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

October 30, 2009

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

SUBJECT: LASALLE COUNTY STATION, UNITS 1 AND 2
NRC INTEGRATED INSPECTION REPORT 05000373/2009004;
05000374/2009004

Dear Mr. Pardee:

On September 30, 2009, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your LaSalle County Station, Units 1 and 2. The enclosed inspection report documents the inspection results, which were discussed on October 16, 2009, with the Site Vice President, Mr. Dave Wozniak, and other members of your staff.

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

Based on the results of this inspection, three findings of very low safety significance were identified. Two of the findings were associated with violations of NRC requirements. The NRC is treating these violations as non-cited violations (NCVs) consistent with Section VI.A.1 of the NRC Enforcement Policy because of the very low safety significance of the violations and because they are entered into your corrective action program (CAP). Additionally, a licensee identified violation which was determined to be of very low safety significance is listed in this report.

If you contest the subject or severity of an NCV, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the LaSalle County Station. In addition, if you disagree with the characterization of any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region III, and the NRC Resident Inspector at the LaSalle County Station. The information that you provide will be considered in accordance with Inspection Manual Chapter 0305.

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

Sincerely,

/RA/

Kenneth Riemer, Chief
Branch 2
Division of Reactor Projects

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

Enclosure: Inspection Report 05000373/2009004; 05000374/2009004
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	05000373; 05000374
License Nos:	NPF-11; NPF-18
Report No:	05000373/2009004; 05000374/2009004
Licensee:	Exelon Generation Company, LLC
Facility:	LaSalle County Station, Units 1 and 2
Location:	Marseilles, IL
Dates:	July 1, 2009, through September 30, 2009
Inspectors:	G. Roach, Senior Resident Inspector F. Ramírez, Resident Inspector M. Mitchell, Region III Health Physics Inspector M. Learn, Region III Branch 2 Reactor Engineer S. Bakhsh, Region III Reactor Inspector J. Bozga, Region III Reactor Inspector V. Meghani, Region III Reactor Inspector J. Neurauter, Region III Reactor Inspector R. Temps, Safety Inspector, Office of Nuclear Materials Safety and Safeguards J. Yesinowski, Illinois Dept. of Emergency Management
Approved by:	Kenneth Riemer, Chief Branch 2 Division of Reactor Projects

Enclosure

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SUMMARY OF FINDINGS

IR 05000373/2009-004, 05000374/2009-004; 7/01/2009 - 9/30/2009; LaSalle County Station, Units 1 & 2; Operability Evaluations, Problem Identification and Resolution, and Event Response.

The report covered a 3-month period of inspection by the resident inspectors and an announced inspection by a regional health physics inspector. Three Green findings were identified, of which two were non-cited violations (NCVs). The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609 “Significance Determination Process” (SDP). Cross-cutting aspects were determined using IMC 0305, "Operating Reactor Assessment Program." Findings for which the SDP does not apply may be "Green," or be assigned a severity level after Nuclear Regulatory Commission (NRC) management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealed Findings

Cornerstone: Initiating Events

- Green. The inspectors identified a finding of very low safety significance for the licensee failing to recognize that an existing alarm condition in the unit 2 digital electro-hydraulic control system (DEHC) trip logic would result in a turbine trip and subsequent reactor scram when weekly turbine trip testing was performed. The licensee entered this issue into its corrective action program (CAP) as issue report (IR) 953784.

  The finding was greater than minor because it affected the initiating events objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during power operations and was associated with the cornerstone attribute of Configuration Control. The inspectors determined that the finding was Green, or of very low safety significance, by answering no to the IMC 0609 Phase 1 Screening Worksheet question “Does the finding contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available?” The finding had a cross-cutting aspect in the area of Human Performance (resources) in that the site’s design documentation was not complete and accurate with regards to the necessary ramifications of a control module communications failure (H.2(c)). (Section 4OA3)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding of very low safety significance and an associated NCV of Technical Specification (TS) 5.4.1, “Procedures”, for the failure to provide adequate procedural guidance to operations personnel when performing the quarterly SBLC operability test on unit 2. Specifically, operations personnel performing LOS-SC-Q1, “SBLC pump operability test,” did not posses appropriate procedural guidance while performing this test and, as a result, did not declare both trains for the Standby Liquid Control (SBLC) system inoperable and did not enter the associated limiting condition for operation (LCO) action statements as required per TSs.

  The inspectors determined that the finding was more than minor because it was associated with the procedure quality attribute of the Mitigating Systems Cornerstone, and it affected the cornerstone objective to ensure availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, operations personnel would not have been able to return SBLC to a standby configuration if needed in case of an anticipated transient without a scram (ATWS) in 120 seconds as required by the design basis. The finding was determined to be of very low safety significance using the SDP Phase 2. This finding was also related to the cross-cutting area of Human Performance (resources) because the procedure used for this evolution was inaccurate in that it provided improper guidance to maintain SBLC operability provided that a dedicated operator was briefed and stationed locally. The licensee entered this issue into the corrective action program. Corrective actions taken by the licensee included the future revision of procedure LOS-SC-Q1 to remove the statement that indicates that the system can be maintained operable during the surveillance and to include an emergency restoration attachment with steps to quickly return the system to its standby configuration if required in case of an ATWS. (Section 1R15)

Cornerstone: Miscellaneous

- Green. The inspectors identified a Green (Severity Level IV) NCV of 10 CFR 50.72 (b)(3)(v) for the licensee's failure to make a required non-emergency eight-hour notification to the NRC for a loss of safety function of a system which was required to remove residual heat from the reactor. The licensee entered this issue into their CAP as IR 971982.

  The inspectors determined that the finding should be evaluated using the traditional enforcement process, since the failure to make a required report to the NRC had the potential to impact the agency's ability to perform its regulatory function. The finding was considered to be Severity Level IV, as the NRC Enforcement Policy states, in part, that "the severity level of a violation involving the failure to make a required report to the NRC will be based upon the significance of and the circumstances surrounding the matter that should have been reported." As such, the ability of the operators to restore a train of the residual heat removal (RHR) system by non-extraordinary means and in a timely manner (without experiencing an unplanned mode change) to a shutdown cooling lineup was considered by the inspectors to have mitigated the effects of the loss of functionality of the decay heat removal system to a very low safety impact on the plant. (Section 4OA2)

B. Licensee-Identified Violations

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

REPORT DETAILS

Summary of Plant Status

Unit 1

The unit began the inspection period operating at full power. On July 18, 2009, the unit was shutdown to replace 1B reactor recirculation (RR) pump degraded seals. The unit was restarted on July 21, 2009, and was returned to full power on July 24, 2009. On September 6, 2009, power was reduced to 71 percent for control rod pattern adjustments, channel distortion testing, control rod scram timing, and main steam isolation valve and turbine control valve surveillances. The unit was returned to full power on September 7, 2009, where it operated for the remainder of the inspection period.

Unit 2

The unit began the inspection period at full power. On August 15, 2009, the unit experienced a reactor scram during the main turbine trip during turbine overspeed testing. The unit was maintained in Mode 3, “Hot Shutdown” during DEHC repairs. The unit was restarted on August 18, 2009, and was returned to full power on August 20, 2009. On September 12, 2009, power was reduced to 63 percent for control rod pattern adjustment, control rod scram timing, main steam isolation valve surveillance, turbine control valve surveillance and feedwater pump surveillance. The unit was returned to full power on September 13, 2009, where it operated for the remainder of the inspection period.

1. REACTOR SAFETY

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

1R01 Adverse Weather Protection (71111.01)

.1 External Flooding

a. Inspection Scope

The inspectors evaluated the design, material condition, and procedures for coping with the design basis probable maximum flood. The evaluation included a review to check for deviations from the descriptions provided in the Updated Final Safety Analysis Report (UFSAR) for features intended to mitigate the potential for flooding from external factors.

As part of this evaluation, the inspectors reviewed an analysis of the expected flooding level due to the new dry cask storage cask haul path. The inspectors verified that with the addition of the haul path, the maximum external site probable maximum precipitation flood level experienced by safety-related structures was below the flood threshold and within the UFSAR.

Additionally, the inspectors performed a walkdown of the haul path and affected areas to verify that assumptions used in the analysis were conservative as well as to identify any other modifications to the site which would inhibit site drainage during a probable maximum precipitation event or allow water ingress past a barrier.

This inspection constituted one external flooding sample as defined in Inspection Procedure (IP) 71111.01-05.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Quarterly Partial System Walkdowns

a. Inspection Scope

The inspectors performed partial system walkdowns of the following risk-significant systems:

- Unit 1 reactor core isolation cooling (RCIC) with high pressure core spray (HPCS) out-of-service;
- Unit 2 B and C RHR trains with A RHR out-of-service; and
- Unit 1 A train diesel generator (DG).

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

These activities constituted three partial system walkdown samples as defined in IP 71111.04-05.

b. Findings

No findings of significance were identified.

.2 Semi-Annual Complete System Walkdown

a. Inspection Scope

On August 8, 2009, the inspectors performed a complete system alignment inspection of the service water (WS) on Units 1 and 2 to verify the functional capability of the system.

This system was selected because it was considered risk significant in the licensee's probabilistic risk assessment. The inspectors walked down the system to review mechanical and electrical equipment line ups, electrical power availability, system pressure and temperature indications, as appropriate, component labeling, component lubrication, component and equipment cooling, hangers and supports, operability of support systems, and to ensure that ancillary equipment or debris did not interfere with equipment operation. A review of a sample of past and outstanding WOs was performed to determine whether any deficiencies significantly affected the system function. In addition, the inspectors reviewed the CAP database to ensure that system equipment alignment problems were being identified and appropriately resolved. Documents reviewed are listed in the Attachment to this report.

