July 27, 2005

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

SUBJECT: LASALLE COUNTY STATION, UNITS 1 AND 2
NRC INTEGRATED INSPECTION REPORT 05000373/2005003;
05000374/2005003

Dear Mr. Crane:

On June 30, 2005, the U.S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your LaSalle County Station, Units 1 and 2. The enclosed report documents the results of this inspection discussed on July 13, 2005, with the Site Vice-President, Ms. Susan Landahl, and other members of your staff.

The inspection examined activities conducted under your license as they related 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 findings and one self-revealed finding of very low safety significance were identified. All of these findings also involved violations of NRC requirements. However, because the findings associated with these violations were of very low safety significance and because the issues were entered into the licensee's corrective action program, the NRC is treating these issues as Non-Cited Violations in accordance with Section VI.A.1 of the NRC's Enforcement Policy.

If you contest the subject or severity of any Non-Cited Violation in this report, 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 copies 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 NRC Resident Inspectors' Office at the LaSalle County Station.

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Sincerely,

/RA/

Bruce L. Burgess, Chief
Branch 2
Division of Reactor Projects

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

Enclosure: Inspection Report 05000373/2005003; 05000374/2005003
w/Attachment: Supplemental Information

cc w/encl: Site Vice President - LaSalle County Station
LaSalle County Station Plant Manager
Regulatory Assurance Manager - LaSalle County Station
Chief Operating Officer
Senior Vice President - Nuclear Services
Senior Vice President - Mid-West Regional Operating Group
Vice President - Mid-West Operations Support
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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-373; 50-374
License Nos:	NPF-11; NPF-18
Report No:	05000373/2005003; 05000374/2005003
Licensee:	Exelon Generation Company, LLC
Facility:	LaSalle County Station, Units 1 and 2
Location:	2601 N. 21st Road Marseilles, IL 61341
Dates:	April 1 through June 30, 2005
Inspectors:	D. Kimble, Senior Resident Inspector D. Eskins, Resident Inspector M. Jordan, NRC Contractor M. Mitchell, Radiation Protection Inspector R. Winter, Engineering Inspector J. Yesinowski, Illinois Dept. of Emergency Management
Approved by:	Bruce L. Burgess, Chief Branch 2 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000373/2005003, 05000374/2005003; 04/01/2005 - 06/30/2005; LaSalle County Station, Units 1 & 2; Fire Protection, Operator Performance During Non-Routine Plant Evolutions and Events, Identification and Resolution of Problems, and Other Report.

The inspection was conducted by resident inspectors and regional inspectors. The report covers a 3-month period of baseline resident inspection, and announced baseline inspections in radiation protection and engineering. Four findings of very low safety significance (Green) were identified, each with an associated Non-Cited Violation. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using NRC Inspection Manual Chapter (IMC) 0609 “Significance Determination Process” (SDP). Findings for which the SDP does not apply may be "Green," or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. Inspector-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. A finding of very low safety significance was identified by an NRC inspector conducting a routine observation of licensee maintenance activities associated with the removal and replacement of containment isolation check valve 1E51-F028. An associated Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI, “Corrective Action,” was also identified.

  The performance deficiency identified by the inspectors involved inadequate ignition controls for the hot work being performed on the valve. The finding was of more than minor significance in that it had a direct impact on the cornerstone objective. Specifically, the licensee’s performance deficiencies allowed sparks from the work to reach an uncovered safety-related cable tray in the vicinity of the work location. Because the safety-related cable tray in question contained only cables associated with the Unit 1 reactor core isolation cooling system (RCIC), which was inoperable, unavailable, and within the allowed outage time permitted by plant Technical Specifications at the time of the finding, and because all the cables in the tray were qualified to IEEE-383-1974, “Institute of Electrical and Electronic Engineers (IEEE) Standards for Type Test of Class 1E Electric Cables, Field Splices and Connections for Nuclear Generation Stations,” for flame retardation, the inspectors determined the finding to have been of very low safety significance (Green) and within the licensee's response band. Corrective actions planned and completed by the licensee included: revocation of hot work fire watch qualifications for all station mechanics; assignment of the station’s Fire Marshal to provide direct oversight of remaining 1E51-F028 hot work activities once they resumed; and new and revised hot work training for all mechanical maintenance personnel prior to their recertification for the performance of hot work activity. The finding was determined to involve the cross-cutting aspect of problem identification and resolution. (Sections 1R05.3 and 4OA2.1)

- Green. A finding of very low safety significance was self-revealed when Unit 1 reactor power inadvertently rose to approximately 103.17 percent on February 23, 2005, and went unnoted by the on-watch control room crew for several minutes. A Non-Cited Violation of Condition 2.C (1) of NRC Facility Operating License No. NPF-11 for LaSalle County Station, Unit 1, was also identified.

  The performance deficiency for this finding involved the collective distraction of on-watch control room personnel that removed crew focus from their primary duty of monitoring reactor and plant parameters. The finding was of more than minor significance in that it had a direct impact on initiating events cornerstone objective "to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations." Because the finding only affected the probability of a reactor trip and no mitigating systems were impacted, it was determined to have been of very low safety significance (Green) and within the licensee's response band. Corrective actions taken by the licensee in response to the event included: maintaining the reactor recirculation flow control valves in manual, pending the results of investigation into possible faults with the recirculation flow controllers; immediate relief of the on-watch Unit 1 control room crew; changing several plant process computer alarms (MWth, MWe, and reactor pressure) from low-level alarms, which annunciate only briefly and then are automatically silenced, to higher level alarms that require operator action to silence the alarm tones; and establishment of robust physical barriers around the recirculation flow control switches to prelude them from being inadvertently bumped. The finding was determined to have involved the cross-cutting aspect of human performance. (Sections 1R14.2 and 4OA4)

Cornerstone: Mitigating Systems

- Green. A finding of very low safety significance was identified by inspectors, who determined that the licensee failed to take timely and effective corrective action for water intruding into safety-related electrical junction boxes and control cabinets via electrical conduit from the outside. An associated Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," was also identified.

  The inspectors determined that the performance deficiency associated with this issue centered around the licensee's failure to give proper priority to the issue and the actions needed to resolve it. The inspectors determined that the finding was of more than minor significance in that it had a direct impact on the Mitigating Systems cornerstone objective. Because the finding did not represent the loss of any safety function for any system or train, and because it was determined not to be potentially significant with respect to any external events such as seismic, flooding, tornado, etc., the inspectors determined it to be of very low safety significance (Green) and within the licensee's response band. Corrective actions taken or planned by the licensee include: a complete extent-of-condition review of all through roof conduits that may be susceptible to water intrusion; drilling of weep holes in all susceptible junction boxes; repairs to damage caused by water intrusion; and the sealing of the leaking conduit on Unit 1, Division 1 and Division 2 safety-related ventilation systems. The finding was determined to involve the cross-cutting area of identification and resolution of problems. (Section 4OA2.4)

- Green. The inspectors identified a finding of very low safety significance. During a review of test procedures used to maintain standby liquid control (SBLC) tank volume and concentration within Technical Specification limits, the inspectors identified that the licensee had used inaccurate and nonconservative instruments to measure SBLC tank level. An associated Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XI, “Test Control,” was also identified.

  The inspector-identified performance deficiency associated with this issue was a failure by the licensee’s staff to utilize adequate test equipment for the performance of safety-related Technical Specification surveillance measurements of SBLC solution tank level. The inspectors determined that the finding was of more than minor significance in that it had a direct impact on the Mitigating Systems cornerstone objective. The finding was determined to be of very low safety significance because subsequent licensee analyses of SBLC tank concentrations and volumes, in accordance with GL 91-18, demonstrated that the errors in SBLC tank volume in question were sufficiently small as to not have jeopardized the capability of SBLC to have performed its safety function for either unit. Corrective actions by the licensee included: additions of sodium pentaborate chemical to each unit’s SBLC tank to adjust chemistry to well within the Technical Specification required band; revision of SBLC tank sampling procedures; and the establishment of administrative controls to ensure that each unit’s SBLC tank volume and sodium pentaborate solution concentration are being maintained well away from Technical Specification limits; and the procurement of new T-squares instruments for measuring SBLC tank level, which were manufactured in accordance with 10 CFR 50, Appendix B, Quality Assurance Program controls and requirements. (Section 4OA5.1)

B. Licensee-Identified Violations

No violations of significance were identified.

REPORT DETAILS

Summary of Plant Status

Unit 1

The unit began the inspection period operating at full power. On May 12, 2005, power was reduced to approximately 83 percent in response to a heater drain system transient experienced during restoration of a heater drain valve to service following minor maintenance. The unit returned to operation at full power on May 13, 2005. On May 22, 2005, power was reduced to approximately 67 percent to permit a control rod sequence exchange and control rod surveillance testing. Return to full power was delayed due to an air leak on a feedwater heater normal drain valve that required additional troubleshooting and repair. The unit returned to operation at full power on May 24, 2005, and continued operating at or near full power for the remainder of the inspection period.

Unit 2

The unit began the inspection period operating at full power. On April 5, 2005, power was reduced to approximately 83 percent to facilitate troubleshooting and repairs to low pressure feedwater heater level control. Troubleshooting and repairs were completed and the unit was returned to full power on the same day. On May 21, 2005, power was reduced to approximately 57 percent for a control rod sequence exchange, control rod surveillance testing, and recovery of maintenance control rods. Operation at full power was resumed on May 22, 2005. On June 18, 2005, the unit reduced power to approximately 65 percent to conduct minor repairs on the heater drain system. The unit returned to operation at full power on June 19, 2005, and continued operating at or near full power for the remainder of the inspection period.

1. REACTOR SAFETY

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

1R01 Adverse Weather (71111.01)

a. Inspection Scope

The inspectors performed a walkdown of the licensee's preparations for summer weather for selected systems, including conditions that could lead to loss of off-site power and conditions that could result from high temperatures. During the inspection, the inspectors focused on plant specific design features and the licensee's procedures used to mitigate or respond to adverse weather conditions. Additionally, the inspectors reviewed the Updated Final Safety Analysis Report (UFSAR) and performance requirements for systems selected for inspection, and verified that operator actions were appropriate as specified by plant specific procedures. The inspectors' reviews focused specifically on the following plant systems:

- Lake temperature monitoring;
- Reactor building closed loop cooling water (RBCCW); and
- Offsite power supplies to emergency distribution busses.

This review constituted a single inspection sample.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Quarterly Partial System Alignment Verifications

a. Inspection Scope

The inspectors performed partial walkdowns of the following equipment trains to verify operability and proper equipment lineup. These systems were selected based upon risk significance, plant configuration, system work or testing, or inoperable or degraded conditions.

- Unit 1 and Unit 2 Division 2 emergency diesel generators (EDGs) during maintenance on the common Division 1 EDG;
- 'B' diesel fire pump during maintenance activities on the 'A' diesel fire pump;
- Unit 1 Division 1 EDG, core standby cooling systems (CSCS), and electrical distribution system during Division 2 EDG inoperability; and
- Unit 1 reactor core isolation cooling (RCIC) following maintenance.

The inspectors verified the position of critical redundant equipment and looked for any discrepancies between the existing equipment lineup and the required lineup.

These partial equipment alignment verifications constituted four inspection samples.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Quarterly Fire Zone Inspections

a. Inspection Scope

The inspectors walked down the following risk significant areas looking for any fire protection issues. The inspectors selected areas containing systems, structures, or components that the licensee identified as important to reactor safety.

- Fire Zone 2B1, Unit 1 reactor building elevation 820'6";
- Fire Zone 2B2, Unit 1 reactor building elevation 820'6";

- Fire Zone 3B1, Unit 2 reactor building elevation 820'6”;
- Fire Zone 5D1, Unit 1 high pressure core spray switchgear zone 687'0";
- Fire Zone 7B1, Unit 1 high pressure core spray diesel generator room 710'6"; and
- Fire Zone 7B4, Unit 1 high pressure core spray diesel day tank room 710'6".

The inspectors reviewed the control of transient combustibles and ignition sources, fire detection equipment, manual suppression capabilities, passive suppression capabilities, automatic suppression capabilities, barriers to fire propagation, and any contingency fire watches that were in effect.

These reviews constituted six inspection samples.

b. Findings

No findings of significance were identified.

.2 Annual Fire Drill Observation

a. Inspection Scope

To evaluate the readiness of licensee personnel to fight fires, on May 1, 2005, the inspectors observed the fire brigade respond to a simulated electrical fire in the Unit 2 Turbine Building. The following aspects of the response were reviewed:

- Use of protective clothing and self-contained breathing apparatus (SCBAs);
- Use of fire hoses to demonstrate the capability to reach all necessary fire hazard locations without flow constrictions;
- Testing of hose nozzle patterns prior to entering the fire area;
- Entry into the fire area in a controlled manner;
- Presence of sufficient fire fighting equipment at the scene for the fire brigade to properly perform their fire fighting duties;
- Effectiveness and clarity of the fire brigade leader’s directions;
- Efficiency and effectiveness of radio communications between plant operators and fire brigade members;
- Checking for fire victims and fire propagation into other plant areas; and
- Effectiveness of simulated smoke removal operations.

