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

February 4, 2011

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

SUBJECT: LASALLE COUNTY STATION, UNITS 1 AND 2
NRC INTEGRATED INSPECTION REPORT
05000373/2010005; 05000374/2010005; 07200070/2010001

Dear Mr. Pacilio:

On December 31, 2010, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your LaSalle County Station, Units 1 and 2. The enclosed report documents the results of this inspection, which were discussed on January 13, 2011, with the Site Vice President, Mr. David Rhoades, 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, three NRC-identified and one self-revealed finding of very low safety significance were identified. The findings involved violations of NRC requirements. However, because of their very low safety significance, and because the issues were entered into your corrective action program, the NRC is treating the issues as non-cited violations (NCVs) in accordance with Section 2.3.2 of the NRC Enforcement Policy. Additionally, a licensee identified violation is listed in Section 4OA7 of this report.

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

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

Sincerely,

/RA/

Kenneth Riemer, Chief
Branch 2
Division of Reactor Projects

Docket Nos. 50-373; 50-374; 72-070
License Nos. NPF-11; NPF-18

Enclosure: Inspection Report 05000373/2010005; 05000374/2010005; 07200070/2010001
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	05000373; 05000374; 07200070
License Nos:	NPF-11; NPF-18
Report No:	05000373/2010005; 05000374/2010005; 07200070/2010001
Licensee:	Exelon Generation Company, LLC
Facility:	LaSalle County Station, Units 1 and 2
Location:	Marseilles, IL
Dates:	October 1, 2010, to December 31, 2010
Inspectors:	G. Roach, Senior Resident Inspector F. Ramírez, Resident Inspector/Acting Senior Resident N. Shah, Region III Project Engineer M. Learn, Region III Reactor Engineer, MCID, DNMS R. Edwards, Region III Reactor Engineer, MCID, DNMS R. Jickling; Region III Senior EP Engineer, DRS C. Moore, Region III Operations Engineer, DRS V. Meghani, Region III Reactor Inspector, DRS J. Neurauter, Region III Reactor Inspector, DRS B. Palagi, Region III Senior Operations Engineer, DRS R. Temps, Senior Safety Inspector, NMSS/DSFST J. Yesinowski, Illinois Dept. of Emergency Management
Approved by:	Kenneth Riemer, Chief Branch 2 Division of Reactor Projects

Enclosure

TABLE OF CONTENTS

SUMMARY OF FINDINGS

IR 05000373/2010-005, 05000374/2010-005, 07200070/2010-001; 10/01/2010 - 12/31/2010; LaSalle County Station, Units 1 & 2; Followup of Events and Licensee Event Reports; Other Activities.

This report covers a three-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. Two Green findings and two Severity Level IV violations were identified by the inspectors. These findings were considered non-cited violations (NCVs) of U.S. Nuclear Regulatory Commission (NRC) regulations. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP); the cross-cutting aspects were determined using IMC 0310, "Components Within the Cross-Cutting Areas." Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealed Findings

Cornerstone: Mitigating Systems

- Green. A finding of very low safety significance (Green) and an associated NCV of Technical Specification (TS) 5.4.1, "Procedures", was self-revealed, for the failure to follow procedural guidance specified in procedure MA-AA-716-210, "Performance Centered Monitoring Process." Specifically, a control relay for the Unit 2 Division 3 switchgear room ventilation was inappropriately classified for its preventive maintenance schedule and had a recommended replacement frequency of 'as required' instead of the 10 year frequency required, by procedure, for this type of equipment. As a result, when this relay failed, it caused the switchgear room ventilation system (VD) to trip and the unexpected unavailability and inoperability of the Unit 2 high pressure core spray (HPCS) system.

  The inspectors determined that the finding was of more than minor significance because it affected the Mitigating Systems Cornerstone attribute of Human Performance (human error pre-event), and it affected the cornerstone objective to ensure availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, since HPCS is a single train, this constituted a loss of safety function. The finding was determined to be of very low safety significance using an SDP Phase 3 analysis. As part of the corrective actions for this issue, the licensee re-classified the control relay to Critical, high duty cycle, to help ensure that replacement of the component occurs at the appropriate time-based frequency. The inspectors did not identify a cross-cutting aspect associated with this finding. (Section 4OA3)

Cornerstone: Initiating Events

- Green. During an inspection of pre-operational testing activities of an independent spent fuel storage installation (ISFSI) at the LaSalle County Station, the inspectors identified a finding of very low safety significance with an associated NCV of Part 10 of the Code of Federal Regulations (CFR) Part 50, Appendix B, Criterion III, "Design Control," for the

licensee's failure to perform adequate evaluations to upgrade the single failure proof crane. Specifically, the inspectors identified five examples where the licensee failed to perform adequate evaluations in accordance with American Society of Mechanical Engineers (ASME) NOG-1-2004, "Rules for Construction of Overhead and Gantry Cranes (Top Running and Bridge, Multiple Girder)," requirements. The reactor building crane was designed to meet Seismic Category I requirements, and the licensee used compliance with ASME NOG-1-2004 as the design basis for their crane upgrade to a single failure proof crane. The inspectors determined that the failure to perform adequate evaluations was contrary to ASME NOG-1-2004 requirements and was a performance deficiency. The licensee documented the conditions in Issue Report (IR) 957014, IR 1093028, and IR 1098435 and initiated actions for calculation revisions and field modifications.

The finding was of more than minor significance because it was associated with the Initiating Events Cornerstone attribute of Equipment Performance and affected the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the failure to perform adequate evaluations affected the licensee's ability to provide reasonable assurance that loads would not be dropped during critical lifts. The inspectors evaluated the finding using IMC 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," and based on a "No" answer to all of the questions in the Initiating Events column of Table 4a, determined the finding to be of very low safety significance (Green). This finding has a cross-cutting aspect in the area of Human Performance, Work Practices because the licensee did not ensure supervisory and management oversight of work activities, including contractors, such that nuclear safety is supported (IMC 0310, H.4(c)). (Section 4OA5)

Miscellaneous Matters

- Severity Level IV. The inspectors identified an NCV of 10 CFR 72.212 (b)(2)(i)(B), "Conditions of a General License Issued Under 72.210," for the licensee's failure to perform adequate evaluations of the ISFSI pad. Specifically, the inspectors identified five examples where the licensee failed to design the ISFSI pad to adequately support the static and dynamic loads of the stored casks, considering potential amplification of earthquakes through soil-structure interaction. The licensee documented the conditions in IRs 900610, 966506 and 1102633. As an interim corrective action, the licensee provided a technical paper containing justification for partial loading of the pad with 10 casks.

  Because this violation was related to an ISFSI license, it was dispositioned using the traditional enforcement process in accordance with Section 2.2 of the Enforcement Policy. The inspectors determined that the deficiency was of more than minor significance because, if left uncorrected, a failure of the ISFSI pad could lead to a more significant safety concern. The inspectors determined that the violation could be screened using Section 6.5.d.1 of the NRC Enforcement Policy as a Severity Level IV Violation. (Section 4OA5)

- Severity Level IV. The inspectors identified an NCV of 10 CFR 72.146, "Design Control," for the licensee's failure to perform adequate evaluations to ensure compliance with 10 CFR 72.212(b)(3) and 10 CFR 72.122 (b)(2)(i). Specifically, the inspectors identified that the licensee failed to evaluate that the reactor site parameters including analyses of

tornado effects were enveloped by the cask design basis, and perform additional analysis to ensure compliance with 10 CFR 72.122(b)(2)(i). The licensee documented the condition in IR 1137279 and initiated a new calculation to demonstrate compliance.

Because this violation was related to an ISFSI license, it was dispositioned using the traditional enforcement process in accordance with Section 2.2 of the Enforcement Policy. The violation was determined to be of more than minor significance because the licensee failed to have an evaluation to assure transfer cask (HI-TRAC) integrity during a tornado event and an additional calculation was required. The licensee's new calculation determined that overturning and sliding of the HI-TRAC on the refuel floor would not occur during a tornado. Therefore, the violation screened as having very low safety significance (Severity Level IV). (Section 4OA5)

B. Licensee-Identified Violations

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

REPORT DETAILS

Summary of Plant Status

Unit 1

The unit began the inspection period operating at full power. On December 11, 2010, power was reduced to approximately 78 percent to perform control rod scram time testing, main steam isolation valve scram functional testing, a rod sequence exchange, and maintenance rod recovery actions. The unit was returned to full power on December 12, 2010, where it operated for the remainder of the inspection period.

Unit 2

The unit began the inspection period operating at full power. On December 4, 2010, power was reduced to approximately 75 percent for control rod pattern adjustment, channel distortion testing, and quarterly surveillances. The unit was restored to full power on December 5, 2010, where it operated for the remainder of the inspection period.

1. REACTOR SAFETY

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

1R04 Equipment Alignment (71111.04)

.1 Quarterly Partial System Walkdowns

a. Inspection Scope

The inspectors performed a partial system walkdown of the risk-significant Unit 1A diesel generator (DG).

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

These activities constituted one partial system walkdown sample as defined in Inspection Procedure (IP) 71111.04-05.

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

.1 Routine Resident Inspector Tours (71111.05Q)

a. Inspection Scope

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

- auxiliary building, elevation 710 (Fire Zone 4F3);
- Unit 1 cable spreading room, elevation 749 (Fire Zone 4D1);
- Unit 2 cable spreading room, elevation 749 (Fire Zone 4D2); and
- Unit 2 low pressure core spray (LPCS) pump room, elevation 694 (Fire Zone 3H4).

The inspectors reviewed areas to assess if the licensee had implemented a fire protection program that adequately controlled combustibles and ignition sources within the plant, effectively maintained fire detection and suppression capability, maintained passive fire protection features in good material condition, and implemented adequate compensatory measures for out-of-service, degraded or inoperable fire protection equipment, systems, or features in accordance with the licensee's fire plan. The inspectors selected fire areas based on their overall contribution to internal fire risk as documented in the plant's Individual Plant Examination of External Events with later additional insights, their potential to impact equipment which could initiate or mitigate a plant transient, or their impact on the plant's ability to respond to a security event. Using documents listed in the Attachment to this report, the inspectors verified that fire hoses and extinguishers were in their designated locations and available for immediate use; that fire detectors and sprinklers were unobstructed; that transient material loading was within the analyzed limits; and fire doors, dampers, and penetration seals appeared to be in satisfactory condition. The inspectors also verified that minor issues identified during the inspection were entered into the licensee's CAP. Documents reviewed are listed in the Attachment to this report.

These activities constituted four quarterly fire protection inspection samples as defined in IP 71111.05-05.

b. Findings

No findings were identified.

1R06 Flooding (71111.06)

.1 Underground Vaults

a. Inspection Scope

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

- Unit 1 circulating water and non-essential service water (SW) power and control cable vault;
- Unit 2 circulating water and non-essential SW power and control cable vault; and
- switchyard breaker control power cable vault.

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

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Resident Inspector Quarterly Review (71111.11Q)

a. Inspection Scope

On December 15, 2010, the inspectors observed a crew of licensed operators in the plant's simulator during licensed operator requalification examinations to verify that operator performance was adequate, evaluators were identifying and documenting crew performance problems, and training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

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

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

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

b. Findings

No findings were identified.

.2 Annual Operating Test Results (71111.11B)

a. Inspection Scope

The inspectors reviewed the overall pass/fail results of the individual job performance measure operating tests, and the simulator operating tests (required to be given per 10 CFR 55.59(a)(2)) administered in 2010, as part of the licensee's operator licensing requalification cycle. These results were compared to the thresholds established in IMC 0609, Appendix I, "Licensed Operator Requalification Significance Determination Process (SDP)." The evaluations were also performed to determine if the licensee effectively implemented operator requalification guidelines established in NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," and IP 71111.11, "Licensed Operator Requalification Program." The documents reviewed during this inspection are listed in the Attachment to this report.

Completion of this section constituted one biennial licensed operator requalification inspection sample as defined in IP 71111.11B.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Routine Quarterly Evaluations (71111.12Q)

a. Inspection Scope

The inspectors evaluated degraded performance issues involving the risk-significant circulating water system.

In addition, as a separate sample, the inspectors reviewed the licensee's 10 CFR 50.65 (a)(3) periodic evaluation to verify that it had been completed within the time constraints of the Maintenance Rule, that the licensee had reviewed its (a)(1) goals, (a)(2) performance criteria, effectiveness of corrective actions and the use of operating experience.

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

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

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

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

b. Findings

No findings were identified.

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

.1 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

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

- Unit 2 digital electro-hydraulic control pressure switch replacement;
- Unit 2 Division 1 core standby cooling system;
- Unit 2 A emergency diesel generator (EDG); and
- high winds and tornado watch while Unit 2 EDG was out-of-service.

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

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

b. Findings

No findings were identified.