These activities constituted one complete system walkdown sample as defined in IP 71111.04-05.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Routine Resident Inspector Tours (71111.05Q)

a. Inspection Scope

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

- Unit 1 RCIC/ low pressure core spray (LPCS) room 673' elevation (fire zone 2I4);
- Unit 1 hydrogen seal oil (fire zone 5B7);
- Unit 1 motor-driven reactor feed pump (MDRFP) room (fire zone 5B9);
- Unit 2 B and C RHR pump room (fire zone 3I3); and
- Unit 1 RCIC and LPCS support room 694' elevation (fire zone 2H4).

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

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

b. Findings

No findings of significance were identified.

1R07 Annual Heat Sink Performance (71111.07)

.1 Heat Sink Performance

a. Inspection Scope

The inspectors reviewed the licensee's testing of the 2A DG heat exchanger to verify that potential deficiencies did not mask the licensee's ability to detect degraded performance, to identify any common cause issues that had the potential to increase risk, and to ensure that the licensee was adequately addressing problems that could result in initiating events that would cause an increase in risk. The inspectors reviewed the licensee's observations as compared against acceptance criteria, the correlation of scheduled testing and the frequency of testing, and the impact of instrument inaccuracies on test results. Inspectors also visually verified the as found condition of various internal heat exchanger components, and reviewed the licensee's final engineering assessment report for the heat exchanger prior to restoration. Documents reviewed for this inspection are listed in the Attachment to this report.

This annual heat sink performance inspection constituted one sample as defined in IP 71111.07-05.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Resident Inspector Quarterly Review (71111.11Q)

a. Inspection Scope

On September 16, 2009, the inspectors observed a crew of licensed operators in the plant's simulator during licensed operator requalification training to verify that operator performance was adequate, evaluators were identifying and documenting crew performance problems and training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

- licensed operator performance;
- crew's clarity and formality of communications;
- ability to take timely actions in the conservative direction;
- prioritization, interpretation, and verification of annunciator alarms;

- correct use and implementation of abnormal and emergency procedures;
- control board manipulations;
- oversight and direction from supervisors; and
- ability to identify and implement appropriate TS actions and Emergency Plan actions and notifications.

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

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

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Routine Quarterly Evaluations (71111.12Q)

a. Inspection Scope

The inspectors evaluated degraded performance issues involving the following risk-significant components:

- emergency core cooling system pump maintenance program; and
- motor-operated valve (MOV) maintenance program.

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

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

The inspectors assessed performance issues with respect to the reliability, availability, and condition monitoring of the system. In addition, the inspectors verified the maintenance history of the above components and noted when the licensee changed the frequency of maintenance/inspection windows to ensure that the components maintained their effectiveness. The inspectors also verified that maintenance issues were entered into the CAP with the appropriate significance characterization. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings of significance were identified.

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

.1 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

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

- emergent Unit 1 B RR seal replacement;
- Unit 1 RCIC emergent minimum flow valve repair;
- walkdown of Unit 2 A RHR maintenance outage protected pathways;
- walkdown of shutdown electrical lineup post Unit 2 reactor scram; and
- walkdown of the protected pathway during the 2A DG work window.

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

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

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

.1 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following issues:

- Unit 1 D RHR/WS pump establishment of new baseline inservice testing (IST) criteria;
- American Society of Mechanical Engineers (ASME) section XI pressure test for RR flange;
- Unit 1 loss of shutdown down cooling during jumper removal in the containment isolation valve control circuit;
- manual recovery actions for SBLC during testing; and
- degraded 2A DG fuel oil chemistry.

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

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

b. Findings

Introduction: The inspectors identified a finding of very low safety significance (Green) and an associated NCV of TS 5.4.1, "Procedures", for the failure to provide adequate procedural guidance to the operations personnel when performing the quarterly SBLC operability test in accordance with procedure LOS-SC-Q1, "SBLC pump operability test". As a result, operations personnel failed to declare both trains of the SBLC system inoperable during the performance of the LOS-SC-Q1, and did not enter the associated LCO Action Statements as required per the LaSalle TS.

Description: On July 31, 2009, the inspectors observed Unit 2 operations personnel perform the A SBLC pump quarterly run using LOS-SC-Q1, "SBLC pump operability test." As specified in LOS-SC-Q1, in order to maintain operability of the SBLC system and its availability for on-line risk purposes, an operator was briefed to restore SBLC system to a standby condition if it was necessary in case of an ATWS. This operator would be stationed in the vicinity of the system while the surveillance was being performed and would complete the necessary actions for the restoration of SBLC system. Because SBLC has no automatic actuation features and must be manually initiated, the assignment of a dedicated operator in the field to maintain operability of the system was considered permissible as specified by the above procedure and also the TS basis for this surveillance requirement.

The inspectors noted that LOS-SC-Q1 did not contain an emergency restoration procedure that would instruct the operator on the manipulations needed to promptly return the system to standby. The inspectors also noted that in order to maintain the system operable the operators would have had to complete the valve manipulations in 120 seconds as specified in OP-LA-101-111-1002, "LaSalle Operation Philosophy Handbook." Per the station's design basis, the 120-second requirement is established for suppression pool temperature limitations. The inspectors questioned several operators about what restoration actions they would take to return the system to its standby configuration. The answers provided to the inspectors from the different operators were inconsistent. Without clear procedural guidance (emergency restoration steps) and the short duration timeframe available to restore system per the design basis, the inspectors determined that there was not enough time to perform the manipulations needed to restore SBLC to standby if during the surveillance the system was needed to respond to an ATWS event.

Since procedure LOS-SC-Q1 provided guidance that the SBLC system could be maintained operable during the surveillance, provided that a dedicated operator was briefed and stationed locally, the licensee did not consider the inoperability of the system. In addition, since the design basis specified that SBLC is needed 120-seconds after an ATWS, the inspectors concluded that the operations personnel could not perform the manipulations necessary to return the system to standby in that time and as such, both trains of SBLC should have been declared inoperable and the LCO Action Statements as required per the site's TS 3.1.7, "Standby Liquid Control Systems" should have been entered.

<u>Analysis</u>: The inspectors concluded that the failure to enter TS LCO 3.1.7 Condition B, "Two SLC subsystems inoperable" during the performance of procedure LOS-SC-Q1 "SBLC pump operability test" constituted a performance deficiency that warranted evaluation using the SDP. Using IMC 0612, Appendix B, "Issue Screening", the inspectors determined that the finding was of more than minor significance because it was associated with the procedure quality attribute of the Mitigating Systems Cornerstone, and it affected the cornerstone objective to ensure availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. To further assess significance of the finding, the inspectors used IMC 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," and determined that Mitigating Systems was the only cornerstone affected. Using the Mitigating Systems column on the Phase 1 SDP characterization worksheet, the inspectors determined that the finding constituted a loss of safety function due to both trains of SBLC system being inoperable for greater than design basis time. As a result, the inspectors transitioned to SDP Phase 2 where the finding screened as Green. The finding was also determined to have been related to the cross-cutting area of Human Performance, as defined in IMC 0305, "Operating Reactor Assessment Program". Specifically, the finding was related to the Resources Component because procedure LOS-SC-Q1, which is performed as the quarterly surveillance for the SBLC system, was not adequate in that it provided inappropriate guidance to maintain the system operable while the surveillance was being performed. As a result, the associated LCO Action Statements were not entered. (Aspect H.2(c))

Enforcement: Technical Specifications 5.4.1, “Procedures”, requires that written procedures shall be established, implemented, and maintained as recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, Appendix A, Section 4, “Procedures for Startup, Operation, and Shutdown of Safety-Related BWR Systems,” specifically addresses the need to have appropriate procedures for the operation of the SBLC system. The licensee developed procedure LOS-SC-Q1, “SBLC Pump Operability Test” to implement that requirement. Contrary to the above, Procedure LOS-SC-Q1 was not appropriate to the circumstances, in that it did not provide adequate precautionary guidance to account for the inoperability of the SBLC system during the surveillance performance. Consequently, on July 31, 2006, and in multiple occasions before that, the licensee failed to declare both trains of SBLC system inoperable during the surveillance and did not enter the associated LCO Action Statements required per the TS. Because this finding was determined to be of very low safety significance and has been entered into the licensee’s CAP (IR 966512), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. The licensee’s corrective actions include the future revision of Procedure LOS-SC-Q1 to remove the statement that indicates that the system can be maintained operable during the surveillance and to include an emergency restoration attachment with steps to quickly return the system to its standby configuration if required in case of an ATWS. (NCV 05000373/2009004-01; 05000374/2009004-01)

1R19 Post-Maintenance Testing (71111.19)

.1 Post-Maintenance Testing

a. Inspection Scope

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

- Auxiliary electrical equipment room ventilation envelope pressurization test;
- 1B RR seal operational pressure test;
- Unit 2 “C” RHR pump following a planned maintenance outage;
- Unit 2 turbine bypass valves following actuator maintenance; and
- Technical Support Center DG ASCO® Transfer Switch.

These activities were selected based upon the structure, system, or component's ability to impact risk. The inspectors evaluated these activities for the following (as applicable): the effect of testing on the plant had been adequately addressed; testing was adequate for the maintenance performed; acceptance criteria were clear and demonstrated operational readiness; test instrumentation was appropriate; tests were performed as written in accordance with properly reviewed and approved procedures; equipment was returned to its operational status following testing (temporary modifications or jumpers required for test performance were properly removed after test completion); and test documentation was properly evaluated. The inspectors evaluated the activities against TS, the UFSAR, 10 CFR Part 50 requirements, licensee procedures, and various NRC generic communications to ensure that the test results adequately ensured that the equipment met the licensing basis and design requirements. In addition, the inspectors reviewed CAP documents associated with post-maintenance tests to determine whether the licensee was identifying problems and entering them in the CAP and that the problems were being corrected commensurate with their importance to safety. Documents reviewed are listed in the Attachment to this report.

This inspection constituted five PMT samples as defined in IP 71111.19-05.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

.1 Surveillance Testing

a. Inspection Scope

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

- LOS-RI-Q5, Unit 1 RCIC cold start (Routine);
- LOS-SC-Q1, Unit 2 A SBLC quarterly run (Routine);
- LOS-HP-Q1, Unit 1 HPCS quarterly run (Routine);
- LOS-DG-M3, 1B DG monthly idle start (Routine);
- review of Unit 2 containment atmosphere monitoring equipment following indications of a potential leak in containment (reactor coolant system (RCS) leakage); and
- LOS-RH-Q1; RHR WS pump and valve IST (IST).

