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

May 11, 2011

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

SUBJECT: LASALLE COUNTY STATION, UNITS 1 AND 2, NRC INTEGRATED INSPECTION REPORT 05000373/2011002; 05000374/2011002; 07200070/2011001

Dear Mr. Pacilio:

On March 31, 2011, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your LaSalle County Station, Units 1 and 2. The enclosed report documents the results of this inspection, which were discussed on Wednesday, April 13, 2011, with the Site Vice President, Mr. David Rhoades, and other members of your staff.

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

Based on the results of this inspection, three NRC-identified findings of very low safety significance were identified. The findings involved violations of NRC requirements. However, because of their very low safety significance, and because the issues were entered into your corrective action program, the NRC is treating the issues as non-cited violations (NCVs) in accordance with Section 2.3.2 of the NRC Enforcement Policy.

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

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

Sincerely,

/RA/

Kenneth Riemer, Chief
Branch 2
Division of Reactor Projects

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

Enclosure: Inspection Report 05000373/2011002; 05000374/2011002; 07200070/2011001
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	05000373; 05000374; 07200070
License Nos:	NPF-11; NPF-18
Report No:	05000373/2011002; 05000374/2011002; 07200070/2011001
Licensee:	Exelon Generation Company, LLC
Facility:	LaSalle County Station, Units 1 and 2
Location:	Marseilles, IL
Dates:	January 1, 2011, through March 31, 2011
Inspectors:	R. Ruiz; Senior Resident Inspector F. Ramírez, Acting Senior Resident Inspector A. Scarbeary, Acting Resident Inspector N. Shah, Region III Project Engineer M. Learn, Region III Reactor Engineer, MCID, DNMS R. Edwards, Region III Reactor Engineer, MCID, DNMS L. Rodriguez, Region III Reactor Engineer, MCID, DNMS J. Bozga, Reactor Inspector, DRS J. Neurauter, Senior Reactor Inspector, DRS M. Mitchell, Region III Health Physics Inspector, DRS 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 05000373/2011-002, 05000374/2011-002, 07200070/2011-001; 1/1/2011 – 3/31/2011; LaSalle County Station, Units 1 & 2; Maintenance Risk Assessments and Emergent Work Control; and Operability Evaluations.

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

A. NRC-Identified and Self-Revealed Findings

Cornerstone: Mitigating Systems

- Green. A finding of very low safety significance and associated NCV of 10 CFR 50.65(a)(4), Maintenance Rule, was identified by inspectors for the licensee's failure to implement all necessary prescribed risk management actions during a Unit 2 Reactor Core Isolation Cooling (RCIC) system maintenance window. Specifically, the licensee failed to post protected equipment signs for the Unit 2 systems whose unavailability would have taken the unit into a Red risk condition. The licensee entered this issue into their corrective action program (CAP).

  The inspectors determined that this performance deficiency is a finding and greater than minor because the licensee failed to implement prescribed compensatory measures of posting signs and barricades to protect the high pressure core spray (HPCS) equipment during the RCIC work window, hence degrading the HPCS safety function during this time; which is similar to Example 7.g in IMC 0612, Appendix E. The inspectors performed a Phase 1 screening with assistance from the Regional Senior Reactor Analyst (SRA) using IMC 0609, Appendix K, "Maintenance Risk Assessment and Risk Management Significance Determination Process," Flowchart 2, "Assessment of Risk Management Actions." The calculated change in Incremental Core Damage Probability (ICDP), or actual increase in risk during this work window, was 5.7×10^{-9}, and the incremental large early release probability (ILERP), was 3.3×10^{-10}. In accordance with Flowchart 2, since the ICDP was less than 1×10^{-6} and the ILERP was less than 1×10^{-7}, the finding screened as Green. This finding has a cross-cutting aspect in the area of Human Performance, Work Practices, because the licensee failed to conduct first and second verifications and use independent peer checks or other human error prevention techniques when evaluating risk-significant and/or Technical Specification (TS)-related activities, which led to the missed postings for the protected pathway equipment (H.4(a)). (Section 1R13)

Cornerstone: Barrier Integrity

- Green. A finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion V, Instructions, Procedures, and Drawings, was identified by the inspectors for the licensee's failure to follow steps 3.6 and 3.7 of procedure CC-AA-201, Revision 8, "Plant Barrier Control Program." Specifically, two airlock doors were opened simultaneously for a period of time sufficient to allow reactor building air pressure to surpass the TS allowed value for operability of secondary containment. The licensee entered this issue into its CAP as action requests (ARs) 1182255 and 1195987, and, at the time of this report, was in the process of conducting an apparent cause evaluation to determine the causes of the occurrence and to develop corrective actions.