The inspectors’ review of this annual fire drill constituted a single inspection sample.

b. Findings

No findings of significance were identified.

.3 Observation of 1E51-F028 Hot Work

a. Inspection Scope

On May 5, 2005, during removal and replacement of the Unit 1 RCIC barometric condenser vacuum pump return line containment isolation check valve, 1E51-F028, the inspectors reviewed the licensee's maintenance activities specifically for any fire protection issues. The inspectors reviewed the control of transient combustibles and ignition sources, fire detection equipment, manual suppression capabilities, passive suppression capabilities, automatic suppression capabilities, barriers to fire propagation, and the performance of the hot work fire watch that was in place for the planned maintenance activities.

These observations constituted a single inspection sample.

b. Findings

Introduction

A finding of very low safety significance (Green) was identified by an NRC inspector conducting a routine observation of licensee maintenance activities associated with the removal and replacement of containment isolation check valve 1E51-F028. During grinding to remove the valve from the system, the inspector observed that the licensee's ignition control barriers for the hot work were insufficient, and that sparks from the grinding were reaching an uncovered and unprotected safety-related cable tray near the work location. A Non-Cited Violation (NCV) of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," was also identified.

Description

At approximately 7:30 a.m. on the morning of May 5, 2005, the NRC Resident Inspector was conducting a routine baseline inspection associated with the removal and replacement of the 1E51-F028 check valve. Upon arrival at the work site, the inspector noted that the fire blanketing that had been provided for the job did not extend out from the work location by the 35 feet specified in plant ignition control procedures. In fact, in the inspector's estimation, the fire blanketing only extended out from the location of the hot work by approximately 6 to 8 feet. The inspector pointed this out to the Mechanical Maintenance Supervisor in the area providing oversight for the work, who informed the inspector that the fire blanket coverage was adequate because a fire watch had been stationed and could observe any sparks being introduced into areas beyond the fire blanket protection. Additionally, the Mechanical Maintenance Supervisor told the inspector that in his estimation any sparks from the hot work that would travel out beyond the existing fire blanket protection would be "self-extinguishing," and thus of no concern.

When the actual hot work (grinding) began, the inspector again voiced several concerns over the ignition controls present for the job to the Maintenance Department personnel present. Sparks from the grinding activities were observed to be entering an uncovered cable tray containing electrical cables. The cable tray, which was approximately 4 to

6 feet from the location of the hot work was labeled indicating that it contained safety-related Division 1 equipment. Because the inspector's earlier concerns voiced to the personnel present at the job location had not resulted in any visible corrective actions, the inspector left the work location and proceeded to the main control room to discuss the issue with the on-watch Operations Shift Manager. In response to the inspector's concerns, the Shift Manager suspended all work at the 1E51-F028 job site at approximately 11:30 a.m.

The licensee initiated an investigation into the inspector's concerns, and determined that they were valid. As part of the immediate corrective action for the issue, the licensee revoked hot work fire watch qualifications for all station mechanics, and assigned the station's Fire Marshal to provided direct oversight of the 1E51-F028 hot work activities once they resumed later that day. Hot work training for mechanical maintenance personnel was revised, and all department personnel were subsequently retrained and requalified on hot work activity performance.

Analysis

The inspectors determined that there was a licensee performance deficiency associated with the fire blanket coverage provided for the job. Specifically, the fire blanket coverage was inadequate in that it did not fully contain all the sparks being generated from the cutting activity, and was not in compliance with the licensee's established procedure governing hot work ignition controls. Procedure OP-MW-201-004, "Fire Prevention for Hot Work," Section 4.2, "Fire Prevention Precautions," required fire blanket coverage out to 35 feet from the work location. In this case, the fire blanket coverage went out to approximately 6 to 8 feet from the work location.

In addition, a second performance deficiency associated with the duties of the fire watch was identified. Procedure OP-MW-201-004, Section 3.4.2, discussed the duties of the fire watch, and required that each fire watch was responsible for stopping the hot work in the event of any safety problems, such as sparks coming in contact with combustible material, etc. In reviewing this issue, the inspectors noted that although the fire watch had been stationed for the hot work per the licensee's ignition control procedures, as was the case with the rest of the maintenance personnel involved in the job, the fire watch was not concerned with the lack of fire blanket coverage or the fact that sparks were traveling out beyond the coverage provided.

The objective of the Initiating Events Cornerstone of Reactor Safety is "to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations." In accordance with NRC Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," the inspectors determined that the finding was of more than minor significance in that it had a direct impact on this cornerstone objective. More specifically, one of the key attributes associated with this cornerstone objective is protection against fires, and the inspectors determined that the licensee's performance deficiencies had contributed to an increased likelihood of a fire occurring in the vicinity of the 1E51-F028 work site.

The inspectors determined that the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," and conducted a Phase 1 characterization and initial screening. Because the finding was associated with fire protection, this was accomplished using IMC 0609, Appendix F, Attachment 1, “Fire Protection SDP Phase 1 Worksheet.”

The inspectors assigned a duration factor (DF) of 0.01 for the finding because it was associated with a condition lasting less than three days. Additionally, a generic fire area frequency of 2E-3 was assigned due to the finding involving hot work control deficiencies. These factors combined to yield a ΔCDF [core damage frequency] of 2E-5, which indicated a potentially significant (high) degradation in fire prevention and administrative controls and the need for a Phase 2 analysis.

During the Phase 2 Fire Protection SDP, the inspectors obtained assistance from NRC Region III fire protection engineers. Because the safety-related cable tray in question contained only cables associated with the Unit 1 RCIC system, which was inoperable, unavailable, and within the allowed outage time permitted by plant Technical Specifications at the time of the finding, and because all the cables in the tray were qualified to IEEE-383-1974, “Institute of Electrical and Electronic Engineers (IEEE) Standards for Type Test of Class 1E Electric Cables, Field Splices and Connections for Nuclear Generation Stations,” for flame retardation, the inspectors determined the finding to have been of very low safety significance (Green) and within the licensee's response band. Because the finding involved the cross-cutting aspect of identification and resolution of problems, it is also discussed in Section 4OA2.1, “Identification and Resolution of Problems,” in this report.

Enforcement

As discussed in NRC Inspection Report 05000373/2005002; 05000374/2005002, Sections 1R05.2 and 4OA4, the inspectors had previously identified similar performance deficiencies associated with hot work ignition controls in February 2005 during the licensee’s L2R10 refuel outage. These deficiencies had ultimately resulted in a small Class ‘A’ fire in the 2B residual heat removal (RHR) corner room on February 16, 2005.

The hot work within the 2B RHR corner room on February 16, 2005, as well as the hot work associated with 1E51-F028, both involved maintenance and repair on equipment subject to the requirements of 10 CFR 50, Appendix B. Criterion XVI of this appendix states, in part, that: “Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected.” Contrary to this requirement, the inspectors determined that the licensee’s corrective actions for the hot work ignition control deficiencies that had led to the 2B RHR corner room fire on February 16, 2005, were largely ineffective, to the extent that the inspectors were able to identify the same deficiencies during hot work associated with the 1E51-F028 check valve on May 5, 2005. A subsequent apparent cause evaluation (ACE) by the licensee determined that maintenance personnel qualified to perform hot work and hot work fire watches had numerous misconceptions regarding ignition control procedural requirements. As discussed above, in response to this issue and the subsequent ACE the licensee revoked hot work fire watch qualifications for all station mechanics, assigned the station's Fire Marshal to provide direct oversight of the 1E51-F028 hot work activities once they resumed later that day, and required new and revised hot work training for all mechanical maintenance personnel prior to them being recertified for the performance of hot work activity. The licensee had entered this issue into their corrective action program as issue report (IR) 332429.

Because the licensee had entered the issue into their corrective action program and the finding is of very low safety significance (Green), this violation of 10 CFR 50, Appendix B, Criterion XVI, is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000373/2005003-01)

1R06 Flood Protection Measures (71111.06)

.1 Annual External Flooding Review

a. Inspection Scope

The inspectors reviewed the licensee's flooding mitigation plans and equipment to determine consistency with design requirements and the risk analysis assumptions related to seasonal external flooding. As discussed in NRC Inspection Report 05000373/2003003; 05000374/2003003, design basis documentation indicated that LaSalle was classified as a "dry" site since external flooding was not a threat to the plant. This was based on the top of the LaSalle dike being at the 710 foot elevation and the plant grade being at 710 feet, 6 inches. Probable Maximum Flooding (PMF) is at an elevation of 704 feet, 4 inches. As a result, the inspectors focused on changes made to the facility over the past year that might affect the site's "dry" classification. Walkdowns and reviews performed considered design measures, seals, drain systems, contingency equipment condition and availability of temporary equipment and barriers, performance and surveillance tests, procedural adequacy, and compensatory measures.

This annual external flooding review constituted a single inspection sample.

b. Findings

No findings of significance were identified.

.2 Semiannual Internal Flooding Review

a. Inspection Scope

The inspectors reviewed the licensee's flooding mitigation plans and equipment to determine consistency with design requirements and the risk analysis assumptions related to internal flooding. The following specific plant areas particularly susceptible to internal flooding were inspected:

- Unit 1 and Unit 2 condenser pits; and
- Unit 1 and Unit 2 core standby cooling system rooms.

Walkdowns and reviews performed considered design measures, seals, drain systems, contingency equipment condition and availability of temporary equipment and barriers, performance and surveillance tests, procedural adequacy, and compensatory measures.

This semiannual internal flooding review constituted a single inspection sample.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope

The inspectors observed a training crew during an evaluated simulator scenario and reviewed licensed operator performance in mitigating the consequences of events. The scenario included a reactor scram and entry into the licensee's emergency plan. Areas observed by the inspectors included: clarity and formality of communications, timeliness of actions, prioritization of activities, procedural adequacy and implementation, control board manipulations, managerial oversight, emergency plan execution, and group dynamics.

The observation of this training scenario represented a single inspection sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Quarterly Resident Inspector Maintenance Effectiveness Reviews

a. Inspection Scope

The inspectors reviewed the licensee's handling of performance issues and the associated implementation of the Maintenance Rule (10 CFR 50.65) to evaluate maintenance effectiveness for the selected systems. The following systems were selected based on being designated as risk significant under the Maintenance Rule, being in the increased monitoring (Maintenance Rule category a(1)) group, or due to an inspector identified issue or problem that potentially impacted system work practices, reliability, or common cause failures:

- Unit 1 and Unit 2 emergency diesel generator and core standby cooling system ventilation systems after discovery of water intrusion into an electrical junction box;
- Unit 1 reactor manual control system (RMCS) after multiple system trips; and
- Unit 1 RCIC system vacuum discharge isolation valve, 1E51-F028, replacement and testing.

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

The inspectors' review of these issues constituted three inspection samples.

b. Findings

No findings of significance were identified.

.2 Biennial Maintenance Effectiveness Evaluation

a. Inspection Scope

The inspectors examined the periodic evaluation report completed for the period of July 2002 through June 2004. To evaluate the effectiveness of (a)(1) and (a)(2) activities, the inspectors examined a sample of (a)(1) Action Plans, Performance Criteria, Functional Failures, and Issue Reports (IRs). These same documents were reviewed to verify that the threshold for identification of problems was at an appropriate level and the associated corrective actions were appropriate. Also, the inspectors reviewed the Maintenance Rule procedures and processes. The inspectors focused the inspection on the following four systems: the direct current electrical system (DC); the emergency diesel generators (EDGs); the reactor recirculation system (RR); and the feedwater system (FW).

The inspector verified that:

- The periodic evaluation was completed within the time restraints defined in 10 CFR 50.65 (once per refueling cycle, not to exceed 24 months). The inspector also ensured that the licensee reviewed its goals, monitored Structures, Systems, and Components (SSCs) performance, reviewed industry operating experience, and made appropriate adjustments to the Maintenance Rule program as a result of the above activities;

- The licensee balanced reliability and unavailability during the previous refueling cycle, including a review of safety significant SSCs;

- That (a)(1) goals were met, that corrective action was appropriate to correct the defective condition, including the use of industry operating experience, and that (a)(1) activities and related goals were adjusted as needed; and

- That the licensee had established (a)(2) performance criteria, examined any SSCs that failed to meet their performance criteria, and reviewed any SSCs that have suffered repeated maintenance preventable functional failures including a verification that failed SSCs were considered for (a)(1).

In addition, the inspectors reviewed Maintenance Rule self-assessments that addressed the Maintenance Rule program implementation.

The inspectors' biennial review of these systems constituted four inspection samples.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed and observed emergent work, preventive maintenance, or planning for risk significant maintenance activities. The inspectors observed maintenance or planning for the following activities or risk significant systems undergoing scheduled or emergent maintenance:

- Troubleshooting and repair of the Unit 1 drywell floor drain sump leak rate detection system;
- Signs of water intrusion found in diesel generator room ventilation systems;
- Unit 1 RCIC system vacuum discharge isolation valve, 1E51-F028, replacement and testing;
- Unit 2 Division 1 125 Vdc troubleshooting and repairs; and
- Unit 2 Alterex brush arcing troubleshooting and repairs.