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

- did preconditioning occur;
- were the effects of the testing adequately addressed by control room personnel or engineers prior to the commencement of the testing;
- were acceptance criteria clearly stated, demonstrated operational readiness, and consistent with the system design basis;
- plant equipment calibration was correct, accurate, and properly documented;
- as-left setpoints were within required ranges; and the calibration frequency were in accordance with TSs, the UFSAR, procedures, and applicable commitments;
- measuring and test equipment calibration was current;
- test equipment was used within the required range and accuracy; applicable prerequisites described in the test procedures were satisfied;
- test frequencies met TS requirements to demonstrate operability and reliability; tests were performed in accordance with the test procedures and other applicable procedures; jumpers and lifted leads were controlled and restored where used;
- test data and results were accurate, complete, within limits, and valid;
- test equipment was removed after testing;

- where applicable for IST activities, testing was performed in accordance with the applicable version of Section XI, ASME code, and reference values were consistent with the system design basis;
- where applicable, test results not meeting acceptance criteria were addressed with an adequate operability evaluation or the system or component was declared inoperable;
- where applicable for safety-related instrument control surveillance tests, reference setting data were accurately incorporated in the test procedure;
- where applicable, actual conditions encountering high resistance electrical contacts were such that the intended safety function could still be accomplished;
- prior procedure changes had not provided an opportunity to identify problems encountered during the performance of the surveillance or calibration test;
- equipment was returned to a position or status required to support the performance of its safety functions; and
- all problems identified during the testing were appropriately documented and dispositioned in the CAP.

Documents reviewed are listed in the Attachment to this report.

This inspection constituted four routine surveillance testing samples, one IST sample, and one RCS leak detection inspection sample as defined in IP 71111.22, Sections -02 and -05.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

.1 Emergency Preparedness Drill Observation

a. Inspection Scope

The inspectors evaluated the conduct of a routine licensee emergency drill on September 1, 2009, to identify any weaknesses and deficiencies in classification, notification, and protective action recommendation development activities. The inspectors observed emergency response operations in the Technical Support Center to determine whether the event classification, notifications, and protective action recommendations were performed in accordance with procedures. The inspectors also attended the licensee drill critique to compare any inspector-observed weakness with those identified by the licensee staff in order to evaluate the critique and to verify whether the licensee staff was properly identifying weaknesses and entering them into the CAP. This evolution was planned to be evaluated and included in performance indicator (PI) data regarding drill and exercise performance. As part of the inspection, the inspectors reviewed the drill package listed in the Attachment to this report.

This EP drill observation inspection constituted one sample as defined in IP 71114.06-05.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

.1 Plant Walkdowns and Radiation Work Permit Reviews

a. Inspection Scope

The inspectors assessed the adequacy of the licensee's internal dose assessment process for internal exposures in excess of 50 millirem committed effective dose equivalent. There were no internal exposures greater than 50 millirem committed effective dose equivalent.

This inspection constituted one sample as defined in IP 71121.01-5.

The inspectors also reviewed the licensee's physical and programmatic controls for highly activated and/or contaminated materials (non-fuel) stored within the spent fuel pool or other storage pools.

This inspection constituted one sample as defined in IP 71121.01-5.

b. Findings

No findings of significance were identified.

.2 High Risk Significant, High Dose Rate, High Radiation Area, and Very High Radiation Area Controls

a. Inspection Scope

The inspectors held discussions with the Radiation Protection Manager concerning high dose rate, high radiation area, and very high radiation area controls and procedures, including procedural changes that had occurred since the last inspection, in order to assess whether any procedure modifications substantially reduced the effectiveness and level of worker protection.

This inspection constituted one sample as defined in IP 71121.01-5.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

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

4OA1 Performance Indicator Verification (71151)

.1 Mitigating Systems Performance Index - Emergency AC Power System

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index (MSPI) - Emergency AC Power System PI for both Unit 1 and Unit 2 for the period from the third quarter 2008 through the second quarter 2009. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the Nuclear Energy Institute (NEI) Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5 were used. The inspectors reviewed the licensee's operator narrative logs, MSPI derivation reports, IRs, event reports and NRC Integrated Inspection Reports for the period of July 2008 through June 2009, to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee's IR database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

This inspection constituted two MSPI emergency AC power system samples as defined in IP 71151-05.

b. Findings

No findings of significance were identified.

.2 Mitigating Systems Performance Index - High Pressure Injection Systems

a. Inspection Scope

The inspectors sampled licensee submittals for the MSPI - High Pressure Injection Systems PI for both Unit 1 and Unit 2 for the period from the third quarter 2008 through the second quarter 2009. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5 were used. The inspectors reviewed the licensee's operator narrative logs, IRs, MSPI derivation reports, event reports and NRC Integrated Inspection Reports for the period of July 2008 through June 2009, to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee's IR database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and noted that all licensee identified reporting issues had been corrected. Documents reviewed are listed in the Attachment to this report.

This inspection constituted two MSPI high pressure injection system samples as defined in IP 71151-05.

b. Findings

No findings of significance were identified.

.3 Mitigating Systems Performance Index – Residual Heat Removal System

a. Inspection Scope

The inspectors sampled licensee submittals for the MSPI – RHR System PI for both Unit 1 and Unit 2 for the period from the third quarter 2008 through the second quarter 2009. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5 were used. The inspectors reviewed the licensee's operator narrative logs, IRs, event reports, MSPI derivation reports, and NRC Integrated Inspection Reports for the period of July 2008 through June 2009, to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee's IR database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings of significance were identified.

.4 Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspectors sampled licensee submittals for the Occupational Radiological Occurrences PI for the period from the second quarter 2008 through the second quarter 2009. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5 were used. The inspectors reviewed the licensee's assessment of the PI for occupational radiation safety to determine if indicator-related data was adequately assessed and reported. To assess the adequacy of the licensee's PI data collection and analyses, the inspectors discussed with radiation protection staff, the scope and breadth of its data review, and the results of those reviews. The inspectors independently reviewed electronic dosimetry dose rate and accumulated dose alarm and dose reports and the dose assignments for any intakes that occurred during the time period reviewed to determine if there were potentially unrecognized occurrences. The inspectors also conducted walkdowns of numerous locked high and very high radiation area entrances to determine the adequacy of the controls in place for these areas. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one occupational radiological occurrences sample as defined in IP 71151-05.

b. Findings

No findings of significance were identified.

.5 Radiological Effluent Technical Specifications/Offsite Dose Calculation Manual Radiological Effluent Occurrences

a. Inspection Scope

The inspectors sampled licensee submittals for the Radiological Effluent TS/Offsite Dose Calculation Manual Radiological Effluent Occurrences PI for the period of July 2008 through July 2009. The inspectors used PI definitions and guidance contained in the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline", Revision 5 to determine the accuracy of the PI data reported during those periods. The inspectors reviewed the licensee's IR database and selected individual reports generated since this indicator was last reviewed to identify any potential occurrences such as unmonitored, uncontrolled, or improperly calculated effluent releases that may have impacted offsite dose. The inspectors reviewed gaseous effluent summary data and the results of associated offsite dose calculations for selected dates between July 2008 and July 2009, to determine if indicator results were accurately reported. The inspectors also reviewed the licensee's methods for quantifying gaseous and liquid effluents and determining effluent dose. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one radiological effluent occurrences sample as defined in IP 71151-05.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness, Public Radiation Safety, Occupational Radiation Safety, and Physical Protection

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

a. Scope

As part of the various baseline inspection procedures discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify that they were being entered into the licensee's CAP at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. Attributes reviewed included: 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. Minor issues entered into the licensee's CAP as a result of the inspectors' observations are included in the attached List of Documents Reviewed.

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

b. Findings

No findings of significance were identified.

.2 Daily Corrective Action Program Reviews

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

b. Findings

No findings of significance were identified.

.3 Annual Sample: Review of Operator Workarounds

a. Scope

The inspectors evaluated the licensee's implementation of their process used to identify, document, track, and resolve operational challenges. Inspection activities included, but were not limited to, a review of the cumulative effects of the operator workarounds (OWAs) on system availability and the potential for improper operation of the system, for potential impacts on multiple systems, and on the ability of operators to respond to plant transients or accidents.

The inspectors performed a review of the cumulative effects of OWAs. The documents listed in the Attachment to this report were reviewed to accomplish the objectives of the inspection procedure. The inspectors also reviewed operator challenges, which create an obstacle to normal plant operation, rather than the more severe obstacle to safe plant operation created by an OWA. The inspectors reviewed both current and historical operational challenge records to determine whether the licensee was identifying operator challenges at an appropriate threshold, had entered them into their CAP and proposed or implemented appropriate and timely corrective actions which addressed each issue. Reviews were conducted to determine if any operator challenge could increase the possibility of an Initiating Event, if the challenge was contrary to training, required a change from long-standing operational practices, or created the potential for inappropriate compensatory actions. Additionally, all temporary modifications were reviewed to identify any potential effect on the functionality of Mitigating Systems, impaired access to equipment, or required equipment uses for which the equipment was not designed. Daily plant and equipment status logs, degraded instrument logs, and operator aids or tools being used to compensate for material deficiencies were also assessed to identify any potential sources of unidentified OWAs. In addition, interviews were conducted with equipment operators and licensed control room operators to determine if longstanding workarounds existed and had in turn been proceduralized into a part of accepted practice.

This review constituted one OWA annual inspection sample as defined in IP 71152-05.

b. Findings

No findings of significance were identified.