1R15 Operability Evaluations (71111.15)

.1 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following issues:

- standby liquid control (SBLC) system test tank seismic issues;
- Unit 1 B residual heat removal (RHR) discharge check valve degradation; and
- Unit 2 reactor recirculation (RR) flow control valve seal leak.

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

This operability inspection constituted three samples as defined in IP 71111.15-05.

b. Findings

No findings were identified.

1R18 Plant Modifications (71111.18)

.1 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the following temporary modification:

- implementation of the Racklife computer model to monitor Unit 2 spent fuel pool (SFP) storage racks degradation.

The inspectors compared the temporary configuration changes and associated 10 CFR 50.59 screening and evaluation information against the design basis, UFSAR and TS, as applicable, to verify that the modification did not affect the operability or

availability of the affected system. The inspectors also compared the licensee's information to operating experience information to ensure that lessons learned from other utilities had been incorporated into the licensee's decision to implement the temporary modification. The inspectors, as applicable, performed field verifications to ensure that the modifications were installed as directed; the modifications operated as expected; modification testing adequately demonstrated continued system operability, availability, and reliability; and that operation of the modifications did not impact the operability of any interfacing systems. Lastly, the inspectors discussed the temporary modification with operations, engineering, and training personnel to ensure that the individuals were aware of how extended operation with the temporary modification in place could impact overall plant performance. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

(1) (URI) Implementation of the Racklife computer model to monitor Unit 2 spent fuel pool storage racks degradation

Introduction: The inspectors identified an unresolved item (URI) associated with the potential failure to conduct an adequate 10 CFR 50.59 evaluation for the implementation of the Racklife computer code as a method to calculate Boraflex degradation of the Unit 2 SFP. This item remains unresolved pending further review by the NRC staff.

Description: On June 26, 1996, the NRC published Generic Letter (GL) 96-04: "Boraflex Degradation in Spent Fuel Pool Storage Racks." The licensee was required to respond to this letter since the SPF for Unit 2 used Boraflex as a neutron absorber. The response required an assessment of the capability of Boraflex to maintain 5 percent sub-criticality margin and a description of the proposed actions if this margin could not be maintained by Boraflex. The licensee responded to GL 96-04 on November 6, 1996, by providing an assessment of the Boraflex condition in the Unit 2 SFP. The assessment was based on coupon testing, rack exposure management and the margin to criticality existing at the time. In this response, Racklife is mentioned as an Electrical Power Research Institute (EPRI)-sponsored calculational model that is under development and the licensee stated that the Racklife model's predictions would be used in the future to support the unit 2 SFP rack management strategy and to identify the need for additional activities to offset any degradation.

In 2005, through a 50.59 Screening, the licensee revised the UFSAR Section 9.1.2.2 "Unit 2 Spent Fuel Pool" to describe a comprehensive Boraflex monitoring program that included Boraflex coupon surveillance (onsite and off-site). In addition, the change to the UFSAR added periodic neutron blackness testing (Badger testing) and the use of EPRI's Racklife computer code to model Boraflex degradation. Subsequently, in 2006, an additional 50.59 Screening was performed to again revise Section 9 of the UFSAR to specify that the licensee will conduct Badger testing every 3 years for as long as Boraflex is credited to help control the Unit 2 SFP reactivity.

In accordance with licensee TS, a K_{eff} of less than 0.95 must be maintained to ensure operability of the SFP. Using a criticality analysis for the most reactive fuel, the licensee

determined that even with 57 percent cell degradation, the acceptance criterion of K_{eff} of less than 0.95 will still be met (factors for that determination include fuel enrichment, pool temperature, etc). After applying a factor of safety of 5 percent, the licensee established 52 percent degradation as the cell operability criteria. As a result, any cell that exhibits a higher percentage of degradation is declared inoperable and is unusable.

The Racklife computer model is not part of the criticality analysis that is used to meet the TS operability criteria. However, the Racklife computer model, which is run every 6 months, provides an updated percent of degradation value for each cell. This input from Racklife allows the licensee to manage the storage capacity of the Unit 2 SFP and is what the licensee uses to determine if spent fuel can be stored in any particular cell. These results are used to declare cells inoperable.

Using industry guidance provided in Nuclear Energy Institute (NEI) 96-07, Revision 1, "Guidelines for 10 CFR 50.59 Implementation," the resident inspectors determined that implementing Racklife is a departure from a method of evaluation described in the UFSAR. By implementing Racklife to help manage the Unit 2 SFP storage capacity, the licensee changed to a different method of evaluation from the one described in the UFSAR. This new method has not been approved by the NRC. The licensee's 50.59 screening document dismisses this screening question (Does the proposed activity involve an adverse change to an element of a UFSAR described evaluation methodology, or use of an alternative evaluation methodology, that is used in establishing the design bases or used in the safety analyses?) by stating the use of Racklife does not influence the criticality analysis. The inspectors plan to engage personnel in the Nuclear Reactor Regulation office to ensure that the licensee is implementing the 50.59 guidelines and processes appropriately and to ensure that the use of the Racklife computer model by all licensees is treated consistently.

An Unresolved Item is open pending further review by the NRC staff.
(URI 05000374/2010005-06)

1R19 Post-Maintenance Testing (71111.19)

.1 Post-Maintenance Testing

a. Inspection Scope

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

- Unit 2 A EDG idle start;
- Unit 1 1B reactor water clean-up pump; and
- Units 1 and 2 circulating water discharge gates.

These activities were selected based upon the structure, system, or component's ability to impact risk. The inspectors evaluated these activities for the following (as applicable): the effect of testing on the plant had been adequately addressed; testing was adequate for the maintenance performed; acceptance criteria were clear and demonstrated operational readiness; test instrumentation was appropriate; tests were performed as written in accordance with properly reviewed and approved procedures; equipment was

returned to its operational status following testing (temporary modifications or jumpers required for test performance were properly removed after test completion); and test documentation was properly evaluated. The inspectors evaluated the activities against TS, the UFSAR, 10 CFR Part 50 requirements, licensee procedures, and various NRC generic communications to ensure that the test results adequately ensured that the equipment met the licensing basis and design requirements. In addition, the inspectors reviewed CAP documents associated with post-maintenance tests to determine whether the licensee was identifying problems and entering them in the CAP and that the problems were being corrected commensurate with their importance to safety. Documents reviewed are listed in the Attachment to this report.

This inspection constituted three post-maintenance testing samples as defined in IP 71111.19-05.

b. Findings

No findings were identified.

1R20 Outage Activities (71111.20)

.1 Refueling Outage Activities – Crane and Heavy Lifts Inspection (OpESS FY 2007-003)

a. Inspection Scope

During the period from November 29, 2010, through December 3, 2010, the inspectors performed a review of the licensee's control of heavy loads program in accordance with the NRC's Operating Experience Smart Sample (OpESS) FY 2007-03, Revision 2, "Crane And Heavy Lift Inspection, Supplemental Guidance for IP 71111.20." Specifically, the inspector reviewed the licensee's upgrade of the reactor building crane load handling system to single-failure-proof equivalency for reactor vessel head lifts. Guidelines for single-failure-proof equivalence, detailed in industry initiative NEI 08-05, "Industry Initiative on Control of Heavy Loads," Revision 0, dated July 2008, have been endorsed by the NRC as indicated in NRC Regulatory Issue Summary 2008-28, "Endorsement of Nuclear Energy Institute Guidance for Reactor Vessel Head Heavy Load Lifts," dated December 1, 2008. The inspection included the following activities:

- Reviewed licensee's implementation of safe load paths, load handling procedures, and industry standards addressing the following topics: training of crane operators, use of special lifting devices, use of slings, and inspection, testing, and maintenance of the crane. The design of the crane was reviewed as part of the reactor building crane upgrade to single-failure-proof to support ISFSI heavy load handling activities (see Section 4OA5);
- Reviewed documents that demonstrated single-failure-proof equivalence for the reactor building load handling system when used for reactor vessel head lifts;
- Reviewed licensee's management of the risk associated with maintenance involving movement of heavy loads;
- Reviewed licensee's changes to the UFSAR related to the heavy loads handling program.

Documents reviewed during the inspection are listed in the Attachment to this report.

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22)

.1 Surveillance Testing

a. Inspection Scope

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

- LOS-CS-Q1, secondary containment damper operability test (Routine); and
- LOS-RH-Q1, RHR (low pressure coolant injection) and RHR SW pump and valve in-service Testing (IST).

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

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

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

Documents reviewed are listed in the Attachment to this report.

This inspection constituted one routine surveillance testing sample and one inservice testing sample as defined in IP 71111.22, Sections -02 and -05.

b. Findings

No findings were identified.

1EP4 Drill Evaluation (71114.04)

.1 Training Observation

a. Inspection Scope

Since the last NRC inspection of this program area, emergency action level and Emergency Plan changes were implemented based on the licensee's determination, in accordance with 10 CFR 50.54(q), that the changes resulted in no decrease in effectiveness of the Plan, and that the revised Plan as changed continues to meet the requirements of 10 CFR 50.47(b) and Appendix E to 10 CFR Part 50. Revisions to the emergency action levels and Emergency Plan were reviewed by the inspectors in the Exelon Nuclear Radiological Emergency Plan Annex for LaSalle Station, Revisions 30 and 31. The inspectors conducted a sampling review of the Emergency Plan changes and a review of the emergency action level changes to evaluate for potential decreases in effectiveness of the Plan. However, this review does not constitute formal NRC approval of the changes. Therefore, these changes remain subject to future NRC inspection in their entirety. Documents reviewed are listed in the Attachment to this report.

This emergency action level and emergency plan changes inspection constituted one sample as defined in IP 71114.04 05.

b. Findings

No findings were identified.

1EP6 Drill Evaluation (71114.06)

.1 Training Observation

a. Inspection Scope

The inspector observed a simulator training evolution for licensed operators on December 15, 2010, which required emergency plan implementation by a licensee operations crew. This evolution was planned to be evaluated and included in performance indicator (PI) data regarding drill and exercise performance.
The inspectors observed event classification and notification activities performed by the crew. The inspectors also attended the post-evolution critique for the scenario.

The focus of the inspectors' activities was to note any weaknesses and deficiencies in the crew's performance and ensure that the licensee evaluators noted the same issues and entered them into the CAP. As part of the inspection, the inspectors reviewed the scenario package and other documents listed in the Attachment to this report.

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

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

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

4OA1 Performance Indicator Verification (71151)

.1 Safety System Functional Failures

a. Inspection Scope

The inspectors sampled licensee submittals for the safety system functional failures Performance Indicator (PI) for Units 1 and 2 for the period from the fourth quarter 2009 through the third quarter 2010. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, dated October 2009, and NUREG-1022, "Event Reporting Guidelines 10 CFR 50.72 and 50.73" definitions and guidance, were used. The inspectors reviewed the licensee's operator narrative logs, operability assessments, maintenance rule records, maintenance WOs, IRs, event reports and NRC Integrated Inspection Reports for the period of October 2009 through September 2010, to validate the accuracy of the submittals. The inspectors also reviewed the licensee's IR database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

This inspection constituted two safety system functional failures samples as defined in IP 71151-05.

b. Findings

No findings were identified.

.2 Mitigating Systems Performance Index - Heat Removal System

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index (MSPI) - Heat Removal System performance Units 1 and 2 for the period from the fourth quarter 2009 through the third quarter 2010. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline,"

Revision 6, dated October 2009, were used. The inspectors reviewed the licensee's operator narrative logs, IRs, event reports, MSPI derivation reports, and NRC Integrated Inspection Reports for the period of October 2009 through September 2010, to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee's IR database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

This inspection constituted two MSPI heat removal system samples as defined in IP 71151-05.

b. Findings

No findings were identified.

.3 Mitigating Systems Performance Index - Cooling Water Systems

a. Inspection Scope

The inspectors sampled licensee submittals for the MSPI - Cooling Water Systems PI Units 1 and 2 for the period from the fourth quarter 2009 through the third quarter 2010. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, dated October 2009, were used. The inspectors reviewed the licensee's operator narrative logs, IRs, MSPI derivation reports, event reports and NRC Integrated Inspection Reports for the period October 2009 through September 2010, to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee's IR database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

This inspection constituted two MSPI cooling water system samples as defined in IP 71151-05.

b. Findings

No findings were identified.

4OA2 Identification and Resolution of Problems (71152)

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

a. Inspection Scope

As part of the various baseline IPs discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant

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

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

b. Findings

No findings were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

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

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

b. Findings

No findings were identified.

.3 Semiannual Trend Review

a. Inspection Scope

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

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

This review constituted one semiannual trend inspection sample as defined in IP 71152-05.

b. Findings

No findings were identified.