The inspectors also reviewed the licensee's evaluation of plant risk, risk management, scheduling, and configuration control for these activities in coordination with other scheduled risk significant work. The inspectors verified that the licensee's control of activities considered assessment of baseline and cumulative risk, management of plant configuration, control of maintenance, and external impacts on risk. In-plant activities were reviewed to ensure that the risk assessment of maintenance or emergent work was complete and adequate, and that the assessment included an evaluation of external factors. Additionally, the inspectors verified that the licensee entered the appropriate risk category for the evolutions.

The inspectors' reviews of these issues constituted five inspection samples.

b. Findings

No findings of significance were identified.

1R14 Operator Performance During Non-Routine Plant Evolutions and Events (71111.14)

.1 Operator Response to Loss of Safety-Related 480 Vac Buses 235X and 235Y

a. Inspection Scope

The inspectors performed several hours of continuous control room observation to evaluate operator performance in coping with an unexpected trip of the Bus 235X and 235Y 4160 Vac feeder breaker on June 21, 2005. The inspectors reviewed operator logs and plant computer data to determine how the unit responded and to verify that operator actions were appropriate, and consistent with operator training and plant procedures. The licensee's planned recovery actions, procedures, reactivity manipulation briefings, and contingency plans were also reviewed by the inspectors to identify any personnel performance issues. In addition, the inspectors verified that any problems encountered during the non-routine evolution were identified by the licensee, and appropriately entered into the corrective action program.

The observation of this non-routine evolution by the inspectors constituted a single inspection sample.

b. Findings

No findings of significance were identified.

.2 (Closed) Unresolved Item 05000373/2005002-11: Inadvertent Reactor Recirculation Flow Increase Results in Unit 1 Reactor Power Excursion to 103.17 Percent

a. Inspection Scope

On February 23, 2005, the inspectors responded to the control room following notification from the licensee that the Unit 1 licensed reactor power limit of 3489 megawatts thermal (MWth) had been exceeded by approximately 3.17 percent for several minutes following an unplanned and unexpected increase in reactor recirculation flow. The inspectors observed plant parameters and status; evaluated the performance of plant systems and licensee actions; and confirmed that the licensee properly reported the event as required by Section 2.F(a) of Facility Operating License No. NPF-11. The inspectors further determined that no nuclear fuel thermal limits were violated, and that the event was bounded by the events discussed in the UFSAR.

Following the event, the licensee conducted a comprehensive root cause analysis of operator performance for the event. The inspectors' review of this root cause analysis and inspection of operator performance for the event constituted a single inspection sample.

b. Findings

Introduction

A finding of very low safety significance (Green) was self-revealed when Unit 1 reactor power inadvertently rose to approximately 103.17 percent on February 23, 2005, and went unnoted by the on-watch control room crew for several minutes. A Non-Cited Violation of Condition 2.C (1) of NRC Facility Operating License No. NPF-11 for LaSalle County Station, Unit 1, was also identified.

Description

On February 23, 2005, at approximately 11:41 a.m., Unit 1 exceeded License Condition 2.C (1), which limits the maximum thermal power of the unit to 3489 MWth. Unit 1 reached a peak transient power of approximately 3599.5 MWth, or 103.17 percent of the licensed limit, for about eight minutes.

At approximately 11:46 a.m., the Unit 1 control room supervisor (CRS), a licensed senior reactor operator (SRO) observed that Unit 1 electrical power had increased from 1194 megawatts electric (MWe) to 1223 MWe, and directed the on-watch nuclear station operator (NSO), a licensed reactor operator (RO) to lower reactor power to 95 percent. From approximately 11:47 a.m. to 11:48 a.m., the NSO attempted to reduce reactor power using the “LOWER” pushbutton on the reactor recirculation (RR) ganged (i.e., master) flow control station. After two attempts to lower power using the RR ganged flow control station, the NSO did not believe that power was responding as it should have, and he placed the RR flow controllers for each RR loop’s flow control valve (FCV) into manual and closed them both to approximately 80 percent at 11:49 a.m. The FCVs responded, and reactor power was reduced to about 3471 MWth, or approximately 99.5 percent. Plant power was subsequently stabilized at about 95 percent while the licensee began an investigation of the event.

While the event did not trigger any control board annunciator alarms (none should have been triggered based on a review of the event by the inspectors), several low-level plant process computer alarms were actuated on increasing plant pressure and power. However, because the on-watch operations crew had collectively become distracted by several other ongoing work activities, it was several minutes before the crew became aware that reactor power had changed.

At approximately 5:34 p.m., the licensee contacted the NRC Region III Director of Reactor Projects via telephone, in accordance with the reporting conditions of Section 2.F(a) of the Unit 1 license, to discuss the event. A subsequent written report was sent on March 9, 2005.

Analysis

The inspectors determined that the collective distraction of on-watch control room personnel that removed crew focus from their primary duty of monitoring reactor and plant parameters constituted a licensee performance deficiency.

Specifically, the Unit 1 crew allowed themselves to become so involved with ongoing work activities on the morning of February 23, 2005, that they were no longer cognizant of changes that were occurring in critical reactor parameters.

The objective of the Initiating Events Cornerstone of Reactor Safety is "to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations." In accordance with NRC Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," the inspectors determined that the finding was of more than minor significance in that it had a direct impact on this cornerstone objective.

The inspectors determined that the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," and conducted a Phase 1 characterization and initial screening. Because the finding only affected the probability of a reactor trip and no mitigating systems were impacted, the inspectors determined the finding to have been of very low safety significance (Green) and within the licensee's response band. Because the finding involved the cross-cutting aspect of human performance, it is also discussed in Section 4OA4, “Cross-Cutting Aspects of Findings,” in this report.

Enforcement

Condition 2.C (1) of NRC Facility Operating License No. NPF-11 for LaSalle County Station, Unit 1, limits the maximum thermal power of the unit to 3489 MWth. Contrary to this requirement, on February 23, 2005, from approximately 11:41 a.m. until approximately 11:49 a.m., Unit 1 reached a peak transient power of about 3599.5 MWth, or 103.17 percent of the licensed limit.

The licensee’s investigation into potential equipment problems associated with the reactor recirculation flow controller was entered into their corrective action program as IR 304613. Operations crew performance issues were entered into the corrective action program as IR 305612, and became the focal point for the licensee’s root cause analysis.

Corrective actions taken by the licensee in response to the event included: maintaining the reactor recirculation flow control valves in manual, pending the results of investigation into possible faults with the recirculation flow controllers; immediate relief of the on-watch Unit 1 control room crew; changing several plant process computer alarms (MWth, MWe, and reactor pressure) from low-level alarms, which annunciate only briefly and then are automatically silenced, to higher level alarms that require operator action to silence the alarm tones; and establishment of robust physical barriers around the recirculation flow control switches to prelude them from being inadvertently bumped.

Because the licensee had entered the issue into their corrective action program and the finding is of very low safety significance (Green), this violation of Condition 2.C (1) of NRC Facility Operating License No. NPF-11 for LaSalle County Station, Unit 1, is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy.
(NCV 05000373/2005003-02)

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the technical adequacy of the following operability evaluations to determine the impact on Technical Specifications, the significance of the evaluations, and to ensure that adequate justifications were documented.

- Unit 1 diesel generator cooling water pump discharge check valves;
- Unit 1 and Unit 2 reactor core isolation cooling system steam line tunnel temperature primary containment isolation Technical Specification allowable value;
- Unit 2 Division 1 seismic qualification of 125 Vdc distribution panels with access covers removed;
- Reactor core maximum fraction of limiting critical power ratio (MFLCPR) errors; and
- Unit 2 instrument nitrogen north automatic depressurization system (ADS) check valve 2IN043 leakage.

Operability evaluations were selected based upon the relationship of the safety-related system, structure, or component to risk.

The inspectors' review of these operability evaluations constituted five inspection samples.

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

.1 Reactor Recirculation Pump Trip due to Instrument Ringing

a. Inspection Scope

The inspectors reviewed Operator Workaround (OWA) No. 353, "Reactor Recirculation Pump Trip due to Instrument Ringing." The inspectors reviewed this workaround's potential to impact the operators' ability to respond to plant transients involving main turbine trips. Specifically, the inspectors examined how such transients could result in erroneous trips of the reactor recirculation pumps, which would complicate operator response to events such as a reactor scram.

This review constituted one inspection sample.

b. Findings

No findings of significance were identified.

.2 Semiannual Cumulative Review of Operator Workarounds

a. Inspection Scope

The inspectors performed a semiannual review of the cumulative effects of operator workarounds. The cumulative effects of workarounds on the reliability, availability, and potential for improper operation of systems were reviewed. Additionally, reviews were conducted to determine if the workarounds could increase the possibility of an initiating event, affect multiple mitigating systems, or impact the operators' ability to respond to accidents or transients.

This review constituted one inspection sample.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors selected the following post-maintenance activities for review. Activities were selected based upon the structure, system, or component's ability to impact risk:

- Unit 1 RCIC system vacuum discharge isolation valve, 1E51-F028, post replacement testing;
- Unit 2 motor driven reactor feed pump post repair testing;
- Unit 1 standby liquid control relief valve post repair testing;
- Unit 2 reactor core isolation cooling system water leg pump post maintenance testing and calibration; and
- Unit 1 RCIC system barometric condenser vacuum pump following replacement.

The inspectors verified by witnessing the test or reviewing the test data that post-maintenance testing activities were adequate for the above maintenance activities. The inspectors reviews included, but were not limited to, integration of testing activities, applicability of acceptance criteria, test equipment calibration and control, procedural use and compliance, control of temporary modifications or jumpers required for test performance, documentation of test data, Technical Specification applicability, system restoration, and evaluation of test data. Also, the inspectors verified that maintenance and post-maintenance testing activities adequately ensured that the equipment met the licensing basis, Technical Specifications, and Updated Final Safety Analysis Report (UFSAR) design requirements.

The inspectors' review of these post maintenance tests constituted five inspection samples.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors selected the following surveillance test activities for review. Activities were selected based upon risk significance and the potential risk impact from an unidentified deficiency or performance degradation that a system, structure, or component could impose on the unit if the condition were left unresolved:

- Unit 1 RCIC vacuum discharge isolation valve local leak rate test for 1E51-F069 and 1E51-F028;
- Unit 2 ‘B’ EDG 24-hour run;
- Unit 2 SCRAM insertion time testing; and
- ‘A’ diesel fire pump flow testing.

The inspectors observed the performance of surveillance testing activities, including reviews for preconditioning, integration of testing activities, applicability of acceptance criteria, test equipment calibration and control, procedural use, control of temporary modifications or jumpers required for test performance, documentation of test data, Technical Specification applicability, impact of testing relative to performance indicator reporting, and evaluation of test data.

The inspectors’ review of these activities constituted four inspection samples.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors reviewed a full scale emergency preparedness drill to evaluate drill conduct and the adequacy of the licensee’s critique of performance to identify weaknesses and deficiencies. The selected drill provided input to the Drill/Exercise NRC Performance Indicator. The inspectors observed the classification of events, notifications to off-site agencies, protective action recommendation development, and drill critique. Observations were compared to the licensee’s critique observations and corrective action program entries. The inspectors verified that there were no discrepancies between observed performance and performance indicator reported statistics. The simulator scenario observed resulted in alert, site area emergency, and general emergency classifications.

This emergency preparedness drill observation constituted a single inspection sample.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Public Radiation Safety

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

.1 Inspection Planning

a. Inspection Scope

The inspectors reviewed the most current Radiological Effluent Release Report to verify that the program was implemented as described in Radiological Effluent Technical Specification/Offsite Dose Calculation Manual (RETS/ODCM) and to determine if Offsite Dose Calculation Manual (ODCM) changes were made in accordance with Regulatory Guide 1.109 and NUREG-0133. The inspectors determined if the modifications made to radioactive waste system design and operation changed the dose consequence to the public. The inspectors verified that technical and/or 10 CFR 50.59 reviews were performed when required and determined whether radioactive liquid and gaseous effluent radiation monitor setpoint calculation methodology changed since completion of the modifications. The inspectors determined if anomalous results reported in the current Radiological Effluent Release Report were adequately resolved.

The inspectors reviewed the RETS/ODCM to identify the effluent radiation monitoring systems and their flow measurement devices, effluent radiological occurrence performance indicator incidents in preparation for onsite follow-up, and the Final Safety Analysis Report (FSAR) description of radioactive waste systems.

This review represented one inspection sample.

b. Findings

No findings of significance were identified.

.2 Onsite Inspection

a. Inspection Scope

The inspectors walked down the major components of the gaseous and liquid release systems (e.g., radiation and flow monitors, demineralizers and filters, tanks, and vessels) to observe current system configuration with respect to the description in the UFSAR, ongoing activities, and equipment material condition.

The inspectors observed the routine processing (including sample collection and analysis) and release of radioactive liquid waste to verify that appropriate treatment equipment was used and that radioactive liquid waste was processed and released in accordance with procedure requirements and observed the sampling and compositing of liquid effluent samples. The licensee did not release liquid effluents during the period from the last inspection to the present. The inspectors observed the routine processing (including sample collection and analysis) and release of radioactive gaseous effluent to verify that appropriate treatment equipment is used and that the radioactive gaseous effluent was processed and released in accordance with RETS/ODCM requirements.