.4 Selected Issue Follow-Up Inspection: Failure to Make a Non-Emergency Event Notification to the NRC Following a Loss of Shutdown Cooling on Unit 1

a. Scope

The inspectors reviewed the licensee's response to a loss of shutdown cooling on Unit 1 while the plant was in a cold shutdown condition. The inspectors verified the plant was returned to a stable, safe shutdown condition with the normal method of decay heat removal through the RHR system restored. In addition, the inspectors reviewed the licensee's root cause for the event and the site's determination for reportability to the NRC of the event.

This review constituted one in-depth problem identification and resolution sample as defined in IP 71152-05.

b. Findings

Introduction: The inspectors identified a Green (Severity Level IV) NCV of 10 CFR 50.72 (b)(3)(v) for the licensee's failure to make a required non-emergency eight-hour notification to the NRC for a loss of safety function of a system which was required to remove residual heat from the reactor.

Description: On July 20, 2009, Unit 1 was in cold shutdown (Mode 4) with the "A" train of the RHR system operating in the shutdown cooling configuration. Prior to transitioning the plant from cold shutdown (Mode 4) to hot shutdown (Mode 3) as a part of the eventual return to power operations, the licensee runs both "A" and "B" loops of

RHR in parallel to flush the system to prevent areas of high radiation levels (hot spots) from building up in stagnant portions of piping. This has the effect of lowering the overall occupational exposure levels of plant staff by lowering dose rates in areas of the plant that are accessible under normal operating conditions.

When both loops of RHR pumps are started the initial flow conditions create the potential for a spurious closure of the common pump suction containment isolation valve 1E12-F009 on a perceived high flow condition. This containment isolation is designed to prevent a loss of coolant accident outside of containment due to a leak in the RHR system. If a higher than expected flow is sensed in the common suction piping, a control relay will cause a closure of 1E12-F009 to stop the potential interfacing system loss of coolant accident. In order to prevent the spurious containment isolation, the licensee proceduralized (LOP-RH-08) the installation of jumpers that bypass the 1B21H-K29 relay, which in turn bypasses the function of the 1B21H-K077 relay, which would cause the containment isolation to occur.

Once both pumps were started the licensee ordered the removal of the jumpers to restore the containment isolation function to normal. During the first leg of the jumper removal process, instrument maintenance division (IMD) technicians performing the task noted a spark. At this point no other abnormal indications were observed. The technicians continued removal of the jumper, secured the panel and left the area to report completion of jumper removal to the control room. Within one minute of the IMD technicians leaving the space, the 1B21H-K077 relay was heard to have changed states by other technicians working in an adjacent panel and the control room observed the closure of the 1E12-F009 common RHR pump shutdown cooling suction isolation valve. The closure of the common suction valve resulted in both “A” and “B” RHR pumps tripping and a complete loss of shutdown cooling. The control room operators reset the containment isolation logic, re-opened the 1E12-F009, and started the “A” RHR pump. The RCS heated up approximately 7 degrees Fahrenheit (F) during the 12 minutes that the shutdown cooling lineup was lost. The licensee exceeded the upper limit on the administrative temperature band that had been established (140 degrees Fahrenheit) prior to the event. The maximum temperature reached in the RCS during the event was 147 degrees Fahrenheit.

The licensee reviewed the event for 10 CFR 50.72 reportability, but determined that it was not reportable as the actual plant conditions never existed that would have resulted in a valid containment isolation signal and that the containment isolation in itself only affected one component in one system and did not span multiple systems. The inspectors concurred with this assessment, but also noted that the safety function of the RHR system to remove decay heat was lost during the 12 minutes that both trains were isolated. This in itself was considered reportable. The inspectors consulted NUREG 1022 “Event Report Guidelines 10 CFR 50.72 and 50.73” and the Office of Nuclear Reactor Regulation staff responsible for the maintenance of the reporting program to validate their concerns regarding the licensee’s failure to report the simultaneous loss of both trains of RHR as a loss of safety function with regards to decay heat removal. NUREG 1022, which is considered the NRC staff’s position on the reporting of nuclear events, says, in part, that “if a single RHR suction line valve should fail in such a way that RHR cooling cannot be initiated, the event would be reportable.” The closing of the common RHR shutdown cooling isolation valve 1E12-F009 represented this exact scenario.

Analysis: The inspectors determined that a failure to make a required non-emergency report to the NRC as required by 10 CFR 50.72 (b)(3)(v) was a performance deficiency warranting further evaluation. The inspectors determined that the finding should be evaluated using the traditional enforcement process, since the failure to make a required report to the NRC had the potential to impact the agency's ability to perform its regulatory function. The finding was considered to be Severity Level IV as the NRC Enforcement Policy states in part that, "the severity level of a violation involving the failure to make a required report to the NRC will be based upon the significance of and the circumstances surrounding the matter that should have been reported." As such, the ability of the operators to restore a train of the RHR system by non-extraordinary means and in a timely manner (without experiencing an unplanned mode change) to a shutdown cooling lineup was considered by the inspectors to have mitigated the effects of the loss of functionality of the decay heat removal system to a very low safety impact on the plant.

Enforcement: 10 CFR 50.72 (b)(3)(v) states in part, "Eight hour reports – any event or condition that at the time of discovery could have prevented the fulfillment of the safety function of structures or systems that are needed to: remove residual heat." Contrary to the above requirement, the licensee failed to make a required 8-hour non-emergency notification to the NRC following the loss of all trains of RHR while in shutdown cooling mode due the spurious closure of the common suction isolation valve 1E12-F009. Because the licensee entered this into their CAP as IR 971982, the issue is being treated as a Severity Level IV NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000373/2009004-02).

4OA3 Follow-Up of Events and Notices of Enforcement Discretion (71153)

.1 Unit 2 Automatic Reactor Scram Due to Main Turbine Overspeed Testing with a Latent Digital Electro-Hydraulic Control Fault Present

a. Inspection Scope

On August 15, 2009, the inspectors observed licensee management and staff respond to an automatic reactor scram on Unit 2. The inspectors responded to the control room verifying that the plant was stable and in a safe, shutdown condition. The scram occurred during weekly turbine trip logic testing due to a latent communications failure that existed between two of the three trip channels. When the third channel was taken to test, the 2 out of 3 logic conditions were met tripping the main turbine which resulted in an automatic reactor scram. The inspectors also observed the licensee's DEHC troubleshooting efforts and decision making regarding the subsequent reactor startup. The inspectors will be reviewing the licensee's root cause report which will be documented in Inspection Report 2009005.

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

b. Findings

Introduction: The inspectors identified a finding of very low safety significance (Green) for the licensee failing to recognize that an existing alarm condition in the Unit 2 DEHC trip logic would result in a turbine trip and subsequent reactor scram when weekly turbine trip testing was performed.

Description: On August 15, 2009, Unit 2 experienced a reactor scram during weekly turbine trip testing. Specifically, the DEHC system trip logic was satisfied resulting in a turbine trip and subsequent automatic reactor scram when control module R was taken to test to simulate an overspeed condition on the main turbine. On August 9, 2009, an EHC minor trouble alarm was received on Unit 2. The alarm indicated to the operators that a communications failure had occurred between the S and T control modules. Subsequent troubleshooting by engineering determined that both the S and T control modules were each still providing valid output signals to the trip logic, but a communication error existed between the cards. After reviewing the vendor manual and consulting with the vendor and corporate engineering, the licensee determined that it was permissible to perform the weekly turbine trip logic surveillances on August 15, 2009.

During the subsequent root cause review, the licensee identified that the communications failure that existed between control modules S and T caused the S module to believe that the T module was in a tripped condition even though T was providing a normal on-line output signal. When testing was performed on August 15, 2009, the R module was placed in test. With R in test and S believing T was tripped, the S module sympathetically went to a tripped condition believing that the two out of three trip logic had been satisfied. With S in trip and R in test for overspeed testing, the main turbine trip logic was now satisfied resulting in the Unit 2 scram.

The resulting fast closure of the turbine stop and control valves caused an expected pressure spike in the RCS. This pressure spike was mitigated by the opening of safety relief valves U and S. The valves opened relieving pressure and subsequently re-closed in accordance with their design function. The licensee performed a required non-emergency 4-hour notification to the NRC Headquarters Operations Officer in accordance with 10CFR50.72 (b)(2)(iv)(B) for a valid reactor protection system actuation signal with the reactor in a critical state.

Analysis: The inspectors determined that a reactor scram, a challenge to the Initiating Events Cornerstone, caused by the licensee's lack of understanding of the plant configuration which resulted from the DEHC trip logic control module communications failure, was a performance deficiency warranting a significance determination. The finding was greater than minor because it affected the initiating events objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during power operations and was associated with the cornerstone attribute of Configuration Control. The inspectors determined that the finding was Green or of very low safety significance by answering "no" to the IMC 0609 Phase 1 Screening Worksheet question "Does the finding contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available?" The finding had a cross-cutting aspect in the area of Human Performance Resources in that the site's design documentation was not complete and accurate with regards to the necessary ramifications of a control module communications failure (Aspect H.2(c)).

Enforcement: The licensee's failure to identify the configuration of the Unit 2 DEHC system following a control module communications failure was a performance deficiency. The licensee did unsuccessfully attempt to address the alarming condition through review of the vendor manual and consultation with the vendor and corporate engineering experts. The licensee remains responsible for all decisions and actions performed onsite and as such, the decision to perform turbine testing with the alarm present which resulted in a scram was a performance deficiency, but not a violation of NRC requirements. The licensee entered this issue into its CAP as IR 953784. (FIN 05000374/2009004-03)

.2 (Closed) LER 05000373/2009002-00, Loss of Shutdown Cooling Due to Spurious Closure of the Shutdown Cooling Suction Isolation Valve

On July 20, 2009, LaSalle Unit 1 was in mode 4 cold shutdown. At 1448 CDT the inboard shutdown cooling suction isolation valve unexpectedly closed, causing a trip of both the 1A and 1B RHR pumps and a loss of shutdown cooling. The licensee entered their abnormal procedures, reopened the valve and restarted the 1A RHR pump at 1500 CDT restoring shutdown cooling. Reactor coolant temperature rose approximately 7 degrees F during the event. The inspectors reviewed the Licensee Event Report (LER) and no findings of significance were identified and no violation of NRC requirements occurred during the event. However, the licensee failed to make an 8-hour notification to the NRC in accordance with 10 CFR 50.72(b)(3)(v) which is further discussed in section 4OA2. The licensee documented the loss of shutdown cooling in IR 943883. Documents reviewed as part of this inspection are listed in the Attachment to this report. This LER is closed.