  The finding was determined to be more than minor because it was associated with the Barrier Integrity Cornerstone attribute of configuration control and affected the cornerstone objective of providing reasonable assurance that physical design barriers (secondary containment) protect the public from radionuclide releases caused by accidents or events. The inspectors performed a Phase 1 SDP review of this finding using the guidance provided in IMC 0609, and a Region III SRA continued the risk assessment using IMC 0609, Appendix H, "Containment Integrity Significance Determination Process." For Unit 1, since an open pathway existed to the environment from the secondary containment, the SRA performed a Phase 2 SDP analysis using the Appendix H guidance. For Unit 2, the SRA performed a Phase 1 SDP analysis using Figure 6.2, "Road Map for LERF [Large Early Release Frequency]-based Risk Significance Evaluation for Type B Findings at Shutdown." The SRA concluded that the total risk associated with this finding is very low and best characterized as Green. This finding has a cross-cutting aspect in the area of Human Performance, Work Control, because the licensee did not appropriately coordinate work activities by incorporating actions to address the impact of the work on different job activities, and the need for work groups to communicate, coordinate, and cooperate with each other during activities in which interdepartmental coordination is necessary to assure plant and human performance (H.3(b)). (Section 1R15)

Cornerstone: Other Findings

- SL IV. A Severity Level IV NCV of 10 CFR 50.72(b)(3)(v) was identified by the inspectors for the licensee's failure to report an event or condition that could have prevented the fulfillment of the secondary containment's safety functions, which are relied upon to control the release of radioactive material. Specifically, the licensee had not properly controlled the opening of two airlock doors that served as a boundary to maintain the ventilation envelope of the reactor building. The licensee entered this issue into its CAP as ARs 1182255 and 1195987, and, at the time of this report, was in the process of conducting an apparent cause evaluation to determine the causes of the occurrence and to develop corrective actions.

  Violations of 10 CFR 50.72 are considered to be violations that potentially impact the regulatory process and are dispositioned using the traditional enforcement process instead of the Reactor Oversight Process SDP. As such, a cross-cutting aspect was not assigned to this violation. (Section 1R15)

B. Licensee-Identified Violations

No violations were identified.

REPORT DETAILS

Summary of Plant Status

Unit 1

The unit began the inspection period operating at full power. On February 1, 2011, the reactor scrammed due to a failure of a high voltage bushing in the 1W main power transformer (MPT). Following repairs, the licensee attempted to restart Unit 1 on February 9, 2011, but had to shut down once more after identifying a steam leak originating from a valve associated with the Reactor Core Isolation Cooling (RCIC) system. The licensee repaired the leak and subsequently restarted the unit, synchronizing to the grid on February 11, 2011. Full power was achieved on February 13, 2011. On March 27, 2011, power was reduced to 65 percent for control rod pattern adjustments and scram time testing of control rods. Unit 1 was restored to 100 percent power the same day and remained there for the rest of the inspection period.

Unit 2

The unit began the inspection period operating at full power. On January 9, 2011, power was reduced to approximately 70 percent for control rod pattern adjustment and was restored to 100 percent that same day. On January 23, 2011, the unit commenced coastdown for refueling outage (RFO) L2R13. Following completion of the outage, the unit was restarted and synchronized to the grid on March 7, 2011. Full power was achieved on March 9, 2011. Lastly, on March 13, 2011, power was reduced to 90 percent in order to perform final testing for the implementation of measurement uncertainty recapture uprate (an increase in overall power produced). After testing was completed on March 14, 2011, the unit achieved its new full power value of 3546 megawatts thermal. The unit remained at full power for the rest of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 Readiness for Impending Adverse Weather Condition – Heavy Snowfall Conditions

a. Inspection Scope

On February 1, 2011, a winter weather advisory was issued for expected snow squalls in the area of the plant. The inspectors observed the licensee's preparations and planning for the significant winter weather potential. The inspectors reviewed licensee procedures and discussed potential compensatory measures with control room personnel. The inspectors focused on plant management's actions for implementing the station's procedures for ensuring adequate personnel availability for safe plant operation and emergency response. The inspectors conducted a site walkdown including walkdowns of various plant structures and systems to check for maintenance work or other apparent deficiencies that could affect system operations during the predicted significant weather. The inspectors also reviewed CAP items to verify that the licensee was identifying adverse weather issues at an appropriate threshold and entering them into their CAP in accordance with station procedures. Specific documents reviewed during this inspection are listed in the Attachment to this report.