.4 Selected Issue Followup Inspection: LaSalle Response to Generic Letter 2008-01: "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems"

a. Inspection Scope

The inspectors reviewed the corrective actions associated with the licensee's response to GL 2008-01. The inspectors verified that the responses to the NRC were timely and that the concerns explained on the letter were adequately addressed. The inspectors ensured that all pertinent emergency core cooling, decay heat removal and containment spray systems were tested and that all potential locations for gas accumulation were identified. If air was found, the inspectors verified that the issue was adequately evaluated and addressed commensurate with its level of safety. Consideration was also given to the classification and prioritization of the resolution of the problem in accordance with its safety significance.

As part of their corrective actions and to account for some areas that were susceptible to gas accumulation, the licensee modified several operating procedures for the affected systems such as fill and vent procedures, operability tests and in-service tests. The inspectors verified these procedure changes were completed appropriately and in a timely manner. Finally, through a review of the CAP entries generated since the issuance of GL 2008-01, the inspectors ensured the licensee is properly trending and tracking the results of their periodic system tests for gas accumulation.

The inspectors verified that the selected CAP entries acceptably addressed the areas of concern associated with the scope of GL 2008-01, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal and Containment Spray Systems" (TI 2515/177, Section 04.01).

This review constituted one in-depth problem identification and resolution sample as defined in IP 71152-05. In addition, this inspection effort counts towards the completion of TI 2515/177 which will be closed in a later inspection report.

b. Findings

No findings were identified.

4OA3 Followup of Events and Notices of Enforcement Discretion (71153)

.1 (Closed) Licensee Event Report (LER) 05000374/2010-01-00: High Pressure Core Spray System Declared Inoperable Due to Failed Room Ventilation Control Relay

a. Inspection Scope

On September 25, 2010, the supply and exhaust fans for the Unit 2 Division 3 switchgear room VD were unexpectedly found tripped. Division 3 switchgear supports the HPCS system. Following this discovery, all Unit 2 Division 3 equipment was declared inoperable and unavailable. As HPCS is a single train system, this failure resulted in a complete loss of system function, requiring the licensee to make an eight hour notification to the NRC under 10 CFR 50.72(b)(3)(v)(D) and subsequent Licensee Event Report (LER) under 50.73(a)(2)(v)(D). The relay was replaced and tested satisfactorily. The cause of the relay failure was subsequently determined to be age-related degradation.

The inspectors reviewed the event described in LER 05000374/2010-01-00 for accuracy and potential violations. In addition, as part of the assessment, the inspectors evaluated the extent-of-condition review and the adequacy of the corrective actions performed by the licensee. Documents reviewed as part of this inspection are listed in the Attachment to this report. This LER is closed.

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

b. Findings

Introduction: A finding of very low safety significance (Green) and an associated NCV of TS 5.4.1, “Procedures”, was self-revealed, for the failure to follow the performance centered monitoring process specified in procedure MA-AA-716-210, “Performance Centered Monitoring Process." As a result, a control relay for the Unit 2 Division 3 ventilation fan was inappropriately classified for its preventive maintenance schedule, causing its failure on September 25, 2010, and the unexpected unavailability and inoperability of the Unit 2 HPCS System.

Description: On September 25, 2010, the supply and exhaust fans for the Unit 2 Division 3 switchgear room VD were unexpectedly found tripped. Division 3 switchgear supports the HPCS system. Following this discovery, all Unit 2 Division 3 equipment was declared inoperable and unavailable. As HPCS is a single train system, this failure resulted in a complete loss of system function, requiring the licensee to make an eight hour notification to the NRC under 10 CFR 50.72(b)(3)(v)(D) and subsequent LER under 50.73(a)(2)(v)(D). The relay was replaced and tested satisfactorily. The HPCS system was inoperable for less than 20 hours.

Subsequent troubleshooting identified that the cause of the Division 3 ventilation failure was the 480V motor control center control relay. This failed relay was removed and sent to the vendor for failure analysis. The vendor determined that the relay had been manufactured in 1985, and that it failed from age-related degradation. To determine the reason why the control relay had never been replaced, the licensee investigated the performance centered maintenance and time-based replacement classification of it. During the investigation, the licensee discovered that the relay was classified as a critical (safety/risk significant), low duty cycle, mild service component. This improper

classification resulted in a replacement recommendation of “as-required.” In accordance with MA-AA-716-210, “Performance Centered Maintenance Process,” and based on the 100 percent duty cycle of this component, this relay should have been classified as a critical, high duty cycle, mild service component. This new classification would result in a replacement frequency recommendation of 10 years.

The licensee determined the apparent cause of the control relay failure to be a lack of a time-based refurbishment/replacement program for high duty cycle (continuously energized) relays. This lack of a time-based replacement frequency was caused by the improper duty cycle classification. As a corrective action, the licensee re-classified the control relay to reflect actual plant conditions and ensure a proper time-based replacement schedule. In addition, an extent-of-condition review identified four other critical, high duty cycle relays in the VD system with the wrong replacement classifications. These were also re-classified to reflect actual plant conditions and ensure proper a time-based replacement frequency.

Analysis: The inspectors concluded that the failure to properly classify the Unit 2 Division 3 ventilation fan control relay in accordance with MA-AA-716-210, “Performance Centered Maintenance Process”, constituted a performance deficiency that warranted evaluation using the SDP. Using IMC 0612, Appendix B, “Issue Screening,” the inspectors determined that the finding was of more than minor significance because it affected the Mitigating Systems Cornerstone attribute of Human Performance (human error pre-event), and it affected the cornerstone objective to ensure availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. To further assess the significance of the finding, the inspectors used IMC 0609, Appendix A, “Determining the Significance of Reactor Inspection Findings for At-Power Situations,” and determined that Mitigating Systems was the only cornerstone affected. Using the Mitigating Systems column on the Phase 1 SDP characterization worksheet, the inspectors determined that the finding constituted a loss of safety function because HPCS system is a single train and it was declared inoperable. As a result, the inspectors transitioned to SDP Phase 2. Using the LaSalle-specific pre-solved table, and using an exposure time of less than 3 days, since HPCS was inoperable for less than 20 hours, the review indicated a finding of low to moderate safety significance or White.

Because of inherent conservatisms assumed in the Phase 2 analyses, the inspectors contacted the Region III senior reactor analyst for LaSalle, who performed further risk analyses via a Phase 3 risk assessment. The senior reactor analyst conducted an SDP Phase 3 analysis using SAPHIRE 8 Version 8.0.7.13 and the LaSalle SPAR Model Version 8.15. A change set was created representing a failure of the HPCS room ventilation. The exposure time was conservatively assumed to be 24-hours.
The dominant scenario involved a loss of vital DC bus A and failures of main feedwater, HPCS, reactor core isolation cooling, and reactor depressurization. The result was a delta core damage frequency (CDF) of 5.9E-8. Considering the results of the analysis, the senior reactor analyst concluded that the risk significance of the finding was best characterized as having very low safety significance (Green). The inspectors did not identify a cross-cutting aspect associated with this finding.

Enforcement: Technical Specifications 5.4.1, “Procedures”, requires that written procedures shall be established, implemented, and maintained as recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33,

Appendix A, Section 9, "Procedures for Performing Maintenance," specifically addresses the need to have appropriate procedures for preventive maintenance that can affect the performance of safety-related equipment. The licensee developed procedure MA-AA-716-210, "Performance Centered Maintenance Process" to implement that requirement. Contrary to the above, the licensee failed to follow the above procedure and improperly classified the control relay for Unit 2 Division 3 ventilation fan. As a result, on September 25, 2010, this control relay failed and the associated Division 3 ventilation tripped. This caused the unexpected unavailability and inoperability of the HPCS system and a loss of safety function for less than 20 hours. Because this finding was determined to be of very low safety significance and has been entered into the licensee's CAP (IR 1117744), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. The licensee's corrective actions included the re-classification of the control relay to critical, high duty cycle, to help ensure that replacement of the component occurs at the appropriate time-based frequency. (NCV 05000373/2010005-02; 05000374/2010005-02)

4OA5 Other Activities

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

a. Inspection Scope

(1) Control of Heavy Loads

The inspectors initiated a review of the licensee's crane and heavy loads program with regards to ISFSI operations in 2009 as previously documented in NRC Inspection Report 05000373/2009004; 05000374/2009004.

As part of the modifications in preparations to ISFSI operations, the licensee upgraded the 125 ton capacity overhead crane in the Reactor Building to a single failure proof crane. The inspectors completed their review of documentation associated with the Reactor Building crane. The review included structural evaluations associated with the seismic design of the new trolley, hoist/reeving equipment, miscellaneous components, crane bridge girders, supporting structural steel, modifications affecting the operating plant, floor loading in the SFP and other floor loading cask placement areas.
The inspectors also reviewed seismic restraints used during placement of the HI-TRAC on top of the storage cask (HI-STORM) during multi-purpose canister (MPC) transfer operations. The associated safety evaluations and screenings were also reviewed.

(2) Dry Run Activities

During this inspection period, the licensee performed preoperational dry run activities in order to fulfill the requirements of the Certificate of Compliance (CoC). The NRC inspectors were onsite to observe dry run activities July 19 through July 23, 2010, and September 21 through 24, 2010. These activities included MPC processing, heavy loads operations inside and outside of the reactor building, review of the licensee's 10 CFR 72.212 Report, crane walkdown inspection, and document review.

The inspectors observed the licensee place the HI-TRAC containing the MPC in the SFP. The inspectors observed the loading and unloading of dummy fuel bundles into the MPC basket. The licensee demonstrated removal of a dummy fuel assembly from

the SFP storage rack, placement of the assembly into the MPC, and retrieval of the fuel assembly from the MPC to the SFP rack. The inspectors observed the licensee remove a HI-TRAC containing a MPC from the SFP and subsequent placement of the HI-TRAC in the washdown pit.

The inspectors observed the licensee perform MPC processing activities. The licensee demonstrated MPC hydrostatic testing, blow-down, vacuum drying, and helium backfilling. The inspectors observed the licensee demonstrate MPC unloading dry run activities.

The inspectors observed transfer of the MPC from the HI-TRAC cask to the HI-STORM in a restrained support structure in the reactor building and the subsequent movement of the HI-STORM outside of the reactor building on a low profile transporter. The inspectors verified adequate communication and team work between departments and adherence to procedures.

The inspectors observed transfer of the HI-STORM overpack from the reactor building to the ISFSI pad via the haul path and placement on its proper location on the ISFSI pad using the vertical cask transporter.

The inspectors reviewed loading and unloading procedures to ensure that they contained commitments and requirements specified in the license, TS, UFSAR and 10 CFR Part 72.

(3) Fuel Selection

The inspectors reviewed the licensee's program associated with fuel characterization and selection for storage. The inspectors reviewed cask fuel selection packages to verify that the licensee was loading fuel in accordance with the TS. The licensee did not plan to load any damaged fuel assemblies during this initial campaign.

(4) Radiation Protection

The inspectors evaluated the licensee's Radiation Protection (RP) Program pertaining to the operation of the ISFSI. The inspectors reviewed the licensee's procedures describing the methods and techniques used when performing dose rate and surface contamination surveys and verified that they ensured dose rate limits and surveillance requirements of the TS were met. The inspectors verified that the licensee's RP staff considered lessons learned from other utilities' spent fuel loading campaigns during development of the radiological controls for the LaSalle County Station loading operations. The inspectors interviewed licensee personnel to verify their knowledge regarding the scope of the work and the radiological hazards associated with transfer and storage of spent fuel. The inspectors reviewed licensee dose rate calculations to verify that the licensee's ISFSI was in compliance with 10 CFR 72.104, "Criteria for Radioactive Materials in Effluents and Direct Radiation from an ISFSI or MRS [Monitored Retrievable Storage Installation]."

(5) Training

The inspectors reviewed the licensee's ISFSI Training Program, which consisted of classroom and on-the-job training to ensure involved staff was adequately trained for the job they were responsible to perform. The inspectors also reviewed training records and

qualifications of individuals performing work activities associated with the ISFSI. The inspectors interviewed licensee personnel to verify that they were knowledgeable in the scope of work that was being performed.

(6) Quality Assurance

The inspectors reviewed the licensee's QA program, as it applied to the ISFSI. LaSalle County Station has incorporated the ISFSI QA program into their established 10 CFR Part 50 QA program as allowed by 10 CFR 72.140(d). The inspectors reviewed procedures pertaining to the receipt inspection of MPCs. The inspectors observed that gauges were within their calibration date and that 99.995 percent pure helium was used during backfilling.