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

4OA5 Other Activities

.1 Preoperational and Operational Testing of an Independent Spent Fuel Storage Installation (ISFSI) (60854.1)

a. Inspection Scope

An inspection of the licensee's activities that support the upcoming dry fuel storage dry run was initiated, which included in-office review of plant modifications and corresponding design calculations. The inspectors identified technical concerns related to the crane and the reactor building steel superstructure calculations supporting the crane upgrade. These calculations were performed to demonstrate compliance with the single failure proof crane design requirements per ASME NOG-1-2004 standard and the plant design basis requirements for the reactor building structural steel. Due to these concerns and additional issues pertaining to the inadequate design implementation of the heavy loads design requirements, the licensee revised the dry cask storage schedule from summer of 2009 to spring of 2010. Due to a change in the licensee's loading schedule, the inspectors' activities related to the inspection are on-going and will be continued into the next several quarters' inspection activities.

b. Findings

No findings of significance were identified at this time.

.2 Onsite Fabrication of Components and Construction of an ISFSI (60853)

a. Inspection Scope

By letter dated August 5, 2009, the NRC exercised enforcement discretion against Holtec International for a violation of NRC requirements regarding 10 CFR 72.48(c)(2). Specifically, Holtec was cited for their failure to obtain a certificate of compliance amendment prior to implementing a change that eliminated a helium leak rate test of the multi-purpose canister (MPC) confinement boundary weldment at fabrication. In Holtec's September 2, 2009, response letter to the NCV, Holtec indicated that leakage testing had been reinstated at the manufacturing facility on July 1, 2009, and that onsite leakage testing of all unloaded MPCs was being scheduled with customers and would be performed prior to loading of those MPCs.

The LaSalle nuclear plant plans to operate an ISFSI utilizing the Holtec dry cask storage system. Six MPCs were previously delivered to the site for use at the ISFSI. It was determined that these MPCs required onsite leakage testing prior to their use. A senior inspector from the NRC's Division of Spent Fuel Storage and Transportation observed Holtec's performance of onsite helium leak testing on one of the six MPCs from August 31 through September 2, 2009.

The inspector met with LaSalle and Holtec personnel and toured the MPCs test area which was near the ISFSI pad. Holtec contracted with Leak Test Specialists (LTS) to perform the helium leak testing of the MPCs at LaSalle. The inspector reviewed LTS procedure MSLT-MPC-HOLTEC, "Helium Mass Spectrometer Leak Test Procedure," Revision 2, approved by Holtec that was used to administer the helium leak test for the MPCs.

The inspector observed the establishment of a vacuum inside the MPC. This test was followed by connection of a mass spectrometer leak detector and introduction of helium on the outside of the MPC, which was "hooded" in plastic. The mass spectrometer leak detector readings were used to calculate the final corrected leak rate. The entire process from start of vacuum drawdown to leak rate determination took about 48 hours.

During the performance of the leak test the inspector noted that when helium was introduced into the plastic hood and the helium concentration value was recorded, the value differed from that described in step 7.22 of MSLT-MPC-HOLTEC. The difference resulted in a more conservative corrected leak rate value being calculated. The observation was discussed with LaSalle, Holtec and LTS personnel. The LTS personnel indicated that the LTS technicians were trained to perform the test as observed by the inspector. The LTS onsite management committed to revise step 7.22 to more accurately reflect the observed practice. Revision 3 of MSLT-MPC-HOLTEC was provided to the inspector. The revised procedure addressed the inspector's observation. The inspector observed the corrected leak rate for the MPC leak test. Overall, no concerns were noted in the test procedure or its implementation.

b. Findings

No findings of significance were identified.

4OA6 Management Meetings

.1 Exit Meeting Summary

On October 16, 2009, the inspectors presented the inspection results to Mr. D. Wozniak, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

.2 Interim Exit Meetings

An interim exit was conducted for:

- The results of the access control to radiologically significant areas inspection were discussed with the Plant Manager, Mr. D. Rhoades, on September 17, 2009. The inspectors confirmed that none of the potential report input discussed was considered proprietary. Proprietary material received during the inspection was returned to the licensee.

- On October 9, 2009, the inspectors presented the inspection results of the preoperational and operational testing of an ISFSI inspection and the onsite fabrication of components and construction of an ISFSI inspection in an interim debrief with Mr. Terry Simpkin of the licensee's staff. Mr. Simpkin acknowledged the information presented. The inspectors asked whether any material examined during the inspection should be considered proprietary. No proprietary information was identified.

4OA7 Licensee-Identified Violations

The following violation of very low safety significance (Green) was identified by the licensee and is a violation of NRC requirements which met the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as an NCV.

- Technical Specification 5.2.2.d requires that procedures be established, implemented, and maintained covering the control of plant staff overtime, to limit the work hours worked by staff performing safety-related functions in accordance with the NRC Policy Statement on working hours (NRC GL 82-12). The NRC's GL 82-12, "Nuclear Power Plant Staff Working Hours" specifies in part that guidelines should be followed that limit individuals to working no more than 72 hours in any 7-day period. Recognizing that very unusual circumstance may arise, requiring deviation from this guideline, such deviation shall be authorized by the plant manager or his deputy, or higher levels of management. Contrary to this requirement, from September 16 through September 18, 2009, an IMD technician worked 8 hours over the 72-hour limit in any 7-day period without prior plant management authorization. This was identified in the licensee's CAP as IR 969479. This finding is of very low safety significance because no human performance issues or significant events were directly linked to personnel fatigued as a result of the hours worked.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

D. Wozniak, Site Vice President
D. Rhoades, Plant Manager
K. Aleshire, Exelon EP Programs Manager
D. Amezaga, GL 89-13 Program Owner
J. Bashor, Site Engineering Director
L. Blunk, Operations Training Manager
D. Carpenter, Senior ISFSI Project Manager
H. Do, Corporate ISI Manager
P. Endress, Design Engineer
J.C. Feeney, NOS Lead Assessor
F. Gogliotti, System Engineering Senior Manager
D. Henly, Design Engineer
W. Hilton, Engineering Supervisor – Mechanical/Structural
K. Ihnen, Nuclear Oversight Manager
A. Kochis, ISI Engineer
R. Leasure, Radiation Protection Manager
K. Taber, Operations Director
B. Maze, ISFSI Project Manager
J. Meyer, Exelon Nuclear Oversight Inspector
J. Miller, NDE Level III
B. Rash, Maintenance Director
J. Rommel, Design Engineering Senior Manager
K. Rusley, Emergency Preparedness Manager
J. Shields, ISI Program Supervisor
S. Shields, Regulatory Assurance
T. Simpkin, Regulatory Assurance Manager
H. Vinyard, Work Management Director
J. Vegara, Regulatory Assurance
W. Trafton, Shift Operations Superintendent
J. White, Site Training Director
G. Wilhelmsen, Design Manager
S. Wilkinson, Chemistry Manager
C. Wilson, Station Security Manager

Nuclear Regulatory Commission

K. Riemer, Chief, Reactor Projects Branch 2

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000373/2009004–01; 05000374/2009004–01	NCV	Failure to declare SBLC system inoperable during surveillance testing
05000374/2009004–02	NCV	Failure to make required non-emergency 50.72 notification to NRC following loss of shutdown cooling
05000374/2009004–03	FIN	Reactor scram during turbine testing

Closed

05000373/2009004–01; 05000374/2009004–01	NCV	Failure to declare SBLC system inoperable during surveillance testing
05000374/2009004–02	NCV	Failure to make required non-emergency 50.72 notification to NRC following loss of shutdown cooling
05000374/2009004–03	FIN	Reactor scram during turbine testing
05000373/2009002–00	LER	Loss of shutdown cooling due to spurious closure of the shutdown cooling suction isolation valve

LIST OF DOCUMENTS REVIEWED

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

<u>1R01 Adverse Weather Protection (71111.01)</u>

Issue Reports:
- 856960; New Dry Cask Haul Path Affects Site Flooding Zones; 12/16/2008

Working Documents:
- EC 373566; Evaluation of the Expected Flooding Level Due to the New Dry Cask Haul Path; 3/25/2009

Drawings:
- S-1720; ISFSI Haul Path Location Plan; Rev. 0

Miscellaneous:
- LSCS-UFSAR 2.4; Hydrologic Engineering; Rev. 14

<u>1R04 Equipment Alignment (71111.04)</u>

Procedures:
- LOP-RI-01E; Unit 1 Reactor Core Isolation Cooling System Electrical Checklist: Rev. 11
- LOP-RI-01M; Unit 1 Reactor Core Isolation Cooling System Mechanical Checklist: Rev. 19

Issue Reports:
- 950159; 1B WS Pump Needs OTBD Packing Adjusted; 8/6/2009

Working Documents:
- LOP-DG-01E; Unit 1 A Diesel Generator Electrical Checklist; 8/22/1997
- LOP-DG-01M; Unit 1 A Diesel Generator mechanical Checklist; 6/12/2003
- LOP-RH-2BM; Unit 2B Residual Heat Removal System Mechanical Checklist; 3/3/2008
- LOP-RH-2CM; Unit 2 C Residual Heat Removal System Mechanical Checklist; 3/3/2008
- LOP-WS-01M; Unit 1 Service Water System Mechanical Checklist; 4/23/2007

Miscellaneous:
- Work Week 200928;7/6-7/12/2009

1R05 Fire Protection (71111.05)

Issue Reports:
- 927960; Neil Fire Protection System Testing; 6/4/2009
- 947561; NRC ID'd: Housekeeping Issues with U2 RB 673'; 7/30/2009