The inspectors reviewed the records of abnormal releases or releases made with inoperable effluent radiation monitors and reviewed the licensee's actions for these releases to ensure an adequate defense-in-depth was maintained against an unmonitored, unanticipated release of radioactive material to the environment.

The inspectors reviewed the licensee’s technical justification for changes made by the licensee to the ODCM as well as to the liquid or gaseous radioactive waste system design, procedures, or operation since the last inspection to determine whether the changes affected the licensee’s ability to maintain effluents As-Low-As-Is-Reasonably-Achievable (ALARA) and whether changes made to monitoring instrumentation resulted in a non-representative monitoring of effluents. The inspectors independently reviewed the Radiological Effluent Release Reports to assess if the licensee’s offsite dose calculations were affected by factors which may have resulted in significant changes. No significant changes were identified.

The inspectors reviewed a selection of monthly, quarterly, and annual dose calculations to ensure that the licensee properly calculated the offsite dose from radiological effluent releases and to determine if any annual RETS/ODCM (i.e., Appendix I to 10 CFR 50) values were exceeded.

The inspectors reviewed air cleaning system surveillance test results to ensure that the system was operating within the licensee’s acceptance criteria. The inspectors reviewed surveillance test results the licensee used to determine the stack and vent flow rates. The inspectors verified that the flow rates were consistent with RETS/ODCM or UFSAR values.

The inspectors reviewed records of instrument calibrations performed since the last inspection for each point of discharge effluent radiation monitor and flow measurement device and reviewed any completed system modifications and the current effluent radiation monitor alarm setpoint value for agreement with RETS/ODCM requirements. The inspectors also reviewed calibration records of radiation measurement (i.e., counting room) instrumentation associated with effluent monitoring and release activities and the quality control records for the radiation measurement instruments.

The inspectors reviewed the results of the interlaboratory comparison program to verify the quality of radioactive effluent sample analyses performed by the licensee. The inspectors reviewed the licensee's quality control evaluation of the interlaboratory comparison test and associated corrective actions for any deficiencies identified. The inspectors reviewed the licensee’s assessment of any identified bias in the sample analysis results and the overall effect on calculated projected doses to members of the public. In addition, the inspectors reviewed the results from the licensee’s quality assurance audits to determine whether the licensee met the requirements of the RETS/ODCM.

These reviews represented eight inspection samples.

b. Findings

No findings of significance were identified.

.3 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed the licensee's self-assessments, audits, Licensee Event Reports, and Special Reports related to the radioactive effluent treatment and monitoring program since the last inspection to determine if identified problems were entered into the corrective action program for resolution. The inspectors also verified that the licensee's self-assessment program was capable of identifying repetitive deficiencies or significant individual deficiencies in problem identification and resolution.

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

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

This review represented one inspection sample.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

.1 Data Submission Issue

a. Inspection Scope

The inspectors performed a review of the data submitted by the licensee for the 2nd Quarter 2005 performance indicators for any obvious inconsistencies prior to its public release in accordance with IMC 0608, “Performance Indicator Program.”

The reviews performed did not constitute in-depth inspections of any of the performance indicators, and as such did not represent any individual inspection samples.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, and Occupational Radiation Safety

.1 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

As part of the various baseline inspection procedures conducted during the period, the inspectors verified that the licensee entered the problems identified during the inspection into their corrective action program. Additionally, the inspectors verified that the licensee was identifying issues at an appropriate threshold and entering them in the corrective action program, and verified that problems included in the licensee's corrective action program were properly addressed for resolution. Attributes reviewed included: the complete and accurate identification of the problem; that timeliness was commensurate with the safety significance; that evaluation and disposition of performance issues, generic implications, common causes, contributing factors, root causes, extent of condition reviews, and previous occurrences reviews were proper and adequate; and that the classification, prioritization, focus, and timeliness of corrective actions were commensurate with safety and sufficient to prevent recurrence of the issue.

b. Findings

One of the findings described elsewhere in this report was related to the cross-cutting area of Problem Identification and Resolution:

- A Green finding and associated NCV described in Section 1R05.3 involved the failure of plant personnel conducting hot work to follow procedural requirements for fire blanket protection in the vicinity of the work site. The improper fire blanket coverage and lack of attentiveness on the part of the assigned fire watch and other licensee personnel responsible for ensuring that ignition controls in the vicinity of hot work were being properly applied were repeat NRC-identified performance deficiencies that had previously resulted in a small Class ‘A’ fire in the 2B RHR corner room in February 2005.

.2 Daily Corrective Action Program (CAP) Reviews

a. Inspection Scope

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

These daily reviews did not constitute any additional inspection samples. Instead, by procedure they were considered part of the inspectors’ daily plant status monitoring activities.

b. Findings

No findings of significance were identified.

.3 Semi-Annual 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 January 2005 through June 2005, although some examples expanded beyond those dates when 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 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 semi-annual trend review did not constitute an additional inspection sample. Instead, by procedure it was considered part of the inspectors' daily plant status monitoring activities.

b. Findings and Issues

No findings of significance were identified. No issues were identified.

.4 Selected Issue Follow-up Inspection: (Closed) Unresolved Item 05000373/2005002-05: Unit 1 Core Standby Cooling System (CSCS) Pump Room Ventilation System Control Cabinet Water Intrusion

Introduction

In early January 2005, licensee engineering and maintenance personnel following up on the failure of a safety-related fan controller (1TIC-VY024) identified that the controller had failed due to electric shorts caused by moisture intrusion into the fan control cabinet (1PL74J). Water was observed to have been dripping from an internal conductor onto a terminal strip inside the cabinet. A junction box (1JB301A) upstream of the electrical conduit entering 1PL74J was opened and inspected and found to have had a considerable amount of accumulated water within it. The water had apparently run down the connecting conduit from the junction box and entered 1PL74J at the conduit penetration. The internal area of 1JB301A showed signs of long-term water intrusion, such as extensive corrosion product buildup.

The inspectors selected the licensee's actions in response to the 1TIC-VY024 failure and discovery of water intrusion into 1PL74J for a more in-depth review. The focus of this inspection was a review of the licensee's extent-of-condition evaluations and corrective actions.

The inspectors' review of this issue constituted a single inspection sample.

a. Effectiveness of Problem Identification

(1) Inspection Scope

The inspectors reviewed the licensee's CAP entries and actions associated with this issue to verify that the identification of the problems by the licensee were complete, accurate, and timely, and that the consideration of extent-of-condition review, generic implications, common cause, and previous occurrences were adequate.

(2) Issues

In general, the licensee's CAP efforts were successful at identifying the apparent cause of the water intrusion and the various underlying causal factors. The licensee's primary CAP product for this issue, Equipment Apparent Cause Evaluation (EACE) 287334, discussed the apparent cause of the problem as being due to water intrusion from the outside via the conduit system that penetrates the building roof. Further, the EACE discussed the fact that the 1PL74J cabinet was not properly sealed against water intrusion, and that the design specifications for the original conduit installations were not entirely adequate.

b. Prioritization and Evaluation of Issues

(1) Inspection Scope

In reviewing the licensee's CAP entries and actions associated with this issue, the inspectors considered the licensee's evaluation and disposition of performance issues, evaluation and disposition of operability issues, and application of risk insights for prioritization of issues.

(2) Findings

Introduction

A finding of very low safety significance (Green) was identified by the inspectors. The inspectors determined that the licensee was slow to fully understand the significance of the water intrusion issue, take effective corrective actions, and, in several cases, had to be prompted by the inspectors before actions were taken. An associated Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," was also identified by the inspectors.

Description

On January 3, 2005, during a rain/snow shower, the control room received a high temperature alarm for the Unit 1 Division 2 CSCS pump room ventilation system (VY). The room temperature controller, 1TIC-VY024, was indicating 120 degrees Fahrenheit. Actual room temperature was verified to be 73 degrees. The erratic behavior of this temperature controller resulted in the potential loss of temperature control for the Division 2 CSCS pump room and, consequently, the inoperability of the 'C' and 'D' residual heat removal service water (RHRSW) pumps and the 'B' spent fuel pool cooling (FC) emergency makeup pump.

Subsequent to the January 3, 2005, adverse weather, the 1TIC-VY024 controller was replaced with a new controller, but still continued to exhibit erratic behavior. The original controller was reinstalled and it was noted during troubleshooting that water dripping into the control panel from an internal conductor onto a terminal strip was causing stray currents which resulted in the erratic behavior of this controller. Further investigation revealed water intrusion inside the conduit connected to this control panel. Corrosion and standing water were also located in junction box 1JB301A associated with this conduit.

On January 5, 2005, repairs were performed to clean and dry the Division 2 VY conduit system and clean, paint, and drill weep holes in junction box 1JB301A. An extent-of-condition walkdown by the licensee noted that the Division 1 VY conduit and control panel also exhibited signs of water intrusion. Long term rust deposits and water dripping within the control panel were observed. This division was not considered inoperable due to a wiring configuration difference between the Division 1 and Division 2 control panels that directed dripping water within the panel away from the terminal strip. This wiring practice was commonly termed as "installing drip loops" and was required for cable terminations of this type per licensee maintenance procedures.

On January 9, 2005, the licensee made repairs to the Division 1 VY conduits in an attempt to stop the water intrusion by sealing the conduit. On February 15, 2005, during a rain shower, inspectors in the plant identified water dripping from weep holes in the Division 1 VY junction boxes from conduit that had supposedly been sealed several days earlier to prevent such water intrusion.

On March 25, 2005, during a rain shower, inspectors again identified water dripping from junction boxes on both Division 1 and Division 2 VY conduits that had previously been repaired for similar water intrusion events. The inspectors' concerns over this issue resulted in several subsequent meetings and discussions with licensee engineering, maintenance, and regulatory affairs personnel. From these discussions, the licensee agreed to accelerate planned extent-of-condition inspections associated with the issue.

On April 13, 2005, licensee electrical maintenance personnel conducting extent-of-condition inspections of similar electrical junction boxes in other plant systems identified a junction box (2TZ-VD013C) in the emergency diesel generator ventilation system (VD) that showed signs of past water intrusion. Although this was a safety-related component having been discovered in a potentially degraded condition, no entry into the licensee's CAP was made nor were on-watch control room operations personnel informed. After several days, the inspectors, who were following the licensee's extent-of-condition inspections, noted that no CAP entry related to this issue had been made and challenged the licensee's electrical maintenance manager on the issue. Following this challenge, licensee personnel placed the issue into the CAP (IR 327468) on April 22, 2005.

On April 28, 2005, the inspectors met with the licensee's senior leadership team, including the Site Vice-President, and raised the following concerns:

- Conduit penetrations on the building roof where leakage was occurring had supposedly been sealed by maintenance, yet the inspectors continued to find evidence of water intrusion into the system during rainy days;
- Some maintenance and engineering personnel that the inspectors had been engaging on the issue were seemingly still involved in a debate over whether the issue represented a maintenance problem, or whether the issue represented an engineering/design problem, instead of being focused on the degraded condition and its resolution;
- Certain key follow-up actions were not scheduled to be completed in the licensee's response plan for several months (e.g., the extent-of-condition determination for all junction boxes susceptible to water intrusion was not scheduled to be completed until the end of October 2005, etc.); and
- As evidenced by the need to prompt the licensee to create a CAP entry for degraded condition found with junction box 2TZ-VD013C on April 13, 2005, the inspectors were seemingly being required to become increasingly involved in driving the corrective actions for the issue to completion.

Following the meeting with the site leadership team, the inspectors noted improvement in the timeliness of the evaluation of issues associated with the degraded condition and in the prioritization given to response actions by the licensee's staff.

Analysis

In reviewing the above time line for the issue, the inspectors determined that there was a performance deficiency associated with the corrective actions taken by the licensee. Specifically, as evidenced by the concerns the inspectors raised with the site's leadership team on April 28, 2005, the inspectors determined that the licensee had not given proper priority to the issue and the actions needed to resolve it. The systems involved, VY and VD, were classified as safety-related and subject to the requirements of 10 CFR 50, Appendix B, yet deficiencies and nonconformances associated with various components within these systems were not being promptly identified or corrected.

The objective of the Mitigating Systems Cornerstone of Reactor Safety is "to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage)." In accordance with NRC Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," the inspectors determined that the finding was of more than minor significance in that it had a direct impact on this cornerstone objective. Specifically, the inspectors concluded that the licensee's failure to promptly identify and correct the water intrusion deficiencies and nonconformances caused an unwarranted reduction in the reliability of components within the VY and VD systems.

The inspectors determined that the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," and conducted a Phase 1 characterization and initial screening. Because the finding did not represent the loss of any safety function for any system or train, and because it was determined not to be potentially significant with respect to any external events such as seismic, flooding, tornado, etc., the inspectors determined it to be of very low safety significance (Green) and within the licensee's response band.