(7) Emergency Preparedness and Fire Protection

The inspectors reviewed the licensee's Emergency Preparedness Plan required by 10 CFR 50.47 for conformance with 10 CFR 72.32(c). The inspectors verified that the licensee incorporated Emergency Action Levels into the Emergency Plan to address the possible emergency scenarios, their classification, and recovery actions associated with the ISFSI.

b. Findings

(1) Failure to Perform Adequate Evaluations for Reactor Building Crane Upgrade

Introduction

The inspectors identified a finding of very low safety significance with an associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to perform adequate evaluations to upgrade their single failure proof crane. Specifically, for evaluations of the Reactor Building crane and crane support structure, the licensee failed to comply with ASME NOG-1-2004, "Rules for Construction of Overhead on Gantry Cranes (Top Running and Bridge, Multiple Girder)." The licensee used compliance with ASME NOG-1-2004 as the basis for their upgrade to single failure proof. The ASME NOG-1-2004 was endorsed by the NRC per Regulatory Issue Summary 2005-25, Supplement 1, "Clarification of NRC guidelines for Control of Heavy Loads," as an acceptable method for satisfying the guidelines of NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants," for single failure proof cranes. This commitment was reflected in the licensee's Engineering Change as well as their MOD 50.59 Screening and subsequent incorporation into the UFSAR. The licensee documented the conditions in IR 957014, IR 1093028, and IR 1098435 and initiated actions for calculation revisions and field modifications.

Description

During review of calculations for the crane and crane support structure, the inspectors identified five examples where the licensee failed to meet the requirements in 10 CFR Part 50 Appendix B, Criterion III, "Design Control."

1. Calculation L-003415, Revision 00B (8/12/09), Reactor Building Crane Supporting Structure Analysis: The crane and support structure design was based on an assumption that sliding would occur at the crane rail/wheel interface thus limiting the

applied loads to frictional forces. This assumption resulted in significantly reduced seismic loads and was inconsistent with the boundary condition requirements stipulated in Section 4153.6 of ASME NOG-1-2004. Additional discrepancies were also identified between the boundary conditions used in the design and the ASME NOG-1-2004 requirements. These discrepancies resulted in revisions to a number of calculations associated with the crane upgrade. The licensee documented the discrepancies in IR 00957014.

2. Calculation L-003411, Revision 2 (7/9/10), Exelon/LaSalle Single Failure Proof Bridge Stress Analysis Report: The inspectors identified multiple errors/discrepancies in the evaluation for the horizontal and vertical seismic restraints. The errors identified for the vertical restraints are noted below. Similar errors were also identified in the calculation for the horizontal restraint. The calculation used bolt allowable stresses from the 13th Edition of the American Institute of Steel Construction Specification instead the 9th Edition. The ASME NOG-1-2004 requirements are based on the 9th Edition. The 9th Edition specifies lower allowable stresses. Errors were identified in the calculation for the bolt group section properties due to the use of incorrect dimensions. For determination of bolt stresses, the calculation addressed the effect of the moment caused by the applied vertical load, but failed to account for the vertical load itself. Based on the above errors, the calculated bolt stress was 11.7 kilopound per square inch, while the revised calculation indicated the stress to be 58.6 kilopound per square inch. This discrepancy was identified during a revision in response to questions posed by the NRC inspectors. The licensee documented the discrepancy in IR 1093028.

3. Calculation L-003411, Revision 2 (7/9/10), Exelon/LaSalle Single Failure Proof Bridge Stress Analysis Report: The inspectors identified that in the crane girder evaluation for loads from the seismic restraint, the effect of the safe shutdown earthquake (SSE) load was addressed; however, the operating basis earthquake (OBE) load case was not addressed and no justification was provided to show that the OBE load case would not govern. Since the allowable stresses for the OBE are smaller than for the SSE, it is possible that the OBE case could be more limiting. Upon identification of the above concerns, the licensee performed more refined analyses and revised the calculation to address the OBE load. The licensee's trolley analysis did not address the "no load on hook" condition and the loaded "hook down" position. The licensee documented the discrepancy in IR 1093028.

4. Calculation L-003400, Revision 0 (9/11/09), Decon Pit Grillage for Cask Loading – Reactor Building El. 843'-6": The inspectors identified that the evaluation of the grillage supporting the HI-TRAC was based on a 33 percent increase in the OBE load case allowable stresses. The load combinations specified in the UFSAR do not allow any increase for the OBE load case. The calculation showed that the OBE load case governed the design and that allowable stresses would be exceeded if no increase was allowed. The licensee documented the discrepancy in IR 1098435.

5. Calculation L-003400, Revision 0 (9/11/09), Decon Pit Grillage for Cask Loading – Reactor Building El. 843'-6": The Inspectors identified that in the evaluation of concrete beams 809 and 810, all critical locations for shear stresses were not addressed. The shear was checked only near the end of the beams where the stirrups are spaced at 3" or 6". The inspectors noted that sections away from the

end could be more critical where the stirrup spacing increased to 12". The licensee documented the discrepancy in IR 1098435.

The crane was not operational as an upgraded single failure proof crane during this period. Resolution of the above items resulted in the licensee performing a number of new calculations and issuing major revisions to the existing calculations demonstrating adequacy of the design after installation of the modifications. The crane was converted to single failure proof following additional calculations and modifications.

Analysis

The inspectors determined that the licensee's failure to perform adequate evaluations to upgrade their single failure proof crane was contrary to the design control measures per 10 CFR Part 50, Appendix B, Criterion III requirements and was a performance deficiency. The inspectors reviewed the examples of minor issues in IMC 0612, "Power Reactor Inspection Reports," Appendix E, "Examples of Minor Issues," and found no examples related to this issue. Consistent with the guidance in IMC 0612, Appendix B, "Issue Screening," the finding was determined to be of more than minor significance because it was associated with the Initiating Events Cornerstone attribute of Equipment Performance and affected the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the failure to perform adequate evaluations of the reactor building crane and crane support structure affected the licensee's ability to provide reasonable assurance that loads would not be dropped during critical lifts.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," Tables 3b and 4a for the Initiating Events Cornerstone. The finding affects the Initiating Events Cornerstone because a reactor building crane heavy load drop could upset plant stability and challenge critical safety functions. Since the finding was a design qualification deficiency confirmed not to result in a heavy load drop, it was screened as a finding of very low safety significance (Green).

Cross-Cutting Aspect

The inspectors identified a Human Performance, Work Practices (H.4.c) cross-cutting aspect associated with this finding. The licensee did not ensure supervisory and management oversight of work activities, including contractors, such that nuclear safety was supported. Specifically, the licensee failed to have adequate oversight of design calculations and documentation for establishing structural adequacy of the crane components and the crane support structure for the crane upgrade to single failure proof. (IMC 0310 H.4(c))

Enforcement

Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control" states, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis for those SSCs to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions. Contrary to the above:

1. Calculation L-003415, Revision 00B (8/12/09), Reactor Building Crane Supporting Structure Analysis: The crane and support structure design was based on an assumption that sliding would occur at the crane rail/wheel interface thus limiting the applied loads to frictional forces. This assumption resulted in significantly reduced seismic loads and was inconsistent with the boundary condition requirements stipulated in Section 4153.6 of ASME NOG-1-2004. Additional discrepancies were also identified between the boundary conditions used in the design and the ASME NOG-1-2004 requirements.

2. Calculation L-003411, Revision 2 (7/9/10), Exelon/LaSalle Single Failure Proof Bridge Stress Analysis Report: The inspectors identified multiple errors/discrepancies in the evaluation for the horizontal and vertical seismic restraints.

3. Calculation L-003411, Revision 2 (7/9/10), Exelon/LaSalle Single Failure Proof Bridge Stress Analysis Report: The inspectors identified that in the crane girder evaluation for loads from the seismic restraint, the effect of the SSE load was addressed but the OBE load case was not addressed and no justification was provided to show that the OBE load case would not govern. The licensee trolley analysis did not address the “no load on hook” condition and the loaded “hook down” position.

4. Calculation L-003400, Revision 0 (9/11/09), Decon Pit Grillage for Cask Loading – Reactor Building Elevation 843’ 6”: The inspectors identified that the evaluation of the grillage supporting the HI-TRAC was based on a 33 percent increase in the OBE load case allowable stresses. The load combinations specified in the UFSAR do not allow any increase for the OBE load case. The calculation showed that the OBE load case governed the design and that allowable stresses would be exceeded if no increase was allowed.

5. Calculation L-003400, Revision 0 (9/11/09), Decon Pit Grillage for Cask Loading – Reactor Building Elevation 843’6”: The inspectors identified that in the evaluation of concrete beams 809 and 810 all critical locations for shear stresses were not addressed.

This violation is being treated as an NCV consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 05000373/2010005-03; 05000374/2010005-03; 07200070/2010-01, Failure to Perform Adequate Evaluation for Reactor Building Crane Upgrade). The licensee documented this violation in their CAP under IR Nos. 957014, 1093028, and 1098435, and initiated actions for calculation revisions and field modifications.

(2) Review of 10 CFR 72.212(b) Evaluations at Operating Plants

a. Inspection Scope

(1) Title 10 CFR 72.212 Report

The inspectors evaluated the licensee’s compliance with the requirements of 10 CFR 72.212 and 10 CFR 72.48. The inspection consisted of interviews with cognizant personnel and a review of documentation. The licensee is required, as

specified in 10 CFR 72.212(b)(1)(i), to notify the NRC of the intent to store spent fuel at the LaSalle ISFSI facility at least 90 days prior to the first storage of spent fuel. The licensee notified the NRC on February 9, 2010, of their intent to store spent fuel using the Holtec HI-STORM 100 Cask System according to CoC No. 72-1014, Amendment 3.

A written evaluation is required per 10 CFR 72.212(b)(2)(i), prior to use, to establish that the conditions of the CoC have been met. "LaSalle County Station Units 1 and 2 10 CFR 72.212 Evaluation Report," Revision 0, dated June 8, 2010, documented the evaluations performed by the licensee prior to use of the 10 CFR Part 72 general license.

The inspectors reviewed and assessed the licensee's 10 CFR 72.212 Evaluation Report. The inspectors reviewed that applicable reactor site parameters, such as fire and explosions, tornadoes, wind-generated missile impacts, seismic qualifications, lightning, flooding and temperature, had been evaluated for acceptability with bounding values specified in the Holtec HI-STORM 100 UFSAR and associated analyses.

The inspectors reviewed several supporting documents referenced in the Evaluation Report, in particular, Calculation L-003353, "LaSalle County Station Independent Spent Fuel Storage Installation Fire Hazard Analysis, Revision 1." This report contained the results of the fire and explosion hazard analysis for the ISFSI haul path and storage location and prescribed physical and administrative controls required during cask movement on the haul path as well as for ISFSI operations.

(2) ISFSI Pad Design

The inspectors reviewed the licensee's ISFSI pad evaluations for compliance with the requirements in 10 CFR 72.212 (b)(2)(i)(B) during ISFSI inspections in 2009.

During the review of ISFSI pad calculations, the inspectors identified an issue of concern regarding the licensee's evaluation of the ISFSI pad. The licensee entered the issue into their CAP as IR 966506. URI 07200070/2008001-01, "ISFSI Pad Analysis Issues," was opened to track resolution of the issue.

The licensee revised their calculations as a result of inspector questioning associated with URI 07200070/2008001-01. Region III staff requested assistance, through a Technical Assistance Request, from the Division of Spent Fuel Storage and Transportation (DSFST) Office, to review the two revised analyses to determine if the licensee's evaluations met regulatory requirements.

b. Findings

(1) Failure to Design the ISFSI Pad to Adequately Support the Static and Dynamic Loads of Stored Casks

Introduction

The inspectors identified a Severity Level IV NCV of 10 CFR 72.212 (b)(2)(i)(B), "Conditions of a General License Issued Under 10 CFR 72.210." Specifically, the inspectors identified five examples where the licensee failed to perform written evaluations prior to use that establish that the cask storage pads and areas have been

designed to adequately support the static and dynamic loads of the stored casks, considering potential amplification of earthquakes. As an immediate corrective action and given the need for the licensee to load ISFSI casks and move them onto the pad, the licensee restricted the total load applied to the ISFSI pad by allowing a maximum of 10 casks. Additionally, they limited cask locations to every other cask location in each direction on the pad, so that for any cask on the pad an open (unused) location would be adjacent to it in both the length and width directions of the pad. Because this restriction on the number of casks and loading pattern significantly reduced the total load distribution on the pad, the licensee concluded that for this reduced loading the concrete pad can adequately support the static and dynamic loads.

Description

The ISFSI pad must be designed to adequately support the static and dynamic loads considering potential amplification of earthquakes through soil structure interaction (SSI), as required by 10 CFR 72.212. The inspectors identified five examples where the licensee failed to meet the requirements of 10 CFR 72.212 (b)(2)(i)(B).