Miscellaneous:
- 2I4 Fire Pre-Att.; Information Sheet for Rx Bldg. 673' Elev. LPCS/RCIC Pump Cubicle; 2/2/2006
- 5B7 Fire pre-att.; Information Sheet for Turbine Building Unit 1 Hydrogen Seal Oil Skid 731'0"; 2/2/2006
- LSCS-FPR Table H.3-2; Combustible Loading and Extinguishing Capability; Rev. 3

1R07 Annual Heat Sink Performance (71111.07)

Issue Reports:
- 969442; New 2A DG Cooler End Cover Coating Issue; 9/23/2009

Procedures:
- ER-AA-330-008; Exelon Service Level I, and Safety-related (Service Level III) Protective Coatings; Rev. 6
- ER-AA-340-1002; Service Water Heat Exchanger Inspection Guide; Rev. 4

Work Documents:
- HX 2DG01A/ER-AA-340-1002; HX Inspection Report: 2A DG Cooler; 9/28/2009
- EC 372326; 0DG Thermal Performance Margin with Tubes Blocked; 10/20/2008
- Ver. 44 9-21-09 1015; 2A DG Window for 9-28-09 week; 9/21/2009

Calculations:
- 97-195 / EC 334017; Thermal Model of ComEd/LaSalle Station Unit 0, 1, and 2 Diesel Generator Jacket Water Coolers; Rev. A

Miscellaneous:
- GL 89-13; Program Basis Document: Heat Exchanger Visual Inspection Acceptance Criteria (VIAC); Rev. 6
- IR 597664; Apparent Cause Evaluation: 2A DG Heat Exchanger Coating Failure; 3/1/2007

1R11 Licensed Operator Requalification Program (71111.11)

Miscellaneous:
- Simulator training scenario 3rd quarter 2009

1R12 Maintenance Effectiveness (71111.12)

Procedures:
- MA-AA-723-301; Periodic Inspection of Limitorque Model SMB/SB/SBD-000 Through 5 Motor Operated Valves; Rev. 5
- LES-EQ-102; Testing of Environmentally Qualified Motors; Rev. 9
- LMS-GM-01; HBC Valve Actuator Grease Inspection and Lubrication Application; Rev. 14

Issue Reports:
- 687445; 2E51-C003 Pump Mechanical Seal Leak; 10/21/2007
- 689882; Instrument Out of Tolerance Review for RI System; 10/26/2007
- 723020; High Vibes/Low Disch Press on RCIC Water Leg Pump; 1/16/2008
- 946225; Unit 1 RCIC Water Leg Pump Discharge Pressure Reading Low; 7/27/2009
- 947308; Control Room Alarm C RHR Low Pressure; 7/29/2009
- 947379; Oil Leak on 2B/C RHR Water Leg Pump Housing Nipple; 7/30/2009
- 950941; Unexpected MCR Alarm. 2C RHR Discharge Pressure Low; 8/7/2009
- 951144; U2 Div II Waterleg Pump Oil Leakage Higher then Previous IR; 8/9/2009
- 951171; Contingency WO for Div 3 WLP & Use for Unit 2 Div 2; 8/9/2009
- 952661; Need Contingency WO for Unit 2 b/c RHR Water Leg Pump; 8/12/2009
- 957133; Water Leg Pump Impeller Diameter Inconsistencies; 8/25/2009
- 961952; Flange Thickness of Spare/Replacement Water leg Pumps; 9/9/2009
- 962839; Serial # of ECCS & RCIC Water Leg Pumps; 9/9/2009
- 965914; Grease Sample Quality from MOV Actuator Degraded; 9/16/2009

Work Documents:
- WO 1083707; 2E22C003 HPCS Water Leg Pump Reservoir; 5/08/2009

Miscellaneous:
- ; U1 Safety-related MOV Listing; 2009
- ; U2 Safety-related MOV Listing; 2009
- VM J-0687; Vendor Equipment Technical Information Program (VETIP) Limitorque – Valve Operators; 5/14/1998

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

Procedures:
- LOP-RR-08; Changing Reactor Recirc Pump Speed from Fast to Slow Speed; Rev. 34
- OP-AA-108-111; Att. 1: Adverse Condition Monitoring and Contingency Plan for Unit 1 RR Pump Seal Degradation (Rev 5); Rev. 4

Issue Reports:
- 938723; 1 B RR Pump Seal Cavity #2 Pressure Spike; 7/4/2009
- 939611; Need to Validate Alarm Condition for 1B RR PMP Seal Trouble; 7/8/2009
- 939756; 1B RR Seal Leakage Input Complex TS Plan Documentation; 7/8/2009
- 940604; AR Subj: After Autopsy of 1B RR Pump Seal Send Offsite for Analysis;7/15/2009
- 943453; 1B RR Pump Seal Cover Carbon Bushing Minor Damage; 7/19/2009
- 943476; 1B33-C001B Replace Pump Seal Leakage Cover in L1R13; 7/19/2009
- 944354; Increased Vibration During Slow Speed Operation; 7/21/2009
- 946713; NRC Question on RI Availablity; 7/28/2009
- 946742; NRC Identified Steam Leak from U1 MDRFP; 7/28/2009
- 946791; NRC Resident Concerns; 7/28/2009
- 946792; Small Fishing Boat Sunk in the LaSalle Lake; 7/28/2009
- 948168; 1B RR Pump Seal Degradation; 7/31/2009
- 948388; 1B RR Pump Upper thrust Bearing Hi temperature PPC Alarm
- 949535; Flowserve Preliminary L1M19 Autopsy of 1B RR Pump Seal; 8/4/2009
- 949537; Recommendations from Flowserve Regarding RR Seal Replacement; 8/4/2009

Work Documents:
- Protected Equipment List, 2A DG Work Window; 9/28/2009

Drawings and Graphs:
- 1B Reactor Recirculation Pump Seal (L1C13), Cavity 1 & 2 Seal Pressure; 2/28/2008 – 8/27/2009
- Unit 1 B Reactor Recirc #2 Seal Pressure; 7/31/2009

Miscellaneous:
- Adverse Condition Monitoring and Contingency Plan for Unit 1 RR Pump Seal Degradation; 6/26/2009
- Email from Jeffrey Miller: 1 B RR Seal & DWEDS Detailed information; 7/8/2009
- LaSalle Operations Log; 7/28 – 7/29/2009
- Licensee Presentation: LaSalle Unit 1 1B RR Pump Seal Degradation; 7/7/2009
- Work Week 200931; 7/27 – 8/2/2009
- INPO TR9-66; Topical Report: Reactor Recirculation and Coolant Pump Seal-Related Events; 6/2009
- IR 931985-02; Issue Resolution Documentation for whether U1 should Shutdown to Replace 1B RR Pump Seal; 6/25/2009
- IR 946327; Complex Troubleshooting Data Sheet for Unit 1, Mode 1, RCIC Min Flow Valve Cycled Excessively during LOS-RI-Q5; 7/27/2009

1R15 Operability Evaluations (71111.15)

Procedures:
- ER-AA-330-009; Pressure Testing Following Replacements with Core Criticality at BWRs: Alternate Requirements for Small Items; Rev. 5
- HU-AA-104-101; Procedure Use and Adherence; Rev. 3
- LOP-RH-07; Shutdown Cooling System Startup, Operation and Transfer; Rev. 58
- LOP-RH-08; Shutdown Cooling System Shutdown; Rev. 33
- LOS-DG-SR2; 0 Diesel generator Action Statement Operability Test; Rev. 20
- LOS-DO-SR2; Diesel Fuel Oil Analysis Verification (New Fuel Oil); Rev. 13
- LTS-900-8; Operation of High/Low Pressure Water Leak Rate Test Rig; Rev. 17

Issue Reports:
- 943767; PMT for ASME Code Class 1 Bolted Repairs after L1M19; 7/20/2009
- 943883; Spurious Isolation of RHR SDC Inboard Isolation; 7/20/2009
- 961947; New Diesel Fuel Oil Water & Sediment Analysis; 9/4/2009

Work Documents:
- DG99-000135; Conduct of ASME Section XI Pressure Testing Following Replacements with Core Criticality at Boiling Water Reactors; 2/19/1999
- LAS03.G03; ISI Program Plan Third Ten-Year Inspection Interval; Rev. 0
- LOS-DO-SR2; Diesel Fuel Oil Analysis Verification; Rev. 13
- PM 96111-01; Data Package Lab Analysis Report by Analysts Inc., for LOS-DO-SR2 Diesel Fuel Oil Att. A (New Fuel Oil); 9/4/2009
- PM 96111-02; Data Package Lab Analysis Report by Analysts Inc., for LOS-DO-SR2 Diesel Fuel Oil Att. B (New Fuel Oil Within 31 Day); 9/4/2009
- TCCP # 1-PO23-09; Checklist for Unit 1 Bypass SDC High Flow and RPV High Pressure Isolation (LOP-RH-08); 7/20/2009

Drawings:
- M-93; P & ID Nuclear Boiler & Reactor Recirculating System; Rev. AV

Miscellaneous:
- Amendment 147/133; Programs and Manuals - Diesel Fuel Oil Testing Program (5.5.10);
- ASTM D 2709 – 96; Standard Test Method for Water and Sediment in Middle Distillate Fuels by Centrifuge; 1999
- ASTM D 4176 – 93; Standard Test Method for Free Water and Particulate Contamination in Distillate Fuels (Visual Inspection Procedures); 1997
- EC 376246; Establish Revised Reference Value and Acceptance Criteria for IST Biennial Comprehensive Pump Testing for 1E12-C300D; Rev. 0
- IR 943883; Equipment Prompt Investigation Report for Spurious Isolation of RHR SDC Inboard Isolation; 7/20/2009