Enforcement

Criterion XVI of 10 CFR 50, Appendix B, states, in part, that: "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected." Contrary to this requirement, the licensee failed to promptly identify and correct nonconforming conditions associated with water from meteorological precipitation intruding into safety-related electrical junction boxes and control cabinets from January 2005 through April 2005. The licensee has entered multiple items associated with this event into their corrective action program (IRs 287742, 287334, 287987, 287351, 287694, 288823, 308000, 301768, and 317267). Among the actions the licensee has performed, or plans to perform, to address this issue include: a complete extent-of-condition review of all through roof conduits that may be susceptible to water intrusion; drilling of weep holes in all susceptible junction boxes; repairs to damage caused by water intrusion; and the sealing of the leaking conduit on Unit 1 Division 1 and Division 2 VY systems. Because the licensee has entered the issue into their corrective action program and the finding is of very low safety significance, this violation of 10 CFR 50, Appendix B, Criterion XVI, is being treated as a NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000373/2005003-03; 05000374/2005003-03)

c. Effectiveness of Corrective Actions

(1) Inspection Scope

The inspectors reviewed multiple related CAP documents associated with the moisture intrusion discovered entering into various safety-related electrical ventilation fan control cabinets and electrical junction boxes. The intent of this review was to determine if the CAP actions addressed generic implications, and to verify that corrective actions were appropriately focused to correct the problem.

(2) Issues

As discussed in the finding documented in the previous section, the licensee's initial corrective actions to seal the conduit roof penetrations were not entirely effective. Inspectors conducting follow-up inspections during rainy days following the licensee's early maintenance and repair activities were able to identify additional signs of water intrusion into electrical junction boxes via conduit penetrating the roof of the building. In discussions with licensee engineering personnel following these inspections, the licensee admitted that some of the sealing work had been performed in cold/wet weather conditions not necessarily conducive to success. Planned sealing activities during the summer months were expected to be more successful.

.5 Selected Issue Follow-up Inspection: Status of Human Performance Cross-Cutting Issue Corrective Actions and Comprehensive Improvement Program

Introduction

During the mid-cycle assessment for the 2004 calendar year inspection program, the NRC staff identified a substantive cross-cutting issue in the area of human performance. The results of this assessment were provided to the licensee on August 30, 2004, in the LaSalle Mid-Cycle Performance Review letter (ADAMS Accession No. ML042450524).

Over the course of the assessment period, the inspectors identified 12 findings/violations of very low safety significance (Green) where human performance was not adequate. The breakdown by cornerstone for these findings/violations was as follows:

- Initiating Events: 2 findings/violations;
- Mitigating Systems: 4 findings/violations; and
- Occupational Radiation Safety: 6 findings/violations.

Specifically, the findings/violations were attributed to inadequate human performance in manipulation of plant equipment outside of the normal work control processes and without permission from the control room, failing to comply with procedural requirements, failure to comply with contaminated and high radiation area posting requirements, and failure to post radiation or high radiation signs on scaffolding platforms. In addition to the 12 findings/violations, another 18 additional minor issues were identified with human performance as the primary or contributing cause. These minor issues by cornerstone were as follows:

- Initiating Events: 5 minor issues;
- Mitigating Systems: 9 minor issues; and
- Barrier Integrity: 4 minor issues.

During the end-of-cycle assessment for the 2004 calendar year inspection program, the NRC staff determined that the substantive cross-cutting issue in the area of human performance continued as a regulatory concern. The results of this assessment were provided to the licensee on March 2, 2005, in the LaSalle County Station Annual Assessment Letter (NRC Inspection Report No. 05000373/2005001; 05000374/2005001 – ADAMS Accession No. ML050620430).

For the end-of-cycle assessment period, the inspectors identified 10 findings/violations of very low safety significance (Green) where human performance was not adequate. Of these 10 findings/violations, 7 were previously evaluated as part of the 12 findings/violations considered during the mid-cycle assessment, and 3 were new. The breakdown of these findings/violations by cornerstone was as follows:

- Mitigating Systems: 2 findings/violations;
- Barrier Integrity: 1 finding/violation; and
- Occupational Radiation Safety: 7 findings/violations.

At the end of the 2004 calendar year inspection program, the NRC staff determined that the effectiveness of the licensee's corrective actions for the human performance substantive cross-cutting issue was indeterminate, as evidenced by the continued occurrence of human performance events/issues at the station. The licensee continued to revise and enhance their human performance improvement plan to address the issue.

The inspectors' review of this issue constituted a single inspection sample.

a. <u>Prioritization and Evaluation of Issues</u>

(1) <u>Inspection Scope</u>

In reviewing the licensee's comprehensive human performance improvement plan and related documents, the inspectors considered the evaluation and disposition of performance issues, evaluation and disposition of operability issues, and application of risk insights for prioritization of issues. The specific focus for the inspectors' review was the time period from June 30, 2004, through June 30, 2005.

(2) Issues

The inspectors found that the licensee had given an appropriately high priority to the actions intended to address the substantive cross-cutting issue in human performance. Individual gap analyses for station departments were rolled up into a single comprehensive action plan intended to address four different areas of human performance: planning, execution, results, and processes. The licensee's comprehensive plan was provided with routine and regular updates as new CAP data became available, and a new human performance coordinator was assigned to the station.

In response to the continuance of the substantive cross-cutting issue in human performance in the 2004 LaSalle End-of-Cycle Assessment letter, the licensee established a new human performance management champion/sponsor and increased the human performance steering committee meetings from quarterly to monthly to provide enhanced management oversight for the improvement process.

b. Effectiveness of Corrective Actions

(1) Inspection Scope

The inspectors reviewed the licensee's comprehensive human performance improvement plan and related documents in detail, with the intent of determining whether or not the CAP actions addressed generic implications, and to verify that corrective actions were appropriately focused to correct the human performance problems. The specific focus for the inspectors' review was the time period from June 30, 2004, through June 30, 2005.

(2) Issues

For the focus period noted above, the inspectors identified 8 findings/violations of very low safety significance (Green) where human performance was not adequate, plus a single unresolved item with human performance attributes that had been transferred to the NRC Office of Investigations for additional review. The breakdown of these findings/violations and URI by cornerstone was as follows:

- Initiating Events: 2 items;
- Mitigating Systems: 2 items;
- Barrier Integrity: 2 items; and
- Occupational Radiation Safety: 3 items.

In addition to the 9 items above that met the threshold for being documented in an inspection report, another 11 minor issues were identified with human performance as the primary or contributing cause for the focus period. These minor issues by cornerstone were as follows:

- Initiating Events: 3 minor issues;
- Mitigating Systems: 5 minor issues;
- Barrier Integrity: 1 minor issue;

- Occupational Radiation Safety: 1 minor issue; and
- Physical Protection: 1 minor issue.

The inspectors next analyzed the data for the focus period with respect to event dates in an effort to identify whether or not the trend in human performance issues was declining, improving, or steady. For the 8 findings/violations and single URI identified during the focus period, the following breakdown was noted:

- 2004 and Pre-Refuel Outage Items: 4 findings/violations;
- January – February 2005 Items (L2R10 Refuel Outage): 3 findings/violations plus 1 URI; and
- March – June 2005 Items: 1 finding/violation.

For the 11 minor issues that were identified with human performance as the primary or contributing cause for the focus period, these broke down as follows:

- 2004 and Pre-Refuel Outage Items: 6 minor issues;
- January – February 2005 Items (L2R10 Refuel Outage): 5 minor issues; and
- March – June 2005 Items: 0 minor issues.

With a significant number of the licensee's human performance issues for the focus period year coinciding with the Unit 2 2005 (L2R10) refueling outage, the inspectors concluded that the licensee's corrective actions intended to specifically address those human performance problems associated with the high-tempo pace of refueling operations were indeterminate at this point. This conclusion was also based upon the fact that the L2R10 refuel outage has been the only refuel outage the licensee has experienced since the substantive cross-cutting issue in human performance was identified in the 2004 LaSalle Mid-Cycle Assessment letter, and essentially provided the inspectors with only a single data point for analysis.

In addition, the inspectors also concluded that, although the licensee still met the criteria for a substantive human performance cross-cutting issue at the end of the focus period, as discussed in NRC IMC 0305, Section 06.06i, "Substantive Cross-Cutting Issues," the more recent data indicated improvement in licensee non-refuel outage human performance. During the several months of non-refuel outage time prior to L2R10, the licensee accumulated 4 findings/violations with human performance cross-cutting aspects and 6 more minor issues. However, during a similar non-refuel outage period following L2R10 there was only a single human performance finding/violation, and no minor issues.

4OA3 Event Follow-up (71153)

Cornerstones: Mitigating Systems and Barrier Integrity

.1 (Closed) Licensee Event Report (LER) 05000374/2005-002-00: Pressure Boundary Leakage Discovered in 2D MSIV Drain Line Weld During Refueling Outage VT-2 Examination.

On March 12, 2005, during a scheduled refueling outage on Unit 2, a pinhole leak in an American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Class 1 weld on the outboard Main Steam Isolation Valve (MSIV), 2B21-F028D, drain line was discovered during a hydrostatic test of the reactor coolant pressure boundary. The leakage source was the weld joint on the 2 inch drain line between the body of the MSIV (2B21-F028D) and the body of the drain valve (2B21-F067D). A subsequent investigation by the licensee's engineering staff determined that the apparent cause of the leak was a weld inclusion or defect from a Class 1 weld completed in 1995.

Corrective actions taken by the licensee included: weld repair and inspection using liquid dye penetrant testing; successful re-performance of the hydrostatic test; and a commitment to perform another weld examination during the next Unit 2 refueling outage.

Because this leak was discovered during hydrostatic testing while Unit 2 was shutdown and there were no signs of corrosion or evidence of a leak during the previous hydrostatic test in 2003, the leak was determined to have first occurred during the March 2005 hydrostatic test and not during critical Unit 2 reactor operations. As a result, no findings of significance or violations of regulatory requirements were identified by the inspectors.

The review and closure of this LER constituted a single inspection sample.

.2 Loss of Safety-Related 480 Vac Buses 235X and 235Y (ENS 41787)

a. Inspection Scope

At approximately 11:40 p.m. on June 21, 2005, the 4160 Vac feeder breaker to 480 Vac safety-related Division 1 Buses 235X and 235Y tripped due to a faulty breaker relay. The loss of buses 235X and 235Y resulted in multiple equipment challenges for the on-watch Operations crew.

Upon notification of the discovery and subsequent entry into a Technical Specification 3.0.3 Action Statement potentially leading to the shutdown of Unit 2, inspectors responded to the plant to monitor the licensee's actions. The inspectors observed plant parameters and status; evaluated the performance of plant systems and licensee actions; and confirmed that the licensee properly reported the event as required by 10 CFR 50.72. The inspectors also verified that no human performance errors complicated the event response.

Assisted by NRC Region III managers and staff, the inspectors screened the event in accordance with NRC Management Directive 8.3, "NRC Incident Investigation Program," to determine whether or not the additional resources of an NRC inspection team were warranted. The inspectors, in conjunction with the regional staff, determined that formation of an NRC Special Inspection Team in response to the event was not required.

The inspectors' response to this event constituted a single inspection sample.

b. Findings

No findings of significance were identified.

4OA4 Cross-Cutting Aspects of Findings

Cornerstone: Initiating Events

Human Performance

One of the findings described elsewhere in this report had human performance deficiencies as its major causal elements.

- A Green finding and associated NCV described in Section 1R14.2 involved the failure of operating crew personnel to note and correct a Unit 1 reactor overpower condition in a timely manner. Because they had become distracted by ongoing work activities in the control room, Unit 1 operators failed to realize that reactor power had risen to approximately 103.17 percent until about eight minutes after the power excursion had taken place.

This human performance deficiency was related to operator crew performance and standards.

4OA5 Other

Cornerstones: Initiating Events and Mitigating Systems

.1 (Closed) Unresolved Item 05000373/2004005-04; 05000374/2004005-04: Standby Liquid Control (SBLC) Boron Tank Volume/Concentration Measurements

a. Inspection Scope

An Unresolved Item (URI) was previously opened to track the licensee's handling of standby liquid control system tank volume and concentration issues and the associated utilization of measuring and test equipment to maintain adequate tank volumes and concentrations.

The inspectors reviewed methods for measuring tank level and concentration for adequacy and accuracy in determining Technical Specification required values.

Instruments, measuring and test equipment, and test controls utilized for these measurements were inspected as well as engineering evaluations and condition reports associated with the SBLC system.

The closing of this URI was the continuation of an inspection item from a previous report, and as such did not represent a stand-alone inspection sample.

b. Findings

Introduction

The inspectors identified a finding of very low safety significance (Green) and an associated NCV during a review of test procedures used to maintain SBLC tank volume and concentration within Technical Specification limits. The inspectors' review identified that the licensee had routinely maintained volume and concentration very near Technical Specification limits, and had performed level measurements using a test device with a non-conservative bias. This failure to use adequate test equipment for a safety-related Technical Specification measurement was determined by the inspectors to be contrary to the requirements of 10 CFR 50, Appendix B, Criterion XI, "Test Control."