1. Calculation L-003447, Revision 3 (8/17/2009), Dynamic Analysis of HI-STORM 100 Cask on LaSalle ISFSI Pads: In lieu of performing a detailed dynamic analysis to determine seismic response of the cask, the licensee used the methodology described in the NUREG/CR-6865, "Parametric Evaluation of Seismic Behavior of Free Standing Spent Fuel Dry Cask Storage System." The inspectors determined that the calculation contained a number of assumptions and did not demonstrate the LaSalle ISFSI pad was bounded by the analyzed pad in NUREG/CR-6865.
   The licensee revised their calculation and performed an SSI analysis to address the oversight. The inspectors reviewed the revised calculation. The licensee entered this issue into their CAP (IR 966506). This NRC-identified violation closes URI 07200070/2008001-01.

2. Calculation L-003447, Revision 3 (8/17/2009), Dynamic Analysis of HI-STORM 100 Cask on LaSalle ISFSI Pads: The inspectors observed that the dynamic analysis did not capture three-dimensional effects, such as torsion, due to a partially loaded pad. An asymmetrically loaded pad will have a torsional dynamic response, and it is anticipated that acceleration in the short direction will be lower for a fully loaded symmetric structure than for the partially loaded nonsymmetrical structure.
   The licensee failed to analyze the pad for the worst case cask configuration on the ISFSI pad and thus failed to adequately address increased torsional dynamic responses on the ISFSI pad. The licensee entered this issue into their CAP (IR 900610).

3. Calculation L-003447, Revision 4 (5/12/2010), Final Design Basis Dynamic Analysis of LaSalle ISFSI Pad: The inspectors observed in the design basis dynamic analysis of the LaSalle ISFSI pad the methodology used to develop the SSI model and ensuing SSI analyses used best estimate soil properties.

   American Society of Civil Engineers (ASCE) Standard 4-98, Section 3.3.1.7 states the following: "The uncertainties in the SSI analysis shall be considered. In lieu of a probabilistic evaluation of uncertainties, an acceptable method to account for uncertainties in SSI analysis is to vary the low strain soil shear modulus. Low strain soil shear modulus shall be varied between the best estimate value times (1+Cv) and

the best estimate value divided by (1+Cv), where Cv is a factor that accounts for uncertainty in the SSI analysis and soil properties. If sufficient, adequate soil investigation data are available, the mean and standard deviation of the low strain shear modulus shall be established for every soil layer. The Cv shall be established so that it will cover the mean plus or minus one standard deviation for every layer. The minimum value of Cv shall be 0.5. When insufficient data are available to address uncertainties in soil properties, Cv shall be taken as no less than 1.0".

The licensee used ASCE 4-98 as industry guidance for completion of the SSI. However, the licensee failed to address uncertainties in the soil in accordance with this standard. Discussions with DSFST staff determined that this omission was non-conservative. The omission reduced the licensee's calculated safety factor and should have been included in the licensee's analysis. The licensee entered this issue into their CAP (IR 1102633).

4. Calculation L-003447, Revision 4 (5/12/2010), Final Design Basis Dynamic Analysis of LaSalle ISFSI Pad: The inspectors observed in the licensee's SSI model the bedrock outcrop, (which is also the base of the SSI model) was modeled as a fixed mass and, therefore, was unable to move and transmit seismic waves. The earthquake control motions were, therefore, applied as an inertia force time history to each mass: cask center of gravity, pad center of gravity, and soil mass center of gravity. This methodology is non-physical. The inspectors recognize that this non-physical methodology may be theoretically correct for a linear analysis; however, the inspectors have no evidence that this methodology is applicable to a nonlinear problem wherein a cask is allowed to slide, tip or lose complete contact with the pad. The inspectors note that in every known SSI methodology that has been reviewed and approved by the NRC, the control motion is applied at a bedrock outcrop or comparable soil layer. This is physically how the earthquake ground motion arrives at the site. The seismic waves arrive at the bedrock outcrop, are filtered and amplified by the soil layers between the rock outcrop and the ground surface and generate motion to the ISFSI pad.

   The licensee did not provide adequate justification and documentation for use of a new SSI analysis methodology. The licensee entered this issue into their CAP (IR 1102633).

5. Calculation L-003447, Revision 4 (5/12/2010), Final Design Basis Dynamic Analysis of LaSalle ISFSI Pad: The inspectors observed in the licensee's analysis, a single set of three-dimensional (two horizontal and one vertical) acceleration time-histories was developed to envelop the 5 percent damped Regulatory Guide 1.60 response spectra to perform the nonlinear SSI analysis. The use of a single set of three-dimensional time-histories is not standard practice for performing a nonlinear SSI analysis. The ASCE 4-98, Section 3.2.2.3(d), "Nonlinear Analysis," states the following: "In general, more than one set of acceleration time-histories, meeting the requirements of Section 2.3, should be used, and the results of the analyses shall be averaged." NUREG/CR-6865 also discusses this same issue and states the following in Section 4.1: "...the seismic response of a dry cask using one time-history might not always lead to a predictable response. It is increasingly obvious that a suite of earthquake inputs should be examined in order to obtain statistically stable mean and standard variation in the response to form the basis for design decision. This would require multiple runs using several earthquake records." The NUREG

further provided evidence that the difference in maximum response among five sets of time histories varies by as much as a factor of six for the same spectral shape. This showed that the effect of the differences in frequency content and phasing within the five sets of time-histories has a significant influence on response. Due to the potentially large differences in response that can result from using different earthquake time-histories as input to a nonlinear SSI analysis, the inspectors determined that the licensee's use of only a single set of acceleration time-histories to perform a non linear SSI analysis may have significantly underestimated the predicted seismic response and thus does not conservatively meet the requirements of 10 CFR 72.212. The licensee entered this issue into their CAP (IR 1102633).

Analysis

The inspectors determined that the previously discussed examples were a violation that warranted a significance evaluation. Consistent with the guidance in Section 2.2 of the NRC Enforcement Policy, ISFSIs are not subject to the SDP and, thus, traditional enforcement will be used for these facilities. The inspectors determined that the violation was of more than minor significance because, if left uncorrected, a failure of the ISFSI pad could lead to a more significant safety concern. Consistent with the guidance in Section 2.6.D of the NRC Enforcement Manual, if a violation does not fit an example in the Enforcement Policy Violation Examples, it should be assigned a severity level:

(1) Commensurate with its safety significance; and (2) informed by similar violations addressed in the Violation Examples. The inspectors determined that the violation could be screened using Section 6.5.d.1 of the NRC Enforcement Policy as a Severity Level IV Violation.

Enforcement

Title 10 CFR 72.212 (b)(2)(i)(B) requires, in part, that the licensee perform written evaluations prior to use, that establish the cask storage pads and areas have been designed to adequately support the static and dynamic loads of the stored casks, considering potential amplification of earthquakes.

Contrary to the above, the licensee's completed evaluation did not adequately evaluate the cask storage pad to support static and dynamics loads of the stored casks considering potential amplification of earthquakes as demonstrated by the following examples:

1. Calculation L-003447, Revision 3 (8/17/2009), Dynamic Analysis of HI-STORM 100 Cask on LaSalle ISFSI pads: The inspectors identified that in lieu of performing a detailed dynamic analysis to determine seismic response of the cask, the licensee used the methodology described in the NUREG/CR-6865. The inspectors determined that the calculation contained a number of assumptions and did not demonstrate the LaSalle ISFSI pad was bounded by the analyzed pad in NUREG/CR-6865.

2. Calculation L-003447, Revision 3 (8/17/2009), Dynamic Analysis of HI-STORM 100 Cask on LaSalle ISFSI Pads: The inspectors identified that the dynamic analysis did not capture three-dimensional effects, such as torsion, due to a partially loaded pad. The licensee failed to analyze the pad for the worst case cask configuration on the

pad and thus failed to adequately address increased torsional dynamic responses on the pad.

3. Calculation L-003447, Revision 4 (5/12/2010), Final Design Basis Dynamic Analysis of LaSalle ISFSI Pad: The inspectors identified that the licensee used ASCE 4-98 as industry guidance for completion of the SSI. However, the licensee failed to address uncertainties in the soil in accordance with this standard. The omission reduced the licensee's calculated safety factor and should have been included in the licensee's analysis.

4. Calculation L-003447, Revision 4 (5/12/2010), Final Design Basis Dynamic Analysis of LaSalle ISFSI Pad: The inspectors identified that the licensee did not provide adequate justification and documentation for use of a new SSI analysis methodology.

5. Calculation L-003447, Revision 4 (5/12/2010), Final Design Basis Dynamic Analysis of LaSalle ISFSI Pad: The inspectors identified that the licensee's analysis used a single set of three-dimensional (two horizontal and one vertical) acceleration time-histories to complete the SSI analysis. The inspectors determined that the licensee's use of only a single set of acceleration time-histories to perform a nonlinear SSI analysis may have significantly underestimated the predicted seismic response and thus does not conservatively meet the requirements of 10 CFR 72.212.

This is a violation of 10 CFR 72.212 (b)(2)(i)(B), "Conditions of a General License Issued Under 72.210." This violation is being treated as an NCV consistent with Section 3.1.1 of the NRC Enforcement Manual. (NCV 05000373/2010005-04; 05000374/2010005-04; 07200070/2010-02, Failure to Design the ISFSI Pad to Adequately Support the Static and Dynamic Loads of Stored Casks). The licensee entered this violation into their CAP (IR 900610, IR 966506, and IR 1102633). This closes URI 07200070/2008001-01.

(2) Failure to Perform Adequate Evaluations to Ensure Compliance with 10 CFR 72.212(b)(3) and 10 CFR 72.122(b)(2)(i)

Introduction

The inspectors identified a Severity Level IV NCV of 10 CFR 72.146, "Design Control," for the licensee's failure to perform adequate evaluations to ensure compliance with 10 CFR 72.122(b)(2)(i) and 10 CFR 72.212(b)(3). Specifically, the inspectors identified that the licensee failed to evaluate that the reactor site parameters, including analyses of tornado missiles, were enveloped by the HI-TRAC design basis and that the HI-TRAC was designed to withstand the effects of natural phenomenon including tornadoes. The licensee documented the conditions in IR 1137279 and initiated actions to evaluate the described condition.

Description

Title 10 CFR 72.122(b)(2)(i), "Overall Requirements," states, in part, that "structures, systems, and components important to safety must be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, lightning, hurricanes, floods, tsunami, and seiches, without impairing their capability to perform their intended design functions."

Title 10 CFR 72.212(b)(3), "Conditions of General License Issued Under 72.210," states that the licensee shall "review the Safety Analysis Report (SAR) referenced in the CoC and the related NRC Safety-Evaluation Report, prior to use of the general license, to determine whether or not the reactor site parameters, including analyses of earthquake intensity and tornado missiles, are enveloped by the cask design bases considered in these reports. The results of this review must be documented in the evaluation made in Paragraph (b)(2) of this section."

The Holtec UFSAR Section 3.4.8.2, "HI-TRAC Transfer Cask," Subsection 3.4.8.2.1, "Intermediate Missile Strike" states, in part, that the "HI-TRAC is always held by the handling system while in a vertical orientation completely outside of the fuel handling building. Therefore, considerations of instability due to a tornado missile strike are not applicable." The Holtec UFSAR did not evaluate the effects of a HI-TRAC tornado missile strike for overturning or sliding as it was determined by the CoC holder to not be a credible event.

However, at the LaSalle County Station, spent fuel storage processing operations are completed on the highest elevation floor of the reactor building, the refuel floor. While on the refuel floor, the HI-TRAC is not engaged to a handling system during processing operations. The reactor building siding and roofing on the refuel floor are designed to blow-in/blow-out or blow off at a predetermined wind pressure during a tornado event to protect the structural integrity of the structural steel, leaving an open pathway to the environment. Therefore, at LaSalle County Station, during a tornado event on the refuel floor, there is a potential that tornado generated missiles and winds could impact SSCs, specifically the HI-TRAC.

During review of Calculation L-003400, "Decontamination Pit Grillage for Cask Loading – Reactor Building EL843," Revision 1, and review of Calculation L-003498, "Tornado Evaluations for Byron, Braidwood, and LaSalle Station Dry Storage Projects," Revision 0, the inspectors noted that the HI-STORM had been evaluated for the effects of a tornado while stored on the pad; however, the effects of a tornado were not addressed for the HI-TRAC while being processed on the refuel floor. The inspectors noted that the HI-TRAC was not analyzed for cask overturning or sliding due to a tornado generated missile strike or tornado wind pressure on the refuel floor.

The inspectors determined that the licensee failed to determine that the reactor site parameters, including analyses of effects of natural phenomenon including tornadoes, were enveloped by the cask design bases and subsequently failed to perform an additional analysis to ensure that the requirements of 10 CFR 72.122(b)(3) were met. Subsequent to the inspectors inquiry the licensee performed Calculation L-003582, "Tornado Analysis for LaSalle HI-TRAC," Revision 0. Calculation L-003582 determined that overturning or sliding of the HI-TRAC at the refuel floor elevation would not occur due to the effects of a tornado. The inspectors reviewed the subsequent calculation.

