-service Testing
JPM	Job Performance Measure
LPCI	Low Pressure Coolant Injection
MCC	Motor Control Center
M&TE	Maintenance and test Equipment
MPC	Multi-Purpose Canister
MWe	megawatts electrical
NCV	Non-Cited Violation
NER	Nuclear Event Report
NOS	Nuclear Oversight
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
PARS	Publicly Available Records
PI	Performance Indicator
PLC	Programmable Logic Controller
PM	Preventative Maintenance
QTR	Quarterly

RCS	Reactor Coolant System
SBO	Station Blackout
SDP	Significance Determination Process
SRP	Standard Review Plan
SSC	Structure, System, or Component
TC	Transfer Cask
TS	Technical Specification
U2	Unit 2
U3	Unit 3
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved item
VETIP	Vendor Equipment Technical Information Program
WO	Work Order