Description

Control room indication of SBLC boron solution tank volume was provided using tank level bubbler tubes. However, the licensee had experienced chronic problems with these tubes becoming clogged with sodium pentaborate crystals, which resulted in erroneous tank alarms on high volume. As a result, verification of Technical Specification values for SBLC tank volume using local level measurements at the tank became the common licensee practice. In the mid-1980's, this measurement was performed using a plumb bob measuring device dropped into the tank from an access on the top. However, due to problems with the weighted end of the plumb bob becoming separated from the measuring tape and being lost in the SBLC tanks, in July 1997, the licensee fabricated a T-square device to use at the access on the top of the tank to obtain level for tank volume surveillances.

On November 23, 2004, plant operators experienced difficulty determining whether or not Unit 1 SBLC volume and concentration were within required Technical Specification limits for a period of approximately ten hours. An on-watch control room operator initially plotted the values for sodium pentaborate solution concentration and volume using control room indications for tank volume and determined that Technical Specification requirements were not being met. Although subsequent review by the licensee's staff determined that the Unit 1 SBLC solution tank met Technical Specification requirements for solution concentration and volume, an extent-of-condition review determined that both Unit 1 and Unit 2 SBLC tanks had routinely been maintained with little margin to the Technical Specification Figure 3.1.7-1 limits for volume and sodium pentaborate solution concentration. Following up on this issue, the inspectors identified that the T-square instrument being used by the licensee as their most accurate means of determining SBLC tank level, and thus volume, had not been manufactured in accordance with the licensee's 10 CFR 50, Appendix B, Quality

Assurance Program requirements. Further review by the licensee's engineering staff determined that the graduations on each T-square unknowingly incorporated a nonconservative bias into each instrument, which combined with the licensee's historical practice of maintaining SBLC tank volume and solution concentration at the lower bounds of the Technical Specification limit created several historical instances where the ability to determine whether or not solution concentration and volume were within Technical Specification limits was dubious at best. Specifically, it was determined that measurements taken with the Unit 1 T-square resulted in volumes that were nonconservative by approximately 34.6 gallons, and that measurements taken with the Unit 2 T-square resulted in volumes that were nonconservative by approximately 19.8 gallons.

Analysis

The inspector-identified performance deficiency associated with this issue was a failure by the licensee's staff to utilize adequate test equipment for the performance of safety-related Technical Specification surveillance measurements of SBLC solution tank level.

The objective of the Mitigating Systems Cornerstone of Reactor Safety is to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences such as core damage. In accordance with NRC IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," the inspectors determined that the finding was of more than minor significance in that it had a direct impact on this cornerstone objective. Specifically, the licensee's manufacture and use of instrumentation with a nonconservative bias, combined with the long-term practice of maintaining SBLC tank concentration very near the lower Technical Specification limit for volume and sodium pentaborate concentration, adversely impacted the licensee's ability to ensure the availability and reliability of the SBLC system.

The inspectors determined that the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," and conducted a Phase 1 characterization and initial screening. The finding was determined to be of very low safety significance because subsequent licensee analyses of SBLC tank concentrations and volumes, in accordance with GL 91-18, demonstrated that the errors in SBLC tank volume in question were sufficiently small as to not have jeopardized the capability of SBLC to have performed its safety function for either unit.

Enforcement

Table 3.2-1 of the licensee's Updated Final Safety Analysis Report (UFSAR) indicated that the SBLC system is subject to the requirements of 10 CFR 50, Appendix B. Criterion XI, "Test Control," of this appendix states, in part, that: "Test procedures shall include provisions for assuring that all prerequisites for the given test have been met, that adequate test instrumentation is available and used, and that the test is performed under suitable environmental conditions." Contrary to this requirement, the licensee failed to fabricate and utilize adequate instrumentation for the performance of safety-related Technical Specification level measurements of each unit's SBLC sodium pentaborate solution tank.

The licensee had entered this issue into their corrective action program as IRs 276755, 276839, 277113, 277439, 281247, and 281238. Corrective actions by the licensee included: additions of sodium pentaborate chemical to each unit's SBLC tank to adjust chemistry to well within the Technical Specification required band; revision of SBLC tank sampling procedures, and the establishment of administrative controls to ensure that each unit's SBLC tank volume and sodium pentaborate solution concentration were being maintained well away from Technical Specification limits; and the procurement of new T-square instruments for measuring SBLC tank level, which are manufactured in accordance with 10 CFR 50, Appendix B, Quality Assurance Program controls and requirements. Because the licensee has entered the issue into their corrective action program and the finding is of very low safety significance, this violation of 10 CFR 50, Appendix B, Criterion XI is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000373/2005003-04; 05000374/2005003-04)

.2 Operational Readiness of Offsite Power (Temporary Instruction 2515/163)

The objective of Temporary Instruction (TI) 2515/163, "Operational Readiness of Offsite Power," was to confirm, through inspections and interviews, the operational readiness of offsite power (OSP) systems in accordance with NRC requirements. The inspectors reviewed licensee procedures and discuss the attributes identified in TI 2515/163 with licensee personnel. In accordance with the requirements of TI 2515/163, inspectors evaluated licensee procedures against the attributes discussed below.

The operating procedures that the control room operator uses to assure the operability of the OSP have the following attributes:

(1) Identify the required control room operator actions to take when notified by the transmission system operator (TSO) that post-trip voltage of the OSP at the plant will not be acceptable to assure the continued operation of the safety-related loads without transferring to the onsite power supply.

(2) Identify the compensatory actions the control room operator is required to perform if the TSO is not able to predict the post-trip voltage at the plant for the current grid conditions.

(3) Identify the notifications required by 10 CFR 50.72 for an inoperable offsite power system when the nuclear station is either informed by its TSO or when an actual degraded voltage condition is identified.

The procedures to ensure compliance with 10 CFR 50.65(a)(4) have the following attributes:

(1) Direct the plant staff to perform grid reliability evaluations as part of the required maintenance risk assessment before taking a risk-significant piece of equipment out-of-service to do maintenance activities.

(2) Direct the plant staff to ensure that the current status of the OSP system has been included in the risk management actions and compensatory actions to reduce the risk when performing risk-significant maintenance activities or when loss of off-site power (LOOP) or station blackout (SBO) mitigating equipment are taken out-of-service.

(3) Direct the control room staff to address degrading grid conditions that may emerge during a maintenance activity.

(4) Direct the plant staff to notify the TSO of risk changes that emerge during ongoing maintenance at the nuclear power plant.

The procedures to ensure compliance with 10 CFR 50.63 have the following attribute:

(1) Direct the control room operators on the steps to be taken to try to recover offsite power within the SBO coping time.

The results of the inspectors' review were forwarded to office of Nuclear Reactor Regulation for further review and evaluation.

The completion of this TI represented one inspection sample.

4OA6 Meetings

.1 Exit Meeting

The inspectors presented the inspection results to the Site Vice-President, Ms. Susan Landahl, and other members of licensee management on July 13, 2005. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

.2 Interim Exit Meetings

Interim exits were conducted for:

- A Biennial Maintenance Effectiveness Periodic Evaluation with the Plant Manager, Mr. D. Enright, on May 6, 2005; and
- A periodic Radioactive Gaseous And Liquid Effluent Treatment And Monitoring Systems inspection with the Duty Plant Manager, Mr. D. Rhoades, on June 17, 2005.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

S. Landahl, Site Vice-President
D. Enright, Plant Manager
R. Chrzanowski, Chemistry Manager
T. Connor, Maintenance Director
L. Coyle, Operations Director
D. Czufin, Site Engineering Director
A. Ferko, Nuclear Oversight Manager
F. Gogliotti, System Engineering Manager
B. Kapellas, Radiation Protection Manager
J. Rappeport, Nuclear Oversight
M. Sharma, Site Maintenance Rule Coordinator
T. Simpkin, Regulatory Assurance Manager
C. Wilson, Station Security Manager

Nuclear Regulatory Commission

B. Burgess, Chief, Reactor Projects Branch 2

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000373/2005003-01	NCV	Failure to Properly Implement Fire Protection Procedure Requirements for Hot Work and Ignition Control Issues (Sections 1R05.3 and 4OA2.1)
05000373/2005003-02	NCV	Operators Fail to Note and Respond to Unit 1 Overpower Condition in a Timely Manner (Sections 1R14.2 and 4OA4)
05000373/2005003-03; 05000374/2005003-03	NCV	Ineffective Corrective Actions for Water Intrusion into Safety-Related Fan Control Cabinets (Section 4OA2.4)
05000373/2005003-04; 05000374/2005003-04	NCV	Nonconservative Uncorrected Bias Associated with Tank Level Instruments Used for Standby Liquid Control System Surveillances (Section 4OA5.1)

Closed

05000373/2005003-01	NCV	Failure to Properly Implement Fire Protection Procedure Requirements for Hot Work and Ignition Control Issues (Sections 1R05.3 and 4OA2.1)
05000373/2005002-11	URI	Unit 1 Reactor Power Excursion to 103.17 Percent (Section 1R14.2)
05000373/2005003-02	NCV	Operators Fail to Note and Respond to Unit 1 Overpower Condition in a Timely Manner (Sections 1R14.2 and 4OA4)
05000373/2005002-05	URI	Unit 1 CSCS Pump Room Ventilation System Control Cabinet Water Intrusion (Section 4OA2.4)
05000373/2005003-03; 05000374/2005003-03	NCV	Ineffective Corrective Actions for Water Intrusion into Safety-Related Fan Control Cabinets (Section 4OA2.4)
05000374/2005-002-00	LER	Pressure Boundary Leakage Discovered in 2D MSIV Drain Line Weld During Refueling Outage VT-2 Examination (Section 4OA3.1)
05000373/2004005-04; 05000374/2004005-04	URI	SBLC Tank Level and Boron Solution Concentration Measurement Issues (Section 4OA5.1)
05000373/2005003-04; 05000374/2005003-04	NCV	Nonconservative Uncorrected Bias Associated with Tank Level Instruments Used for Standby Liquid Control System Surveillances (Section 4OA5.1)

Discussed

None.

LIST OF DOCUMENTS REVIEWED

1R01 Adverse Weather

Procedures:
- EN-LA-402-0005; Extreme Heat Implementation Plan-LaSalle; Revision 5

- WC-AA-107; Seasonal Readiness, Attachment 3 for Reactor Building Closed Cooling Water System; Revision 1

- LaSalle Summer Readiness Duty Team Guide; 05/15/2005

- LaSalle System Condition Report Listing for RBCCW; 05/24/2005

- Open Work Order Listing for LaSalle Unit 2 RBCCW; 05/24/2005

1R04 Equipment Alignment

Issue Reports:
- 272646; Minor Head Gasket Leak - East Side Upper Right; 11/11/2004
- 328045; Fuel Leaking from Piping Flange/Inlet to Tank; 4/23/2005
- 338525; NRC Identified Concerns on Plant Walkdown; 5/25/2005

Procedures:
- LOP-DG-01E; Unit 1 Diesel Generator 1A Electrical Checklist; Revision 7
- LOP-DG -01M; Unit 1 A Diesel Generator Mechanical Checklist; Revision 9
- LOP-DG-03E; Unit 0 Diesel Generator Electrical Checklist; Revision 9
- LOP-DG-03M; Unit 0 Diesel Generator Mechanical Checklist; Revision 8
- LOP-DG -04E; Unit 2 A Diesel Generator Electrical Checklist; Revision 9
- LOP-DG -04M; Unit 2 A Diesel Generator Mechanical Checklist; Revision 8
- LOP-DG-08E; Unit 0 Diesel Generator Cooling System Electrical Checklist; Revision 8
- LOP-DG-08M; Unit 0 Diesel Generator Cooling System Mechanical Checklist; Revision 18
- LOP-RI-01E; Unit 1 Reactor Core Isolation Cooling System Electrical Checklist; Revision 11
- LOP-RI-01M; Unit 1 Reactor Core Isolation Cooling System Mechanical Checklist; Revision 17

Work Orders:
- 509528-01; Minor Head Gasket Leak - Middle East Side of Engine; 11/12/2002

1R05 Fire Protection

Fire Drill Scenario No. 54; 687’ Turbine Building Cable Riser Fire; 3/5/2005

Fire Protection System Drawing, Figure 9.5-1; Revision 3:
- Sheet 27 of 41; Unit 1 Turbine Building Upper Basement, Elevation 687'
- Sheet 28 of 41; Unit 2 Turbine Building Upper Basement, Elevation 687'

Issue Reports:
- 330694; Failed SCBA During Fire Drill; 5/1/2005
- 330720; Reactor Building Elevator Keys; 5/1/2005
- 331281; Fire Brigade Radios; 5/3/2005
- 332429; NRC ID'D Issues with Hot Work During the 1E51-F028 Vlv. Work; 5/5/2005
- 332239; NOS IDs Enhancement for Fire Watch Requirements; 5/5/2005
- 336688; NRC Identified-Miscellaneous NRC Walkdown Issues; 5/19/2005