Analysis

The inspectors determined that the licensee's failure to perform a calculation evaluating the effects of a tornado on the HI-TRAC was a violation that warranted a significance evaluation. Consistent with the guidance in Section 2.2 of the NRC Enforcement Manual, ISFSIs are not subject to the SDP and, thus, traditional enforcement is used for these facilities. The violation was determined to be of more than minor significance using IMC 0612, "Power Reactor Inspection Reports," Appendix E, "Examples of Minor

Issues," Example 3i, in that the licensee's lack of evaluation did not assure cask integrity during a design basis tornado and an additional calculation was required to evaluate the effects of the design basis tornado during canister processing operations in the reactor building refuel floor elevation in accordance with the ISFSI licensing/design basis analysis requirements.

Consistent with the guidance in Section 2.6.D of the NRC Enforcement Manual, if a violation does not fit an example in the Enforcement Policy Violation Examples, it should be assigned a severity level: (1) Commensurate with its safety significance; and (2) Informed by similar violations addressed in the violation examples. The violation screened as having very low safety significance (Severity Level IV). Specifically, Calculation L-003582 determined that overturning and sliding of the HI-TRAC at the refuel floor elevation would not occur during tornado missile impacts.

Enforcement

Title 10 CFR 72.146(a), "Design Control," states, in part, that "The licensee shall establish measures to ensure that applicable regulatory requirements and the design basis, as specified in the license for those SSCs to which this section applies, are correctly translated into specifications, drawings, procedures, and instructions. These measures must include provisions to ensure that appropriate quality standards are specified and included in design documents and that deviations from standards are controlled."

Contrary to the above, on August 9, 2010, the licensee failed to establish measures to ensure that applicable regulatory requirements and the design basis were correctly translated into specifications, drawings, procedures, and instructions. Specifically, the licensee failed to evaluate the effects of natural phenomenon, including tornadoes, on the HI-TRAC. This finding is being treated as an NCV, consistent with Section 3.1.1 of the NRC Enforcement Manual. (NCV 05000373/2010005-05; 05000374/2010005-05; 07200070/2010-03, Failure to Perform Adequate Evaluations to Ensure Compliance with 10 CFR 72.212(b)(3) and 10 CFR 72.122(b)(2)(i)). The licensee documented the violation in IR 1137279 and initiated actions to evaluate the described condition.

4OA6 Management Meetings

.1 Exit Meeting Summary

On January 13, 2011, the inspectors presented the inspection results to Mr. Dave Rhoades and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

.2 Interim Exit Meetings

Interim exits were conducted for:

- The results of the ISFSI dry run readiness inspections were presented on November 9, 2010, to members of the licensee management and staff. The licensee acknowledged the information presented.

- The upgrade of the reactor building load handling system to single-failure-proof equivalence for reactor vessel head lifts inspection with the Site Vice President, Mr. D. Rhoades, on December 3, 2010.
- The licensed operator requalification training annual operating test results with the Operator Training Manager, Mr. L. Blunk, via telephone, on December 7, 2010.
- The annual review of Emergency Action Level and Emergency Plan changes with the licensee's Emergency Preparedness Specialist, J. Hughes, via telephone on December 15, 2010.

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

4OA7 Licensee-Identified Violations

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

License Condition C.25, Fire Protection Program, requires that the licensee shall implement and maintain all provisions of the approved Fire Protection Program as described in the UFSAR for LaSalle County Station as approved in NUREG-0519 “Safety Evaluation Report related to the operation of LaSalle County Station, Unit 1 and 2”. Contrary to the above, on October 12, 2010, foreign material exclusion (FME) was found in the fire suppression header in the Division I shared cable spreading area. The finding was determined to be of very low safety significance because it was assigned a low degradation rating. Specifically, less than 10 percent of the nozzle heads in the system were impacted and there were functional nozzle heads within 10 feet of the non-functional ones. The licensee entered this issue into their CAP as IR 1120517, flushed and returned the system to service satisfactorily and revised the procedure to provide better testing of the fire suppression system in the future.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
D. Rhoades, Site Vice President
P. Karaba, Plant Manager
K. Aleshire, Exelon EP Programs Manager
D. Amezaga, GL 89-13 Program Owner
D. Anthony, Exelon NDE Outage Manager West
J. Bashor, Site Engineering Director
L. Blunk, Operations Training Manager
J. Gumnick, Senior ISFSI Project Manager
H. Do, Corporate Senior ISI Staff Engineer
P. Endress, Design Engineer
M. Entwistle, Operation Training
J.C. Feeney, NOS Lead Assessor
J. Miller, System Engineering Senior Manager
D. Schmit, Engineer Supervisor Mechanical/Structural
J. Houston, Regulatory Assurance
J. Hughes, EP Coordinator
K. Ihnen, Nuclear Oversight Manager
A. Kochis, ISI Engineer
J. Kutches, Manager of Projects
K. Hedgspeth, RP Manager
B. Maze, ISFSI Project Manager
J. Meyer, Maintenance Planner QV Inspector
J. Miller, Senior NDE Specialist
J. Paczolt, Operation Training
B. Rash, Maintenance Director
W. Hilton, Design Engineering Senior Manager
K. Rusley, EP Manager
J. Shields, ISI Program Manager
S. Shields, Regulatory Assurance
T. Simpkin, Regulatory Assurance Manager
K. Taber, Operations Director
W. Trafton, Shift Operations Superintendent
J. Vergara, Regulatory Assurance
G. Vickers, RP Technical Support Manager
H. Vinyard, Work Management Director
J. Washko, Outage Manager
J. White, Site Training Director
G. Wilhelmsen, Design Rapid Response Manager
K. Lyons, Chemistry Manager
M. Martin, Supervisor, Chemistry Programs
C. Wilson, Station Security Manager

Nuclear Regulatory Commission
K. Riemer, Chief, Reactor Projects Branch 2
B. Dickson, Branch Chief, Plant Support Team, DRS/RIII

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000374/2010-01-00	LER	High Pressure Core Spray System Declared Inoperable Due to Failed Room Ventilation Control Relay
05000374/2010005-02	NCV	Failure to Follow Performance Centered Monitoring Process Procedure
05000373/2010005-03 05000374/2010005-03 07200070/2010001-01	NCV	Failure to Perform Adequate Evaluation for Reactor Building Crane Upgrade (Section 4OA5)
05000373/2010005-04 05000374/2010005-04 07200070/2010001-02	NCV	Failure to Design the ISFSI Pad to Adequately Support the Static and Dynamic Loads of Stored Casks (Section 4OA5)
05000373/2010005-05 05000374/2010005-05 07200070/2010001-03	NCV	Failure to Perform Adequate Evaluations to Ensure Compliance with 10 CFR 72.212(b)(3) and 10 CFR 72.122(b)(2)(i) (Section 4OA5)
05000374/2010005-06	URI	Implementation of the Racklife computer model to monitor Unit 2 spent fuel pool storage racks degradation

Closed

05000374/2010-01-00	LER	High Pressure Core Spray System Declared Inoperable Due to Failed Room Ventilation Control Relay
05000374/2010005-02	NCV	Failure to Follow Performance Centered Monitoring Process Procedure
05000373/2010005-03 05000374/2010005-03 07200070/2010001-01	NCV	Failure to Perform Adequate Evaluation for Reactor Building Crane Upgrade (Section 4OA5)
05000373/2010005-04 05000374/2010005-04 07200070/2010001-02	NCV	Failure to Design the ISFSI Pad to Adequately Support the Static and Dynamic Loads of Stored Casks (Section 4OA5)
05000373/2010005-05 05000374/2010005-05 07200070/2010001-03	NCV	Failure to Perform Adequate Evaluations to Ensure Compliance with 10 CFR 72.212(b)(3) and 10 CFR 72.122(b)(2)(i) (Section 4OA5)
07200070/2008001-01	URI	ISFSI Pad Analysis Issues(Section 4OA5)

Discussed

None.

LIST OF DOCUMENTS REVIEWED

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

1R04 Equipment Alignment

Miscellaneous:
- LOP-DG-01E; Unit 1 A Diesel Generator Electrical Checklist; Rev. 7
- LOP-DG-01M; Unit 1 A Diesel Generator Mechanical Checklist; Rev. 9

1R05 Fire Protection

Procedures:
- EC 381673; Restore the Functionality of the Unit 0 Over Lab Pre-action FP System; Rev. 0

Issue Reports:
- 1125869; FME Recovery Results for the Rag found in the FP Piping; 10/13/2010
- 1125332; FME- Rag Found in FP Chem Lab Sprinkler Piping; 10/12/2010
- 1122483; Over-Labs Preaction Spray System Needs Flushing; 10/5/2010
- 1120517; Preaction Sprinkler System Fntcltest Complete w/ Comments; 10/1/2010

Drawings:
- 33; Drawing: Cable Trays over Lab Ceiling Unit 1 & 2; 7/28/1980

Miscellaneous:
- EACE 1120517-02; Equipment Apparent Cause Evaluation Report: Clogging of the Unit 0 Over Lab Pre-action Spray System; 10/1/2010
- LSCS-FPR H 3.4.16; Auxiliary Building Ground Floor – Fire Zone 4F3; Rev. 4
- LSCS-FPR H 4.2.52; Fire Zone 4F3 Auxiliary Building; Rev. 4
- EACE 1120517-02; Clogging of the Unit 0 Over Lab Pre-action Spray System (Draft); 10/1/2010
- Op Log 10/14/2010; Noon Shift LaSalle Operations Log
- Op Log 10/1/2010; LaSalle Operator Log

1R06 Flooding

Issue Reports:
- 1126512; Inspection of Cables in Underground Vaults; 10/14/2010

1R11 Licensed Operator Requalification Program

Miscellaneous:
- Results; Licensed Operator Annual Operating Test; 2010

1R12 Maintenance Effectiveness

Procedures:
- ER-AA-310-1005; Action Plan Development for CM-01 Emergency operation of the post LOCA Accident Primary Containment Atmosphere Hydrogen and Oxygen Monitor; 8/10/2010
- LOA-FLD-001; Flooding; Rev. 12

Issue Reports:
- 977872; Complete Maintenance Rule CDE Complete CDE; 12/9/2009
- 1001472; MR A(1) Determination Required for MS-01 Function
- 977872; U-2 Technical Specification Required Shutdown; 10/12/2009
- 1001472; Complete a(1) Determination; 1/7/2010
- 1096293; Maintenance Rule FASA Deficiency – Product Timeliness; 7/30/2010
- 1096292; Maintenance Rule FASA Deficiency – Failure Reviews; 7/30/2010
- 1018275; Maintenance Rule Assignments; 1/19/2010
- 430997; Discrepancies between U1 and U2 Armco gates; 12/6/2005
- 1147166; The U-2 CW Discharge Gate Tripped its Breaker; 12/1/2010

Miscellaneous:
- Periodic Assessment of the Maintenance Rule Program; 7/2008-6/2010
- PMRQ 61688-01; Inspect, Clean, Fill Oiler as Required; latest date: 8/7/2009

1R13 Maintenance Risk Assessments and Emergent Work Control

Procedures:
- HU-AA-1211; HLA Briefing Worksheet re Control Room Response to an EHC Failure and Subsequent Reactor Scram; undated
- LOA-TORN-001; High Winds / Tornado; Rev. 11

Issue Reports:
- 1130868; NRC Identified Concerns with Outside Storage; 10/26/2010
- 1130838; Safety – Wind has Partially Ripped Door Off of Ops Cabinet; 10/26/2010
- 1130682; Enter LOA-TORN-001 with LaSalle Station under Tornado Watch; 10/26/2010
- 1130186; High Wind Watch Issued for LaSalle County Station Area; 10/25/2010

Miscellaneous:
- Protected Equipment Log – 2A Diesel Generator Protected Pathways; 10/24/2010

1R15 Operability Evaluations

Procedures:
- LOS-LP-Q1; LPCS System Inservice Test; Rev. 52
- LOR-2H13-P601-C308; LPCS Pump Discharge Press Low; Rev. 3

Issue Reports:
- 1100104; Evaluate 2A RR HPU Reservoir Level Trend; 8/10/2010
- 1129847; Seismic Mounting of the SBLC Test Tank – CDBI Question; 10/22/2010
- 1129757; CDBI: SBLC Solution Tank Scaffold; 10/22/2010
- 1129956; Insufficient Detail in 50.50 Summaries in NRC Updates; 10/23/2010
- 1130414; PMID Inadvertently Retired for DG Storage Tank RM Sump; 10/25/2010
- 1131668; Design Analysis 030015(EMD) Re: SBLC Test Tank; 10/27/2010
- 1132019; Update Re: Design Analysis 03001(EMD) & SBLC Test Tank; 10/28/2010