<u>1R19 Post-Maintenance Testing (71111.19)</u>

Procedures:
- LGP-1-1; Normal Unit Startup; Rev. 86
- LIS-EH-203; Unit 2 Turbine Bypass System Response Time Test and Level 8 Trip Test; Rev. 13
- LOP-NB-02; Operations with the Potential to Drain the Reactor Vessel; Rev. 10
- LOP-RR-02; Draining, Filling, and Pressure Testing an Isolated Reactor Recirc Loop; Rev. 20
- LOP-RH-24; Temporary Fill for A, B & C RHR System Discharge Lines; Rev. 9
- LOS-RH-Q1; RHR (LPCI) and RHR Service Water Pump and Valve Inservice Test for Modes 1,2,3,4 and 5; Rev. 71
- LOS-TG-M4; Turbine Bypass Valve Surveillance; Rev. 5
- LOS-TX-Q1; Diesel Generator 0Dg09K Quarterly Test; Rev. 12
- LOS-VC-SR1; Auxiliary Electric Equipment Room HVAC Pressurization Surveillance; Rev. 2

Issue Reports:
- 946225; Unit 1 RCIC Water Leg Pump Discharge Pressure Reading Low; 7/27/2009
- 947308; Control Room Alarm C RHR Low Pressure; 7/29/2009
- 947379; Oil Leak on 2B/C RHR Water Leg Pump Housing Nipple; 7/30/2009
- 947561; NRC Id'd: Housekeeping Issues with U2 RB 673'; 7/30/2009
- 963844; 231B-7 ASCO Transfer Switch Failed During LOS-TX-Q1

Work Documents:
- LOS-RH-M1; Tech Spec Surveillance – Unit 2 RHR C Att. 2C; 7/31/2009
- LOS-RH-Q1; Tech Spec Surveillance – 2C RHR Pump Att. 2C; 7/28/2009
- LOS-RH-Q1; Unit 2 C RHR System Operability and Inservice Test Checklist; Rev. 71
- LOS-VC-SR1, Att. B,D,E; Checklist for Auxiliary Electric Equipment Room HVAC Pressurization Surveillance; 7/17/2009
- WO 1089556; LES-RH-206 ATT C Min Flow Time Delay Relay Test; 5/12/2009

Drawings:
- M-93; P & ID Nuclear Boiler & Reactor Recirculating System; Rev. AX

Miscellaneous:
- ;LaSalle Operations Log; 7/29 – 7/30/2009

1R22 Surveillance Testing (71111.22)

Procedures:
- LOP-NB-03; Troubleshooting Drywell Leakage; Rev. 1
- LOS-DG-M3; 1B(2B) Diesel Generator Operability Test; Rev. 71
- LOS-HP-Q1; HPCS System Inservice test; Rev. 63
- LOS-RH-Q1; RHR (LPCI) and RHR Service Water Pump and Valve Inservice Test for Models 1,2,3,4 and 5; Rev. 71
- LOS-SC-Q1; SBLC Pump Operability/Inservice Test and Explosive Valve Continuity Check; Rev. 28
- LRP-5821-53; 1(2)PL15J Primary Containment Panel Particulate and Nobel Gas Monitor Performance Check; Rev. 1
- OP-LA-101-111-1002; LaSalle Operations Philosophy Handbook; Rev. 23
- WC-AA-101; On-Line Work Control Process; Rev. 16

Issue Reports:
- 946200; NOS ID: Clearance Order Electronic Sign on Not Used; 7/27/2009
- 946203; RCIC Overspeed Connector Needs Repaired; 7/27/2009
- 946225; Unit 1 RCIC Water Leg Pump Discharge Pressure Reading Low; 7/27/2009
- 946303; RCIC Suction Pressure Gauge Disagreement; 7/27/2009
- 946304; RCIC Suction Pressure Gauge Disagreement; 7/27/2009
- 946327; Unit 1 RCIC Min Flow Valve Cycling; 7/27/2009
- 946814; Unit 1 RCIC Barometric Condenser Vac. Tank Low Level Alarm; 7/28/2009
- 946900; Unexpected Alarm on RCIC Barometric Vacuum Tank Level Low; 7/28/2009
- 948170; NRC Identified: No Emergency Restoration ATT For LOS-SC-Q1
- 951498; LOS-DG-M3 As-Found 1VY02A Flow Low; 8/10/2009
- 952624; OP_LA_101-111-1002 Needs Revision for Consistency with PRA; 8/12/2009
- 953484; NRC Question: SBLC Availability During Surveillance; 8/14/2009
- 955588; NRC Question on ARI Initiation During U2 Scram; 8/20/2009
- 960793; NRC Concern: Maintenance of SBLC Operability Status During L; 9/2/2009
- 965578; 345 KV Line 0104 Trip; 9/16/2009
- 966125; Panel 2PL75J – High Particulate Channel Activity;9/17/2009
- 967365; 2PL75J Rad Monitor Alarm; 9/19/2009
- 968370; Hi Rad Alarm From 2PL75J; 9/222/2009
- 968524; Request Set Point Change for 2PL75J Part Channel; 9/22/2009
- 971089; 2PL15J Particulate Channel Alarms; 9/27/2009

Working Documents:
- DCR 991595; Design Analysis Approval: Change NED-I-EIC-0158 to Derive Allowable Values and Expanded Tolerances; 9/19/2000
- EC 360691; CSCS Cooling Water Flow Margins for Operability of the ECCS Cubicle Room Coolers and DG Coolers; Rev. 0
- LOS-2009-35; Procedure Change Request LOS-DG-M3; 8/10/2009
- LOS-DG-SR7; Division 3 Cooling Water System Test Procedure Checklist; 8/10/2009
- WO 1228244; LOS-SC-Q1 U2 A SBLC “Biennial Comprehensive Test” Att. 2A; 7/31/2009
- WO 1231552; LOS-RI-Q5 U1 RCIC Cold-Quick Start, Att. 1A; 7/24/2009
- WO 1239276; LOS-RH-Q1 2A RHR WS Operability & Inservice Test (Biennial); 8/24/2009
- WO 1243592; LOS-HP-Q1 U2 HPCS Comprehensive Pump Test Att 2A (Wk 12); 9/9/2009
- WO 1249644; LOS-DG-M3 1B DG Idle Start Att. 1B-Idle; 8/10/2009

Calculations:
- A.47;LaSalle HRA Notebook: Operator Fails to Initiate SLC Early; 7/12/2007

Graphs:
- 2PL75JPART; Year to Date 2PL75J Particulate Containment Monitoring Log for Various Isotopes; 9/2009

Miscellaneous:
- LaSalle DLoop ATWS Event Tree; 8/21/2009
- LaSalle News Flash from Dave Rhoades, Plant Manager: Unsecured High Rad Boundary Reinforces Need to Focus on Human Performance; 8/24/2009
- LaSalle Operations Log; 7/26 – 7/27/2009
- LaSalle Operations Log; 7/31/2009
- LaSalle Operations Log; 9/14 – 9/23/2009
- LaSalle Operations Log; 9/16/2009
- Review of Surveillances for Week of August 17, 2009 for Availability Maintained with Operator Actions; 8/17-8/23/2009
- Standby Liquid Control (SC) Activities/Notes;
- Amendment 147/133; Reactivity Control Systems; Standby Liquid Control (SLC) System;
- B 3.1.7-1; Reactivity Control Systems; Standby Liquid Control (SLC) System; Rev.0
- C467060024-7435-3/21/2008; LaSalle Event Tree Notebook; Reactivity Control with SBLC and RPV Level Control; 3/21/2008
- IR 946327; Engineering Summary (informal) of RCIC Min Flow Valve Excessive Cycling Issue; date unknown
- LSCS-UFSAR 15.8; Anticipated Transients without Scram (ATWS); Rev. 13
- LSCS-UFSAR 15.8.1; References List for Anticipated Transients without Scram (ATWS); Rev. 16
- LSCS-UFSAR 9.3; Standby Liquid Control System; Rev. 13
- Part 9900; Technical Guidance, Operability Determinations Process; 4/16/2008
- RM LS-MISC-05; Input to Operations for OP-LA-101-111-01002, LaSalle Operations Philosophy Handbook; Rev. 0

<u>1EP6 Drill Evaluation (71114.06)</u>

- EP Drill Package; Third Quarter, 2009

<u>2OS1 Access Control to Radiologically Significant Areas (71121.01)</u>

Procedures:
- RP-AA-220; Bioassay Program; Revision 5
- RP-AA-221; Whole Body Count Data Review; Revision 1
- RP-AA-390; Spent Fuel Pool Material Control; Revision 3
- RP-AA-400; ALARA Program; Revision 5
- RP-AA-401; Operational ALARA Planning and Controls; Revision 9
- RWP 10009373; L2R12 Drywell Emergent Work; Revision 0
- RWP 10009396; L2R12 Emergent Under Vessel Nuclear Instrumentation; Revision 0
- RWP 10009402; L2R12; Emergent Work in the Reactor Building; Revision 0
- RWP 10009447; L1R12 Turbine Building Emergent Work Activities; Revision 0

Issue Reports:
- 848246; Check-In Self-Assessment: Lead Shielding Program; 5/21/2009
- 861378-02; Check-In Self-Assessment: Dose Timekeeping; 6/26/2009
- 848240; Check-In Self-Assessment: Access Control to Radiologically Significant Areas and ALARA Planning and Controls; 7/21/2009
- 874282; Secured High Radiation Area Unlocked; 1/30/2009
- 897924; Unexpected Airborne Air Sample; 3/26/2009
- 907825; Contamination Discovered Outside of a Posted Contaminated Area; 4/15/2009
- 921607; New High Radiation Area Posted; 5/19/2009
- 924020; New High Radiation Area Posted; 5/26/2009
- 925044; Airborne Radiation Area Radwaste Exhaust Filter Room; 5/21/2009
- 941362; Un-necessary Dose Received Recovering Seal Parts; 7/8/2009
- 945167; Mechanical Maintenance Technician Entered High Radiation Area Under Wrong Radiation Work Permit; 7/23/2009
- 948525; Hot Spot Controlled as Administrative High Radiation Area; 8/1/2009
- 950325; Secured High Radiation Area Not Secured; 8/6/2009
- 951830; Potential Adverse Trend in Radiation Worker Practices; 8/11/2009
- 952854; NOS Identified: Radworker Performance Issue Review; 8/13/2009
- 956955; High Radiation Area Found Not Secured; 8/24/2009