LaSalle County Station - Fire Protection Reports (FPR):
- Section H.3.7.4 HPCS Diesel Generator Room - Fire Zone 7B1
- Section H.3.7.7 HPCS Diesel Day Tank Room - Fire Zone 7B4
- Section H.3.5.29 Unit 1 HPCS Switchgear Area - Fire Zone 5D1
- Section H.3.2.2 Elevation 820 feet-6 inches - Fire Zone 2B1
- Section H.3.2.3 Elevation 820 feet-6 inches - Fire Zone 2B2
- Section H.3.3.2 Elevation 820 feet-6 inches - Fire Zone 3B1

LaSalle County Station Final Safety Analysis Report:
- Figure 9.5-1; Fire Protection System; Sheets 5,6, 22 and 29; Revision 3

Procedures:
- OP-MW-201-004; Fire Prevention for Hot Work; Revision 0

1R06 Flood Protection Measures

Drawings:
- 1E-1-4022AB; Schematic Diagram Circulating Water Pump 1B System "CW" Part 2; Revision S

Issue Reports:
- 170329; SSPV Condenser Pit Level Switches, Change Trip Check Frequency; 12/12/2003
- 267942; NRC Identified Enhancements to LOA-FLD-001(FLOODING); 10/28/2004;
- 334882; NRC Questioned Testing Method; 5/13/2005

LaSalle County Generating Station Probabilistic Risk Analysis; Revision 4

LaSalle Operator Requalification Program:
- S-04-4-3; LOA'S; 6/09/2004
- 03C3-03; RPV Ref Leg Failure/LPRM Failure/SDV Hi Level Alarm/Loss MET Tower/Flooding; 05/30/2003
- SEG-02C2-02; Loss of UHS; 04/29/02
- SEG-03C1-02; PR/LOA Scenario; 01/26/2003
- 03C2-06; Smoke in the RB Raceway, 3-Earthquakes Resulting the Following: 1A TDRFP Trip, Flooding in the RB NE Corner Room, L0108 Breaker Trip, Loss of 143-1, Flooding in the RB Raceway, 1B HD Pump Trip, ATWS Blowdown; 01/10/2003

- SEG-05C3-04 (Draft) Flooding Scenario; 5/26/2005

Plant History on Functional Testing of Condenser Pit Level Switches (1LS-CW031 & 32; 2LS-CW31 &32)

Procedures:
- LES-LS-01; Inspection of Magnetrols and Capacity Check for Sumps in Flood Control Zone and Other Related Sumps; Revision 13
- LOA-FLD-001; Flooding; Revision 6
- LOA-DIKE-001; Lake Dike Damage/Failure; Revision 5
- LTS-1000-29; Watertight Door and Penetration Inspection; Revision 11

Updated Final Safety Analysis Report; Revision 15:
- Section 2.0; Site Characteristics
- Section 3.4; Water Level (Flood) Design
- Section 3.9; Mechanical Systems and Components
- Section 3.11.1.4; Evaluation for Flooding and Submergence
- Section 7.7.15.2.8; Sump Monitoring System
- Section 15.6.6; Feedwater Line Break
- Amendment 37 Questions and Answers to Question 212.92; Information Related to Pipe Breaks or Leaks in High or Moderate Energy Lines Outside Containment
- Amendment 35 Questions and Answers to Question 212.85; Alarms to Operators of ECCS Failures Outside Containment

Work Orders:
- 633282-01; IM 2LS-CW031 Cond Pit Level; 2/17/2005

1R11 Licensed Operator Requalification Program

ESG-66; Dynamic Simulator Scenario Guide; Revision 0

1R12 Maintenance Effectiveness

Issue Reports:
- 202748; 1B EDG Cooling Water Flow Found to Be Below Minimum Required; 2/19/2004
- 207155; 2DC17E, Div II 125V Battery Charger Would Not Pick Up Load; 3/9/2004
- 212995; B RR HPU Subloop 1 Breaker Tripped; 4/4/2004
- 321434; RMCS Tripped During Weekly Control Rod Exercising; 4/5/2005
- 321503; RMCS Trip; 4/6/2005
- 324939; RMCS Tripped; 4/15/2005
- 327468; Corrosion Found in Junction Box 2TZ-VD013C; 4/13/2005
- 328225; U1 RMCS Trip; 4/24/2005
- 328888; Inadequate Detail for Externally Sealing of Conduits; 4/26/2005
- 329929; RMCS Trips; 4/28/2005
- 330740; Multiple RMCS Trips on U-1; 5/1/2005
- 331720; Valve 1E51-F028 Failed LLRT Affecting MR PC Function; 5/4/2005
- 332241; Examination Results 1E51-F028 & 1RI32A; 5/5/2005
- 332672; Material Identified in Pipe 1RI32A; 5/6/2005

- 322203; 1E51-F028 Valve Failed Local Rate Test; 4/7/2005
- 345670; 1E51-F028, Cause of Sticking May Not Be Identified; 6/20/2005
- 346844; Insp Results of 1E51-F028, W/O 00820863-01; 5/5/2005

Work Orders:
- 736949-01; EP Type C Appendix J LLRT - 1E51-F069, 1E51-F028; 4/7/05
- 736949-02; Post Type C Appendix J LLRT - 1E51-F069, 1E51-F028; 5/6/05
- 776182-01; IM Unit 1 RDCS Voltage Monitoring; 5/6/2005

Maintenance Rule Periodic Assessment; July 2002 - June 2004; dated December 2004

Reactor Recirculation System (a)(1) Action Plan; dated April 15, 2005

Maintenance Rule System (a)(1) Action Plans:
- Feedwater (a)(1) Action Plan; dated March 1, 2005
- Fuel Assembly (a)(1) Action Plan; dated February 28, 2005
- Instrument Nitrogen (a)(1) Action Plan; dated September 1, 2002

List of Systems Within the Scope of the Maintenance Rule; dated March 2005

List of Functional Failures for Assessment Period from July 2002 to June 2004; dated July 2004

Maintenance Rule Expert Panel Meeting Minutes:
- September 18, 2002
- February 26, 2003
- July 29, 2003
- October 3, 2003
- June 10, 2004

Quarterly System Health Improvement Program (SHIP) Reports:
- DC Quarterly SHIP Report; March 2005
- EDG Quarterly SHIP Report; March 2005
- RR Quarterly SHIP Report; March 2005
- FW Quarterly SHIP Report; March 2005

Procedures:
- ER-AA-310-1002; Maintenance Rule - SSC Risk Significance Determination; Revision 1
- ER-AA-310-1003; Maintenance Rule - Performance Criteria Selection; Revision 2
- ER-AA-310-1004; Maintenance Rule - Performance Monitoring; Revision 2
- ER-AA-310-1005; Maintenance Rule - Dispositioning Between (a)(1) and (a)(2); Revision 2
- ER-AA-310-1006; Maintenance Rule - Expert Panel Roles and Responsibilities; Revision 2
- ER-AA-310-1007; Maintenance Rule - Periodic (a)(3) Assessment; Revision 3
- MA-AA-716-210; Performance Centered Maintenance (PCM) Process; Revision 3
- SA-1213; PRA Basis – LaSalle Maintenance Rule Reliability Performance Criteria; Revision 0

- SA-1243; LaSalle Maintenance Rule Availability/Reliability Performance Criteria Sensitivity Study – Updated for 2003A LaSalle PRA; Revision 0

Focused Area Self-assessment - LaSalle Maintenance Rule Programs (Action Tracking Item (ATI) 00194261-05); dated December 22, 2004

1R13 Maintenance Risk Assessments and Emergent Work Control

Engineering Changes:
- 354533; Drywell Floor Drain Flow Monitoring Instrumentation; Revision 0

Engineering Change Requests:
- 352281; Re: Information on Delta Between Values from Preferred and Alternate Unidentified Leakage; Revision 0

Issue Reports:
- 301473; Work Performed Ahead of Schedule Causing Inoperability of DC; 2/15/2005
- 325278; Unit 1 RF Drywell Floor Drain Sump Discharge Flow Indication; 4/15/2005
- 325597; 1LE-RF002 1.0 GPM Different Than Alternate Monitor; 4/17/2005
- 327468; Corrosion Found in Junction Box 2TZ-VD013C; 4/13/2005
- 328888; Inadequate Detail for Externally Sealing of Conduits; 4/26/2005
- 329828; NRC-Identified DWFDS FUR Recorder Reads 0.0 GPM with 0.25 GPM Leak; 4/28/2005
- 332241; Examination Results 1E51-F028 & 1RI32A; 5/5/2005
- 332672; Material Identified in Pipe 1RI32A; 5/6/2005
- 340659; Unit 2 Generator Alterex Brush Arcing-Sparking; 6/3/2005

Work Orders:
- 817182-03; Remove and Clean Unit 2 Alterex Brushes; 6/3/2005

Procedures:
- LOS-AA-S101; Unit 1 Shiftly Surveillance; Revision 29

1R14 Personnel Performance During Non-Routine Plant Evolutions and Events

Procedures:
- LOA-AP-201; Unit 2 AC Power System Abnormal; Revision 15
- LOA-VR-201; Unit 2 Recovery From a Group 4 Isolation or Spurious Trip of Reactor Building Ventilation; Revision 4
- LOR-2PM01J-A108; 250 Vdc Battery Trouble; Revision 2
- LOP-VX-01; Switchgear Heat Removal System Startup; Revision 12
- LOA-DC-201; Unit 2 DC Power System Failure; Revision 7
- LOA-PC-201; Primary-Secondary Containment Trouble; Revision 10
- LGA-002; Secondary Containment Control; Revision 3

RA05-25; License Condition 2.F(a) Report: Exceeding License Condition 2.C(1); 3/9/2005

Issue Reports:
- 304613; Controller Failed High; 2/23/2005
- 304789; Failure of 1HK-RR023 Results in Unit 1 Operation Greater Than 100 % RTP; 2/23/2005
- 305612; Evaluation of Operations Crew Performance – RR FCV Failure; 2/25/2005
- 307523; Problem with Re-Flash Function for SPDS Button on PPC; 3/2/2005
- 307654; Evaluate LOA-RR-101(102) for Possible Revision; 3/2/2005
- 307657; U1 PPC Alarm Program May Prevent Audible Alarm; 3/2/2005
- 307659; U2 PPC Alarm Program May Prevent Audible Alarm; 3/2/2005

1R15 Operability Evaluations

Analyses:
- L-003126; Seismic Qualification of 125VDC Distribution Panels with Access Covers Removed; Revision 0

Issue Reports:
- 301473; Work Performed Ahead of Schedule Caused Unplanned Inoperability of Unit 2 Div 1DC and Entry into Unit 1 LCO; 5/5/2005
- 326386; Potential Untested DGCWP Discharge CV Safety Function; 4/19/2005
- 333907; MFLCPR Penalty Based on Non-Conservative GE-14 GEXL CPR; 5/10/2005
- 346231; 2IN043 Has Indication of Leak By; 6/22/2005
- 348827; Passport D030 Panels List Pressures Less Than in Bottles

IST Program Check Valve Condition Monitoring Plan; 4/14/2005

Operability Evaluations:
- 05-003; Primary Containment Isolation - RCIC Steam Line Tunnel Temp Instrumentation; Revision 0
- 05-004; Instrument Nitrogen 2IN043 Check Valve; Revision 0

Risk Assessments:
- SA-1381; Risk Assessment for Missed Technical Specifications Surveillance Requirements for SR 3.0.3 for LaSalle DG Cooling Water Pump Check Valves 0DG002, 1DG002, and 1E22-F028; Revision 0

Operations Standing and Daily Orders:
- S05-008; MFLCPR Administrative Limits; 5/10/2005
- S05-012; Compensatory Actions Until 2IN043 is Repaired; 6/24/2005

1R16 Operator Workarounds

Issue Reports:
- 166562; U-2 Reactor Scram Due to Main Turbine Trip/U-1 Htr Problems; 7/7/2003
- 167166; Post Review of SCRAM per LAP-200-7; 7/11/2003
- 332644; Risk Associated with Operator Workarounds/Challenges; 5/5/2005

List of Operator Workarounds and Challenges; 6/6/2005

Operations Department Aggregate Reviews of Operator Workarounds:
- Fourth Quarter 2004 Review; 12/27/2004
- First Quarter 2005 Review; 3/30/2005

Procedure:
- OP-AA-102-103; Operator Workaround Program; Revision 1

1R19 Post-Maintenance Testing

Issue Reports:
- 332672; Material Identified in Pipe 1RI32A; 5/6/2005
- 332705; High LLRT Leakage Rate on 1E51-F028/69; 5/6/2005
- 341720; Abnormal Noise at Vacuum Pump; 6/7/2005
- 342721; Check Valve Sticking Closed; 6/10/2005
- 343067; U-1 RCIC Vacuum Pump Has Minor Shaft Leak; 6/10/2005
- 343780; Rebuild Spare 1E51-C005 RCIC Barometric Cond Vacuum Pump; 6/19/2005
- 345670; 1E51-F028, Cause of Sticking May Not Be Identified; 6/20/2005
- 346844; Insp Results of 1E51-F028, W/O 00820863-01; 5/5/2005