- 1147265; U-2 CW Disch VLV did not Move when Close P/B was Depressed; 12/2/2010
- 1145471; U2 LPCS/A RHR Water Leg Pump 2E21-C002 Low Disch Press; 11/28/2010
- 1131746; U2 LPCS/A RHR Water leg Pump 2E21-C002 Low Disch Press; 10/28/2010
- 1074087; Div 1 Water Leg Pump Degradation; 5/27/2010

Work Orders:
- WO 1372971-01; LOS-LP-Q1 U2 LPCS System Att 2A; 12/24/2010

Calculations:
- A.38; LaSalle HRA Notebook: Operator Fails to Isolate TB CW/SW from Lake;

Miscellaneous:
- OE 10-004; Operability Evaluation of Standby Liquid Control (SBLC) Test Tank (IRs 1131668, 1132019, 1129847); Rev. 0
- OE 05-008; Operability Evaluation of Reactor Recirculation 1B RR Flow Control Valve (CR # 399198); Rev. 0
- EMD 030015; Human Performance Issue with Design Analysis – Adequacy of SBLC test tank 1(2)C412-A002 in the event of seismic event; undated
- AR 952624-02, AR 953484-02; Attachment 7, PRA Operator Actions Familiarization Guide;
- Shift Logs for 10/27/2010 0:05 to 10/28/2010 6:07
- Event 458-090114-1; River Bend Unit 1 Standby Liquid Control System Inoperable Greater than Allowable Outage Time (LER 458-09001); 3/13/2009
- LER 50-458 / 09-001-00; River Bend Station – Unit 1 Standby Liquid Control System Inoperable Greater than Allowable Outage Time; 1/14/2009
- LaSalle Plant Conditions for Units 1 and 2; 10/19/2010
- 3.38; LaSalle HRA Notebook: Operator Fails to Isolate Turbine Building CW/SW from Lake; 7/13/2007
- LaSalle Operations Log – LPCS; 12/20/2010 – 12/25/2010

<u>1R18 Plant Modifications</u>

Procedures:
- LS-MW-107-1001; Change Review for UFSAR Section 9.1.2.2.3; 11/21/2006
- LS-MW-107-1001; Change Review for UFSAR Section 9.1.2.1.3; 11/01/2005

Miscellaneous:
- Fuel Storage Reactivity Summary Sheet; LaSalle Unit 1 Cycles 13 and 14; 11/18/2009
- OE 07-006; Boraflex Panels Utilized in the Unit 2 Spent Fuel Pool Racks; Revision 4
- LS-AA-106-1001; Typical Plant Operations Review Committee Meeting Minutes Template; Revision 1
- LS-AA-106; Plant Operations Review Committee; Revision 6

<u>1R19 Post-Maintenance Testing</u>

Procedures:
- OP-LA-101-111-1002; Attachment 7, PRA Operator Actions Familiarization Guide; Rev. 32
- LOP-RT-01; Reactor Water Clean-Up System (RWCU) Filling, Venting and Pressurizing; Rev. 35
- LOP-RT-02; Reactor Water Clean-Up System (RWCU) – Startup and Pump Transfer; Rev. 36
- LOS-DG-M2; 1A(2A) Diesel Generator Operability Test; Rev. 82
- LOS-HP-Q1; HPCS System Inservice Test; Rev. 63

Issue Reports:
- 1142471; Chemistry Sample Results Unsat for 1B RT Pump; 11/18/2010
- 1130574; 2A DG Frequency Meter 2SI-DG028 Near Calibration Limits; 10/25/2010
- 1158150; Existing Switch does not Match Schematic or Wiring Drawing; 1/3/2011

Work Order:
- WO 1374522-01; LOS-DG-M2 2A Diesel Generator Att. 2A-Idle; 11/6/2010

Calculations:
- A.38; Operator Fails to Isolate TB CW/SW from Lake (LaSalle HRA Notebook)

Miscellaneous:
- RM LS-CRM-05; ARMCO Gate Availability without Installed Motor Driven Closure Capability; Rev. 0
- 3.38; LaSalle HRA Notebook, Operator Fails to Isolate Turbine Building DW/SW from Lake

1R20 Outage Activities

Issue Reports:
- 1137798: Heavy Lifts Not Performed in Accordance with MA-AA-716-022; 11/9/2010
- 1142927; Load Testing of Strongbacks; 11/19/2010
- 1147037; NRC ID: Potential Documentation Clean-Up Needed for Crane Load Pins; 12/1/2010
- 1147510; NRC ID: Wording Missing from UFSAR Change Package; 12/2/2010

Work Orders:
- 1039198-01; Disassemble and Reassemble Reactor Vessel; 2/ 8/2009
- 1137542-01; Annual Inspection of Reactor Building Overhead Crane; 7/17/2009
- 1171381-01; Reactor building Crane, Beam & Hoist Monthly Inspection; 10/ 23/2008
- 1174651-01; Reactor Crane Drum Inspection; 10/15/2008
- 1189273-01; Prior to Refuel Outage Inspection of Reactor Building Overhead Crane Per LEP-HC-101; 9/1/2009
- 1189981-01; Reactor building Crane, Beam & Hoist Monthly Inspection; December 22/2008
- 1270272; Annual Special Lifting devices Inspection; 9/21/2010
- 1276922-01; Quarterly Inspection of Reactor Building Overhead Crane Brakes; 12/30/2009
- 1297159-01; Monthly Inspection of Reactor Building Overhead Crane, Beam & Hoist; 1/22/2010

Engineering Changes:
- EC 371400; Reactor Building Crane – NEI 08-05 Single Failure Proof Equivalency Evaluation; Rev. 0
- EC 372504; Load Pin Sensor Weighing System for Reactor Head Strongback; Rev. 0
- EC 372504; Load Pin Sensor Weighing System for Reactor Head Strongback; Rev. 1

Drawings:
- 105D4776; Dryer and Separator Sling; Rev. 1
- 761E900; Shroud Head and Separators Outline; Rev. 7
- 762E537; Outline Steam Dryer; Rev. 2
- 767E743; Reactor Head Carousel; Rev. 1
- VPF 2029-117; Vessel Outline; Rev. 3
- VPF 3073-1; Vessel Outline; Rev. 7

Calculations:
- 030200(EMD); Qualification of Dryer & Separator Sling F19-E008; Rev. 1
- L-2714; CBI calculation for CRD Hatch Equipment, Equipment Hatch and Personnel Lock, Including Sumps; Rev. 0A
- L-3588; Evaluation of Drywell Head / RPV Insulation Strongback (EPN 0F19-E300); Rev. 0
- L-3592; U1 Reactor Pressure Vessel (RPV) Head Lugs; Rev. 0
- L-3593; U2 Reactor Pressure Vessel (RPV) Head Lugs; Rev. 0
- L-3594; RPV Head Strongback Carousel; Rev. 0
- L-3601; Evaluation of Dryer/Separator Lifting Lugs for NUREG-0612; Rev. 0
- LS-MISC-02; NEI 08-05 – Event Frequency Calculation; Rev. 0

Miscellaneous:
- 50.59 Screening No. L08-241; EC 372504, Rev. 0; Rev. 0
- C-7-4; Operations and Maintenance Manual for P&H Overhead Bridge Cranes; 7/2001
- Exelon Course MC2501; Overhead Crane Operator; Rev. 1
- HR-AA-07-105; Crane Operator Certification Exam; Rev. 0
- LUCR-109; LaSalle FSAR Change; 11/23/2010
- LUCR-227; LaSalle FSAR Change; 11/12/2010
- P&H document 36274-39; Exelon LaSalle Reactor Building Trolley – CN36274, Operation and Maintenance Manual; Rev. 0
- TQ-AA-174; Industrial Safety Training Program; Rev. 0

1R22 Surveillance Testing

Procedures:
- LOS-CS-Q1; Secondary Containment Damper Operability Test; Rev. 32
- LOS-RH-Q1; RHR (LPCI) and RHR Service Water Pump and Valve Inservice Test for Modes 1, 2, 3, 4 and 5; Rev. 75

Issue Reports:
- 1141205; Several Items of Trash Found in U1 B/C RHR PP Room; 11/16/2010

Work Orders:
- 1363727-01; LOS-RH-Q1 1C RHR System Operability Att. 1C; 11/12/2010

1EP4 Emergency Action Level and Emergency Plan Changes

Miscellaneous:
- Exelon Nuclear Radiological Emergency Plan Annex for LaSalle Station; Revs. 29, 30 and 31

4OA1 Performance Indicator Verification

Miscellaneous:
- LaSalle County Station MSPI Data, Heat Removal System (RCIC), 4th Quarter 2009 – 3rd Quarter 2010
- LaSalle County Station MSPI Data, Cooling Water Systems (CSCS), 4th Quarter 2009 – 3rd Quarter 2010
- LaSalle County Station MSPI Data, Safety Systems Functional Failures (SSFF), 4th Quarter 2009 – 3rd Quarter 2010

4OA2 Identification and Resolution of Problems

Procedures:
- LOP-LP-01; Filling and Venting, or Draining the Low Pressure Core Spray System; Rev. 24
- LOS-FC-Q1; Fuel Pool Emergency Makeup Pump Inservice Test and RHR Service Water System Flush; Rev. 28
- LOR-1H13-P601-A406; HPCS Header Pressure High; Rev. 6
- LOR-1H13-P601-A305; HPCS Pump 1E22-C001 Suction Pressure High/Low; Rev. 4
- LOP-HP-01; Filling and Venting the High Pressure Core Spray System; Rev. 26
- LOS-LP-M1; Low Pressure Core Spray System Operability Test; Rev. 15

Issue Reports:
- 816204; Air Pocket Detected in High Point of HPCS System Piping; 9/10/20081025015; Plant Engineering to determine UT inspection locations, details and initiate work requests; 3/31/2010
- 802499; LAS Actions for NRC GL 2008-01 Managing Gas Accumulation; 7/31/2008
- 798176; NRC GL 2008-01 Gas Intrusion Field Activities (1E22-F026); 7/18/2008
- 817966; NOS Id: RCIC, CSCS and Generic Letter 2008-01; 9/15/2008
- 1062732; NOS Id: Declining Trend in Seismic Monitor Performance; 4/28/2010
- 1110077; Vibration Alarms on 2B TDRFP; 9/5/2010
- 1070299; Outstanding Seismic Issues; 5/17/2010
- 1129280; NOS Id: Adverse Trend in Operating Plant Equipment; 10/21/2010
- 1101440; Jan 2010 through June 2010 trend Report Data – Roll-up IR; 8/13/2010
- 1104893; 1A and 1B RR Pump Seal Pressure Trends; 8/23/2010
- 1067656; Lost Control Power Indication for 1E12-F009 Inbd SDC Isol V
- 1071103; TCCP Program Review ID's Repeat Failures

Work Orders:
- 816204-04; Implement OpEval OE08-003 Corrective Action #1; 3/12/2009
- 1062732-02; Document Resolution of Issues; 5/14/2010

Miscellaneous:
- RS-080131; Letter from Keith Jury, Exelon Nuclear VP Licensing and Regulatory Affairs, Nine-Month Response to Generic Letter 2008-001; 10/14/2008
- RS-08-050; Letter from Keith Jury, Exelon Nuclear VP Licensing and Regulatory Affairs, Three Month Response to Generic Letter 2008-001; 10/11/2008
- RS-09-149; Letter from Patrick Simpson, Exelon Licensing Manager, Response to Request for Additional Information Regarding Generic Letter 2008-001; 11/3/2009
- List of all IRs related to water solid UT testing; undated

4OA3 Followup of Events and Notices of Enforcement Discretion

Procedures:
- MA-AA-716-210; Performance Centered Maintenance (PCM) Process; Rev. 10

Issue Reports:
- 1117744; 2VD05C and 2VD07C were found not running; 9/25/2010

Work Orders:
- 1117744-02; Perform a Prompt Investigation on the Event; 9/27/2010
- 1117744-09; Equipment Apparent Cause Evaluation; 11/4/2010
- 1117744-07; Perform Maintenance Rule Cause Determination Evaluation; 10/22/2010

Miscellaneous:
- LER 2010-01-00; High Pressure Core Spray System Declared Inoperable Due to Failed Room Ventilation Control Relay; 9/25/2010