Miscellaneous:
- L2R12 Refueling Outage Report

4OA1 Performance Indicator Verification (71151)

Issue Reports:
- 798767; Discrepancies in Design Analysis CE-LS-003; 7/21/2008
- 800332; U-1 HPCS Pump Excessive Squealing Noise Upon Startup; 7/25/2008
- 804192; 1B RHR Pump Seal Cooler Flowrate < 12.5 GPM; 8/5/2008
- 809466; SSPI Monthly Goals are in Variance; 8/21/2008
- 840907; NRC Identified Error in MSPI Unavailability Reporting; 11/5/2008
- 883847; Retorque Packing for the 2E12-F068B; 2/23/2009
- 915437; Replace the Stem Nut for the 1E12-F068A Next PVT Interval; 5/4/2009
- 915428; Retorque Packing for the 1E12-F068A; 5/4/2009
- 932256; 1A DG MVAR Swings; 6/17/2009
- 932269; 1A DG KVAR Anomaly; 6/17/2009
- 932469; 1A DG Voltage Regulator Pot R3 Resistance Variations; 6/18/2009
- 957705; Identified Historical Error in MSPI Unavailability; 8/26/2009

Miscellaneous:
- RM SA-1561; LaSalle MSPI Basis Document; Rev. 6
- RM SA-1561; LaSalle MSPI Basis Document; Rev. 7

4OA2 Identification and Resolution of Problems (71152)

Procedures:
- OP-AA-102-103; Operator Work-Around Program; Rev. 2

Issue Reports:
- 943883; Spurious Isolation of RHR SDC Inboard Isolation; 7/20/2009
- 946040; Potential Operator Challenge: U-2 CRD Accumulator 54-43;7/27/2009
- 953852; Alterex Temperature Rose Following the Unit 2 Scram; 8/16/2009
- 957667; DEHC Alarm Issues; 8/26/2009
- 957967; Fire Alarm – 2FP10J Trouble; 8/26/2009
- 958057; Responded to an Alarm on the 1PL-15J; 8/27/2009
- 958171; NOS ID: Operator Burden Program Implementation; 8/27/2009
- 923329; CRD FCV Placed in Manual From Auto For Rod Exercising; 5/23/2009
- 902423; Operator Work Around Board Meeting Results; 4/3/2009
- 917852; LD RWCU Rooms Differential Temperature High Alarm; 5/10/2009
- 891117; U-2 CRD FCV Controller; 3/10/2009
- 934253; Observed Increase in U1 Offgas Pre-Treat Activities; 6/23/2009

Issue Reports Resulting from NRC/IEMA Inspection:
- 936955; High Radiation Area Found Not Secured; 8/24/2009
- 939469; NRC Question after Walkdown for Flooding; 7/7/2009
- 940506; IDNS Identified Housekeeping Issue; 7/10/2009
- 940813; NRC Resident Inspector Observation; 7/10/2009
- 941336; IEMA Concern; 7/13/2009
- 942791; IEMA Identified 6 Potential Flooding Hazard in U2 RCIC Room; 7/16/2009
- 946742; NRC Identified Steam Leak from U1 MDRFP; 7/28/2009
- 946791; NRC Resident Concerns; 7/28/2009
- 947561; NRC ID'D: Housekeeping Issues with U2 RB 673'; 7/30/2009
- 948170; NRC Identified: No Emergency Restoration ATT For LOS-SC-Q1; 7/31/2009
- 949458; NRC Observation: 1C41-R002 temperature Indication; 8/4/2009
- 951501; NRC Resident Question Regarding Hot Weather Alert; 8/10/2009
- 953484; NRC Question: SBLC Availability During Surveillance; 8/14/2009
- 955588; NRC Question on ARI Initiation During U2 Scram; 8/20/2009
- 956811; NRC Identified: Miscommunication on Work in Progress; 8/24/2009
- 957475; NRC Identified – Groundwater Leak in U2 Div 1 CSCS Pump Room; 8/25/2009
- 960793; NRC Concern: Maintenance of SBLC Operability Status During L; 9/2/2009

Miscellaneous:
- Agenda20090915.doc; Operator Workaround Board Meeting Agenda; 9/15/2009
- EN 45025; Event Notifications Residual Heat Removal (RHR) Pump Tripped While in Operation for Shutdown Cooling; 4/28/2009
- NCV ML083190078; (PIM) Failure to Report a Reportable Condition; 12/31/2008
- Operator Burden Report – Operator Burden Aggregate Assessment Form; 8/9/2009

4OA3 Follow-Up of Events and Notices of Enforcement Discretion (71153)

Procedures:
- LOP-EH-11; EHC Workstation Alarm Response and Other Information; Rev. 8
- LOR-2H13-P601-A308; Reactor Vessel Water Level 2 Lo-Lo; Rev. 3
- LOR-2H13-P603-B107; Div II ARI Logic Initiated; Rev. 3
- LOR-2H13-P603-B401; Division 1 RX Vessel Water Level 2 Lo-Lo; Rev. 4

Issue Reports:
- 923112; NOS ID: Unit One Scram Documentation; 5/22/2009
- 951150; U-2 DEHC Alarm; 8/9/2009
- 953784; U-2 Automatic Scram from Full Power; 8/15/2009
- 953806; Relief Valve 2C11-F460B Lifted Following the Unit 2 Scram; 8/15/2009
- 953837; S and U SRV's Opened During the Scram; 8/16/2009
- 953845; 2A TDRFP Tripped Twice on Overspeed Following Scram on Unit; 8/16/2009
- 953854; 2FW036A and B had to be Closed Following the Unit 2 Scram; 8/16/2009
- 953861; U2 RT Isolation on U2 Scram; 8/16/2009
- 953872; Unit 2 RT Isolated on Delta Flow; 8/16/2009

Event Notification:
- 45265; Unit 2 Automatic Scram Due to Turbine Trip; 8/15/2009

Engineering Changes:
- EC 347737; Upgrade Wide Range Reactor Level Indication to Mitigate Ringing; 3/1/2005

Drawings:
- 1E-2-4205AB; Schematic Diagram Reactor Recirculation System "RR" (B33) Part 2; Rev. T
- 1E-2-4207CA; Schematic Diagram Alternate Rod Insertion SYS RD (C22) Pt. 6; Rev. C
- 1E-2-4207CD; Schematic Diagram Alternate Rod Insertion System "RD" (C22) Part 9; Rev. D
- 1E-2-4232AJ; Schematic Diagram Primary Containment & Reactor Vessel Isolation Alarm
  1E-2-4205AM; Schematic Diagram Reactor Recirculation System "RR" (B33) Part 12; Rev. V
- System PC (B21H) Part 9; Rev. M
- 1E-2-4232AV; Schematic Diagram Primary Containment & Reactor Vessel Isolation Sys. PC (B21H) Part 21; Rev. D
- M-139; P&ID Nuclear Boiler & Reactor Recirculation System; Revs. AR, AI

Graphs and Charts:
- One Two-Variable Trend: RX Water Level WR Level and RX Pressure WR; 6/15/2009, 22:46
- One Two-Variable Trend: RX Water Level WR Level and RX Pressure WR; 6/15/2009, 22:51
- Two Two-Trend: RX Water Level, RX Pressure; 8/19/2009
- Unit 2 RX Water Level WR Level; 8/17/2009

Miscellaneous:
- Control Room Operator Log; 8/15/2009
- U2 Sequence Events Recorder; 8/15/2009
- U1 ARI Div 2 Init; Scram Post Transient Review Concerns to Address; 5/21/20090

4OA3 Follow-Up of Events and Notices of Enforcement Discretion (71153)

Procedures:
- MSLT-MPC-HOLTEC; Helium Mass Spectrometer Leak Test Procedure; Revs. 2 and 3

Issue Reports:
- 949081; Insufficient Control of Design Analysis; 8/3/2009
- 950425; Reactor Building Crane Seismic Calculation Error; 8/6/2009
- 950435; Reactor Building Crane Weld Calculation Error; 8/6/2009
- 953633; LaSalle County Station UFSAR Table 3.7-1 Has Inconsistencies; 8/14/2009
- 957014; Compliance with NOG-1 Rules for Single Failure Proof Cranes; 8/24/2009

LIST OF ACRONYMS USED

AC	Alternating Current
ADAMS	Agencywide Document Access Management System
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without a Scram
CAP	Corrective Action Program
CFR	Code of Federal Regulations
CRD	Control Rod Drive
DEHC	Digital Electro-Hydraulic Control System
DG	Diesel Generator
F	Fahrenheit
FCV	Flow Control Valve
HPCS	High Pressure Core Spray
IMC	Inspection Manual Chapter
IMD	Instrument Maintenance Division
IP	Inspection Procedure
IR	Issue Report
ISFSI	Independent Spent Fuel Storage Installation
IST	Inservice Testing
LER	Licensee Event Report
LCO	Limiting Condition for Operation
LPCS	Low Pressure Core Spray
LTS	Leak Test Specialist
MDRFP	Motor-Driven Reactor Feed Pump
MOV	Motor-Operated Valve
MPC	Multi-Purpose Canister
MSPI	Mitigating Systems Performance Index
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
OWA	Operator Workaround
PARS	Publicly Available Records
PI	Performance Indicator
PMT	Post-Maintenance Testing
RCIC	Reactor Core Isolation Cooling
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RR	Reactor Recirculation
SBLC	Standby Liquid Control
SDP	Significance Determination Process
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
WO	Work Order
WS	Service Water

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Sincerely,

/RA/

Kenneth Riemer, Chief
Branch 2
Division of Reactor Projects

Docket Nos. 50-373; 50-374
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Letter to C. Pardee from K. Riemer dated October 30, 2009

SUBJECT: LASALLE COUNTY STATION, UNITS 1 AND 2
NRC INTEGRATED INSPECTION REPORT 05000373/2009004;
05000374/2009004

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