Procedures:
- LOS-RI-Q5; Reactor Core Isolation Cooling System Pump Operability and Inservice Test in Mode 1,2, and 3; Revision 22
- LOP-SC-02; Standby Operation of the Standby Liquid Control System; Revision 11
- LOS-SC-Q1; SBLC Pump Operability/ Inservice Test and Explosive Valve Continuity Check; Revision 21
- LTS-100-38; RCIC Vacuum Discharge Isolation Valves Local Leak Rate Test 1(2)E51-F069 and 1(2)E51-F028; Revision 11

Work Orders:
- 616298-02; MM Replace Coupling Spider on MDRFP Aux Lube Oil Pump; 5/16/2005
- 736949-01; EP Type C Appendix J LLRT - 1E51-F069, 1E51-F028; 4/7/2005
- 736949-02; Post Type C Appendix J LLRT - 1E51-F069, 1E51-F028; 5/6/2005
- 748385-01; MM Inspect/Repair the Ball Float; 5/6/2005
- 783430-01; OP LOS-SC-Q1 1B SBLC Pump Quarterly Att 1B; 5/19/2005
- 788773-01; OP LOS-RI-Q5 U1 RCIC Cold-Quick Start Att 1A; 6/10/2005
- 791463-01; IM MDRFP Lube Oil Pressure Low; 5/15/2005

1R22 Surveillance Testing

Issue Reports:
- 007845; 1A DG Failure to Start; 3/06/2000
- 285033; Availability Time for DGS; 12/22/2004
- 305723; Record of Inspection Results per W/O 00784521-01; 2/22/2005
- 322203; 1E51-F028 Valve Failed Local Leak Rate Test; 4/7/2005
- 336350; NRC ID: 2B Diesel Generator Availability During LOS-DG-R2B; 5/18/2005

Procedures:
- LES-FP-31; Fire Pump Logic Controller Test; Revision 7
- LOS-DG-M3; 1B(2B) Diesel Generator Operability Test; Revision 61

- LOS-DG-R2B; 2B Diesel Generator Twenty-Four Hour Run Surveillance; Revision 4
- LTS-100-38; RCIC Vacuum Discharge Isolation Valves Local Leak Rate Test 1(2)E51-F069 and 1(2)E51-F028; Revision 11
- LTS-1000-34; Diesel Fire Pump Flow Test; Revision 15
- LTS-1100-4; Scram Insertion Times; Revision 24

Work Orders:
- 642348-01; ES Diesel Fire Pump “A” Flow Test- Section E.1; 6/2/2005
- 736949; EP Type C Appendix J LLRT - 1E51-F069, 1E51-F028; 4/7/2005

<u>1EP6</u> <u>Drill Evaluation</u>

Drill Nuclear Accident Reporting System and Event Notification Forms; 6/16/2005

Exelon Nuclear Emergency Plan; LaSalle Station Annexes; Revision 18

LaSalle 2nd Quarter 2005 Full Scale PI Mini Drill Plan; 6/16/2005

<u>2PS1</u> <u>Radioactive Gaseous And Liquid Effluent Treatment And Monitoring Systems</u>

Analytics Results of Radiochemistry Cross Check Program Exelon Corporation, LaSalle Station; First through Fourth Quarters 2004

Issue Reports:
- 166079; Source Check Values on PINGS Out of Range; 7/2/2003
- 185738; Process Radiation monitoring Program Calculation Deficiencies; 11/10/2003
- 195907; Iodine-131 Detected in Waste Water Treatment Facility Effluent Composite; 1/17/2004
- 220598; WRGM Sample Conditioning Skid Not Operated Per Vendor Design; 5/12/2004
- 228399; Observed Increase in Main Vent Stack WRGM Activity; 6/14/2004
- 294808; Reverse Flow Causing Wide Range Gas Monitor Nuisance Alarms; 1/27/2005
- 297230; Lower Than Expected Stack Release Rates Observed; 2/3/2005
- 318151; Station Vent Stack Monitor Radiation High Alarm; 3/28/2005
- 328631; Receiving Station Vent Stack Monitor Alarms; 4/25/2005
- 215729-02; RETS Liquid and Gaseous Effluents and Performance Indicator Validation; 5/3/2005
- 336751; NOSID(CH) Outdated ODCM Pages and Discrepancies with Fish Collector; 5/19/2005

Corporate Comparative Audit Report 2003 REMP/ODCM/Non-Radiological Effluent Monitoring/NPDES; December 13, 2003

Focused Area Assessment Plan; Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems; May 13, 2003

LaSalle County Station Annual Radiological Environmental Operating Report:
- 2003 Erata; May 25, 2004
- 2004 Report; May 16, 2005

- 2003 Report; May 7, 2004

LaSalle County Station 2004 Effluent and Waste Disposal Report; April 29, 2005

Updated Final Safety Analysis Report, Revision 15:
- Section 7.6; Other Instrumentation Required for Safety
- Section 11; Radioactive Waste Management

Procedures:
- LAP-1800-4; LaSalle County Station Inoperable Monitor Surveillance Checklist; Revision 19
- LCP-140-26; LaSalle County Station Gaseous Effluent LLD Determination Principal Gamma Emitters and I-131; Particulate Sample; February 14, 2004
- LCO-140-26; LaSalle County Station Gaseous Effluent LLD Determination; I-131 and I-133; Charcoal Sample; February 14, 2004
- LCO-140-26; LaSalle County Station Gaseous Effluent LLD Determination; Principal Gamma Emitters; Gaseous Emissions; February 14, 2004
- LCO-140-26; LaSalle County Station Gaseous Effluent LLD Determination; Dissolved or Entrained Noble Gases; February 14, 2004
- LCO-810-30; Calibration and Performance Testing of the Gamma Spectrometer Systems; Revision 4
- LIS-PR-002; Station Vent Main Stack Effluent and Sampler Flow Rate Monitor Calibration; Revision 16

Off Site Dose Calculation Manual, Revision 2:
- LaSalle Annex Chapter 10; Revision 7
- LaSalle Annex Chapter 11; Revision 6
- LaSalle Annex Chapter 12; Revision 8

NOSA-LAS-03-08; REMP; ODCM; Non-Radiological Effluent Monitoring/NPDES Audit Report; October 24, 2003

4OA2 Identification and Resolution of Problems

Issue Reports:
- 287334; Temperature Controller is Erratic, Not Giving True Readings; 1/4/2005
- 287351; Conduit Has a Leak Up at the Roofline; 1/4/2005
- 287694; Issue Report 287351 – WO 769178-01; 1/5/2005
- 287742; Water in Local Control Panel 1PL73J; 1/5/2005
- 287987; FIN Follow-up From Troubleshooting 1TIC-VY024 Erratic Indication; 1/5/2005
- 290608; Clean Oil Off of Terminal Strips in 2PL24J; 1/13/2005
- 301768; Water Identified in Junction Boxes Feeding 1PL73J; 2/15/2005
- 317267; NRC Identifies Water Dripping from Weep Hole in VY JB; 3/25/2005
- 325273; Site Use of HU-AA-1212 is Not Acceptable; 4/15/2005
- 327468; Corrosion Found in Junction Box 2TZ-VD013C; 4/22/2005
- 328888; Inadequate Detail for Externally Sealing of Conduits; 4/26/2005
- 331299; Procedure Adherence Root Cause Corrective Actions to Prevent Recurrence (CAPRs) Deemed Ineffective; 5/3/2005

Work Orders:
- 770108-01; Water in Local Control Panel 1PL73J; 1/12/2005
- 769178-02; Clean/Dry Out Water in Control Cabinet 1PL74J and 1JB301A; 1/4/2005

Procedures:
- MA-MW-726-022; Electrical Cable Termination and Inspection; Revision 0
- NSWP-E-02; Electrical Cable Termination and Inspection; Revision 5-1
- HU-AA-1212; Technical Task Risk/Rigor Assessment, Pre-Job Brief, Independent Third Party Review, and Post-Job Brief; Revision 0

LaSalle Station Human Performance Improvement Plan, Vision, and Goals:
- 2004 Revisions
- 2005 Revisions

Action Tracking System Items:
- 177037-20; Human Performance Improvement Strategy/Plan; 9/1/2005
- 197237-04; 2004 Human Performance Cross-Functional Focused Area Self-Assessment; 9/16/2004
- 206063; Investigation of Engineering Technical Rigor Issues; 5/18/2004
- 194230-02; Design Engineering Fundamentals Assessment; 4/27/2004
- 260927; Common Cause Analysis of Department and Crew Event Free Clock Resets at LaSalle Station; 11/22/2004
- 278513; Common Cause Analysis of Potential Adverse Trend in Technical Human Performance; 12/2/2004
- 193888-05; Evaluation of Unexpected Corrective Maintenance Focused Area Self-Assessment; 10/22/2004
- 284313-04; Maintenance Radiation Worker Practices Focused Area Self-Assessment; 4/20/2005

4OA3 Event Follow-up

Drawings:
- M116; Sheet 2; Main Steam; Revision M

Licensee Event Reports:
- 05000374/2005-002-00; Pressure Boundary Leakage Discovered in 2D MSIV Drain Line Weld During Refueling Outage VT-2 Examination; 05/04/2005

Procedures:
- LOA-AP-201; Unit 2 AC Power System Abnormal; Revision 15
- LOA-VR-201; Unit 2 Recovery From a Group 4 Isolation or Spurious Trip of Reactor Building Ventilation; Revision 4
- LOR-2PM01J-A108; 250 Vdc Battery Trouble; Revision 2
- LOP-VX-01; Switchgear Heat Removal System Startup; Revision 12
- LOA-DC-201; Unit 2 DC Power System Failure; Revision 7
- LOA-PC-201; Primary-Secondary Containment Trouble; Revision 10
- LGA-002; Secondary Containment Control; Revision 3

4OA5 Other

Procedures:
- LOA-GRID-001; Low Grid Voltage; Revision 2
- LOP-AP-43; Emergency Load Conservation; Revision 1
- OP-AA-106-101-1001; Event Response Guidelines; Revision 7
- LS-AA-1010; Exelon Reportability Reference Manual Table of Contents; Revision 8
- LS-AA-1020; Reportability Tables and Decision Trees; Revision 7
- LS-AA-1110; Reportability Event SAF 1.1 – Declaration of an Emergency Classification; Revision 5
- LS-AA-1400; Event Reporting Guidelines: 10 CFR 50.72 and 10 CFR 50.73 (NUREG-1022); Revision 1
- WC-AA-101; Online Work Control Process; Revision 10
- OP-AA-108-107-1002; Interface Agreement Between Exelon Energy Delivery and Exelon Generation for Switchyard Operations; Revision 0
- OP-AA-108-107-1001; Station Response to Grid Capacity Conditions; Revision 1
- OP-AA-108-107; Switchyard Control ; Revision 2
- LOA-AP-101; Unit 1 AC Power System Abnormal; Revision 20
- LOA-AP-201; Unit 2 AC Power System Abnormal; Revision 15

LIST OF ACRONYMS USED

ACE	Apparent Cause Evaluation
ADS	Automatic Depressurization System
ALARA	As-Low-As-Is-Reasonably-Achievable
ASME	American Society of Mechanical Engineers
CAP	Corrective Action Program
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CRS	Control Room Supervisor
CSCS	Core Standby Cooling System
CY	Calendar Year
DC	Direct Current
DF	Duration Factor
DG	Diesel Generator
d/p	Differential Pressure
DRP	Division of Reactor Projects
EACE	Equipment Apparent Cause Evaluation
EDG	Emergency Diesel Generator
FC	Fuel Pool Cooling
FCV	Flow Control Valve
FSAR	Final Safety Analysis Report
FW	Feedwater
IEEE	Institute of Electrical & Electronic Engineers
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IR	Inspection Report or Issue Report
LER	Licensee Event Report
LOOP	Loss of Off-site Power
MFLCPR	Maximum Fraction of Limiting Critical Power Ratio
MSIV	Main Steam Isolation Valve
MWth	Megawatts Thermal
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NIOSH	National Institute of Safety & Health
NRC	U.S. Nuclear Regulatory Commission
NSO	Nuclear Station Operator
NSWP	Nuclear Station Work Procedure
ODCM	Offsite Dose Calculation Manual
OSP	Offsite Power
OWA	Operator Workaround
PMF	Probable Maximum Flood
RBCCW	Reactor Building Closed Cooling Water
RCIC	Reactor Core Isolation Cooling
RETS/ODCM	Radiological Effluent Technical Specification/Offsite Dose Calculation Manual
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RMCS	Reactor Manual Control System

RO	Reactor Operator
RP	Radiation Protection
RPS	Radiation Protection Specialist
RR	Reactor Recirculation
SBLC	Standby Liquid Control
SBO	Station Blackout
SCBA	Self-Contained Breathing Apparatus
SDP	Significance Determination Process
SRA	Senior Reactor Analyst
SRO	Senior Reactor Operator
SSC	Systems, Structures, and Components
TS	Technical Specification
TSO	Transmission System Operator
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
Vac	Volts Alternating Current
VD	Diesel Generator Ventilation
Vdc	Volts Direct Current
VY	Ventilation System