4OA5 Other Activities

Procedures:
- CC-AA-309; Control of Design Analyses; Rev. 9
- LFP-400-1; Reactor Building Overhead Crane Critical L-Path Surveillance Test Prior to Cask Handling Operations in the Restricted Cask Mode; Rev. 5
- LFP-800-10; HI-STORM Haul Path and ISFSI Dry Run Operations; Rev. 0
- LFP-800-12; MPC Processing Dry Run Operations; Rev. 0
- LFP-800-2; Reactor Building Overhead Crane Cask Mode of Operation for Shipping Casks; Rev. 4
- LFP-800-63; HI-STORM Inspection; Rev. 1
- LFP-800-64; Transporter Operations; Rev. 0
- LFP-800-65; Spent Fuel Cask Site Transportation; Rev. 1
- LFP-800-68; HI-TRAC Preparation; Rev. 0
- LFP-800-69; HI-TRAC Movement within the Reactor Building; Rev. 3
- LFP-800-69; HI-TRAC Movement within the Reactor Building; Rev. 4
- LFP-800-70; HI-TRAC Loading Operations; Rev. 0
- LFP-800-70; HI-TRAC Loading Operations; Rev. 1
- LFP-800-70; HI-TRAC Loading Operations; Rev. 4
- LFP-800-71; MPC Processing; Rev. 0
- LFP-800-71; MPC Processing; Rev. 1
- LFP-800-71; MPC Processing; Rev. 2
- LFP-800-72; HI-STORM Processing; Rev. 1
- LFP-800-74; Helium Cooldown System Operation and MPC Reflood; Rev. 0
- LFP-800-75; LFPMPC Inspection; Rev. 1
- LFP-800-79; MPC Alternate Cooling; Rev. 0
- LFP-800-8; Spent Fuel Cask Contingency Actions; Rev. 0
- LFP-800-8; Spent Fuel Cask Contingency Actions; Rev. 0
- LFP-800-82 MPC Unloading Operations; Rev. 0
- LS-LA-104-101; 72.48 Review Process for Dry Cask Storage; Rev. 1
- NF-LA-622; Fuel Selection and Documentation for LaSalle Dry Cask Loading; Rev. 1
- RP-LA-304-1003; HI-STORM Radiation Survey; Rev. 1
- RP-LA-304-1004; ISFSI Radiation Survey; Rev. 2
- Sm-Aa-102; Warehouse Operations; Rev. 14

Issue Reports:
- 767072; Seismic Storage Issues in the Reactor Building
- 900610: Documenting ISFSI Project Issues/Questions from NRC; 3/13/2009
- 964823; Discrepancy between Design Drawings and Calculations for RB 843 Slab; 6/22/2009
- 966506; Use of NUREG-6865 in the Dynamic Analysis of the ISFSI Slab; 9/17/2009
- 1043750; Concerns on ISFSI Stack Up Calculation L-003423; 3/17/2010
- 1086968; NRC Identified Questions to L-003495 Require Rev. to Calculation; 6/9/2010
- 1087410; NRC Identified Comments on L-003399 Require Rev. to Calculations
- 1092094; Loss of Control Power and Indication; 7/19/2010
- 1093028; NRC Identified Comments on L-003411 Require Rev. to Calculation; 7/13/2010
- 1093347; NRC Observations during LaSalle DCS 72.212 Report Evaluations; July 21, 2010
- 1093426; NRC Identified Questions to L-003493 Require Rev. to Calculation; 6/20/2010

- 1093449; NRC Identified Comments on L-003408; Require Rev. to the Calculation; 7/13/20101093891; NRC Identified Comments on L-003417; Require Rev. to the Calculation; 7/13/2010
- 1093918; NRC Observations during Dry Cask Storage NRC Demonstrations; 7/23/2010
- 1095701; Additional NRC Questions Following Dry Cask Storage Demos; 8/28/2010
- 1098435; NRC Identified Questions to L-003400 Require Rev. to the Calculation; 4/22/2010
- 1099544; Dry Cask Transporter Front Left Wheel Assembly Jumping; 8/9/2010
- 1100670; CMTR Questions Raised During ISFSI DCS NRC Inspection; 8/12/2010
- 1102633; NRC Concerns with LaSalle ISFSI Pad Structural Qualification; 8/17/2010
- 1113084; NRC Identified ISFSI Calculation did not Consider Horizontal Load; 8/9/2010
- 1116522; RB Crane will not go North; 9/22/2010
- 1116816; MPC Rubbed Inside of HI-TRAC During Recovery Demonstration; 9/22/2010
- 1119512; PORC Review of LaSalle DCS Procedures; 9/16/2010
- 1119581; NRC Good Practice Observations during LaSalle Heavy Lift Demos; 9/24/2010
- 1120942; Rx Building Crane Main Hoist not functioning properly; October 1, 2010
- 1123826; Rx Building Crane Magnetorque Circuit found Wired Incorrectly; 10/6/2010
- 1137279; DCS HI-TRAC Evaluation for Tornado Forces; 8/9/2010

Work Orders:
- 1110656-11; Replace RB Crane Trolley; 7/14/2009
- 1298103-01; Dry Cask Storage HI-TRAC Trunnion Inspection; 1/20/2010
- 1298105-01; Dry Cask Storage Lift Yoke Assembly; 1/20/2010

Calculations:
- 710; Calculation Equipment Access Building; Rev. 0A
- HI-2012689; Holtec's Seismic Analysis Methodology for the ISFSI Pad/Cask Assemblage Including Soil Structure Interaction for Sequoyah Nuclear Plant; Rev. 2
- HI-2084236; HI-STORM CoC Radiation Protection Program Dose Rate Limits; Rev. 0
- L-003346; Structural Qualification of the ISFSI Pad at LaSalle under Static Plus Seismic Loading; Rev. 2
- L-003347; Final Design Basis Dynamic Analysis of LaSalle ISFSI Pad; Rev. 4A
- L-003353; ISFSI Dry Cask Storage – Fire Hazards Analysis; Rev. 0
- L-003382; Cask Handling Weight and Cask Handling Dimension for Byron, Braidwood, and LaSalle; Rev. 1
- L-003399; Floor Slab Evaluation for Spent Fuel Cask Loading – Reactor Building 710 and 843; Rev. 1
- L-003400; Decon Pit Grillage for Cask Loading – Reactor Building EL 843; Rev. 0
- L-003400; Decon Pit Grillage for Cask Loading – Reactor Building EL 843; Rev. 1
- L-003400; Decon Pit Grillage for Cask Loading – Reactor Building EL 843; Rev. 2
- L-003408; Reactor Building Crane- Critical Welds; Rev. 2
- L-003408; Reactor Building Crane- Critical Welds; Rev. 3
- L-003408; Reactor Building Crane- Critical Welds; Rev. 4
- L-003409; Reactor Building Crane – Miscellaneous Seismic Calculation; Rev. 3
- L-003410; Reactor Building Crane – NOG-1-2004 SFP Trolley Seismic Analysis; Rev. 1
- L-003411; Exelon/LaSalle Single Failure Proof Bridge Stress Analysis Report; Rev. 1
- L-003411; Exelon/LaSalle Single Failure Proof Bridge Stress Analysis Report; Rev. 2
- L-003411; Exelon/LaSalle Single Failure Proof Bridge Stress Analysis Report; Rev. 4
- L-003415; Reactor Building Crane Supporting Structure Analysis; Rev. 2
- L-003415; Reactor Building Crane Supporting Structure Analysis; Rev. 3A
- L-003417; Reactor Building Crane – Reactor Building Crane Hoist Reeving Equipment; Rev. 1
- L-003436; Reactor Building Crane – Simplified Trolley Model; Rev. 1

- L-003483; Structural Evaluations Associated with the Cask Stack-Up Restraint System; Rev. 0
- L-003493; Evaluations of the Reactor Building Structure for Loads Associated with the Cask Stack-Up Restraint System; Rev. 0
- L-003494; Evaluations of the Seismic Restraint Forces on the HI-STORM and HI-TRAC During Stack-Up; Rev. 0
- L-003495; Stability Evaluations for the HI-STORM and HI-TRAC Casks Inside the Reactor Building; Rev. 0
- L-003495; Stability Evaluations for the HI-STORM and HI-TRAC Casks Inside the Reactor Building; Rev. 1
- L-003497; Dose versus Distance from a HI-STORM 100S Version B Containing the MPC-68; Rev. 0
- L-003498; Tornado Evaluation for Byron Braidwood and LaSalle Nuclear Generating Station Dry Storage Projects; Rev. 0
- L-003499; Cask Stack-Up Bolting Evaluations; Rev. 0
- L-003582; Tornado Analysis for LaSalle HI-TRAC; Rev. 0

Drawings:
- S-1752; Reactor Building ISFSI Cask Stackup Restraint System Sections and Details; May 12, 2010
- HI-5492; Dry Fuel Storage Project Cask Storage Pad; Rev. 5

10 CFR 50.59 Screenings/Evaluations:
- EC 367004; Reactor Building Structure Modification in Support of Dry Cask Spent Fuel Storage and Crane Upgrade; Rev. 4

Miscellaneous:
- ALARA Plan Dry Cask Storage Activities; Rev. 0
- Dry Cask Overview Training for Operators
- Dry Cask Storage Organization Chart
- Dry Cask Storage Readiness Review PORC Presentation
- Dry Storage Position Paper DS-348; Rev. 0
- Dry Storage Position Paper DS-349; Rev. 0
- EP-AA-1005; Radiological Emergency Plan Annex for LaSalle Station; Rev. 29
- Fuel Selection Package – MPC-68-253
- Fuel Selection Package – MPC-68-254
- Fuel Selection Package – MPC-68-266
- Fuel Selection Package – MPC-68-276
- Fuel Selection Package – MPC-68-277
- Fuel Selection Package – MPC-68-278
- Fuel Selection Package – MPC-68-279
- Fuel Selection Package – MPC-68-280
- HI-STORM 100 Dry Cask Storage System Introduction Training for Radiation Protection and Other Disciplines
- HI-STORM Cask Daily Surveillance for EO
- Holtec Document 1676037a; BYNPS ISFSI Tributary Area; 1/30/2009
- Holtec Document 1678056; Byron/Braidwood ISFSI Pad Qualification; 7/26/2010
- Holtec Document 1678059; NRC Pad Concerns; 8/17/2010
- LaSalle County Power Station, Units 1 and 2, 10 CFR 72.212 Evaluation Report, Rev. 0
- LaSalle County Power Station, Units 1 and 2, 10 CFR 72.212 Evaluation Report, Rev. 1
- LaSalle Dry Cask Training Status; 4/26/2009
- LaSalle Identified DCS Procedural Changes

- NOG-1 Compliance Matrix for Byron, Braidwood, and LaSalle Fuel Handling Building and Reactor Building Cranes
- Nuclear Component Transfer List
- Quality Receipt Inspection Package – Helium Gas; 10/28/2009
- Spring 2010 Dry Cask Training Refresher for Operations
- Vertical Cask Transporter Tire Inspection

LIST OF ACRONYMS USED

AC	Alternating Current
ADAMS	Agencywide Document Access Management System
ALARA	As-Low-As-Is-Reasonably-Achievable
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
CAP	Corrective Action Program
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CoC	Certificate of Compliance
DC	Direct Current
DG	Diesel Generator
DNMS	Division of Nuclear Materials Safety
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
DSFST	Division of Spent Fuel Storage and Transportation
EDG	Emergency Diesel Generator
EPRI	Electrical Power Research Institute
FME	Foreign Material Exclusion
GL	Generic Letter
HI-STORM	Storage Cask
HI-TRAC	Transfer Cask
HPCS	High Pressure Core Spray
ISFSI	Independent Spent Fuel Storage Installation
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IR	Issue Report
IST	In-Service Test
LER	Licensee Event Report
LPCS	Low Pressure Core Spray
MCID	Materials Control, ISFSI, and Decommissioning
MOD	Modification
MPC	Multi-Purpose Canister
MSPI	Mitigating Systems Performance Index
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NMSS	Nuclear Material Safety and Safeguards
NRC	U.S. Nuclear Regulatory Commission
PARS	Publicly Available Records System
PI	Performance Indicator
OBE	Operating Basis Earthquake
QA	Quality Assurance
RHR	Residual Heat Removal
RP	Radiation Protection
RR	Reactor Recirculation
SBLC	Standby Liquid Control
SDP	Significance Determination Process
SFP	Spent Fuel Pool
SSC	Systems, Structures, and Components
SSE	Safe Shutdown Earthquake

SSI	Soil Structure Interaction
SW	Service Water
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
VD	Ventilation System
WO	Work Order

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Sincerely,

/RA/

Kenneth Riemer, Chief
Branch 2
Division of Reactor Projects

Docket Nos. 50-373; 50-374; 72-070
License Nos. NPF-11; NPF-18

Enclosure: Inspection Report 05000373/2010005; 05000374/2010005; 07200070/2010001
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Letter to M. Pacilio from K. Riemer dated February 4, 2011

SUBJECT: LASALLE COUNTY STATION, UNITS 1 AND 2
NRC INTEGRATED INSPECTION REPORT
05000373/2010005; 05000374/2010005; 07200070/2010001

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