This inspection constituted one readiness for impending adverse weather condition sample as defined in Inspection Procedure (IP) 71111.01-05.

b. Findings

No findings 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 2 RCIC system during condensate storage tank inspection;
- Unit 2 HPCS system during divisional work window during L2R13; and
- Unit 2 instrument nitrogen system following new installation during L2R13.

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, Updated Final Safety Analysis Report (UFSAR), TS requirements, outstanding work orders (WOs), condition reports, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the 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 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 elevation 786’ of the reactor building (Fire Zone 2D);
- Unit 2 elevation 786’ of the reactor building (Fire Zone 3D);
- Unit 2 heater bay (Fire Zone 5B2);
- Unit 2 primary containment (Fire Zone 3J);
- Unit 2 turbine-driven reactor feed pump rooms (Fire Zone 5A2) and
- Unit 2 motor-driven reactor feed pump room (Fire Zone 5B10).

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 six quarterly fire protection inspection samples as defined in IP 71111.05-05.

b. Findings

No findings 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 2B residual heat removal (RHR) system 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 verified that test acceptance criteria considered differences between test conditions, design conditions, and testing conditions. 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 were identified.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Resident Inspector Quarterly Review (71111.11Q)

a. Inspection Scope

On February 1, 2011, the inspectors observed a crew of licensed operators in the plant's simulator during licensed operator requalification examinations to verify that operator performance was adequate, evaluators were identifying and documenting crew performance problems, and training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

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

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

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

b. Findings

No findings were identified.

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 systems:

- Units 1 and 2 annunciator system; and
- Units 2 failed fuel action plan.

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

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

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

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

b. Findings

No findings were identified.

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

.1 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

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

- Unit 2 Yellow online risk during RCIC maintenance;
- Unit 2 Yellow shutdown safety risk during L2R13; and
- Units 1 and 2 Yellow online risk during switchyard maintenance.

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 three samples as defined in IP 71111.13-05.

b. Findings

Failure to Post Protected Pathway Signs for a Red Risk Path System

Introduction: A finding of very low safety significance and associated NCV of 10 CFR 50.65(a)(4), Maintenance Rule, was identified by the inspectors for the licensee's failure to implement all necessary prescribed risk management actions during a Unit 2 RCIC system maintenance window. Specifically, the licensee failed to post protected equipment signs for Unit 2 systems whose unavailability would have taken the unit into a Red risk condition.

Description: On January 11, 2011, during a protected pathway walkdown for the Unit 2 RCIC work window, the inspectors identified that the Division III 2B Diesel Generator (DG), the HPCS-dedicated diesel, and its associated room ventilation were not labeled with the appropriate protected equipment signs in the field. Further reviews revealed that this equipment was also not listed in the Protected Equipment Log (OP-LA-101-111-1002, Att. B), kept by operations staff in the control room. These two systems were, however, listed in the Paragon online risk assessment program as Red risk equipment that should be protected according to the station's Protected Equipment Program procedure, OP-AA-108-117, during the RCIC work window.

According to interviews of the operations staff involved, a Senior Reactor Operator (SRO-1) was conducting an advance review of the upcoming week's activities and utilized a list of protected path equipment on a sheet that was previously prepared and included in the WO package provided. These sheets are kept on an internal plant computer drive where they can be easily accessed and used for maintenance work windows. The inspectors noted from the interview that SRO-1 did not verify the information on the protected path equipment sheet with the Paragon program.

Additionally, the day before the work window commenced, the dayshift Shift Manager and another SRO (SRO-2) discussed upcoming protected pathways that would be implemented the following day. They were unable to locate the protected path sheet that should have been included with the Unit 2 RCIC system WO package and, as a result, printed a new sheet from the internal computer drive. The inspectors noted from the interview that the SRO-2 was distracted while trying to perform a Paragon risk assessment of the work window and did not identify that the 2B DG and associated support system components were not listed on the sheet. As a result, these components were not added to the protected equipment sheet that was utilized by the field operators to post protected equipment signs in the plant, and were not posted.

The inspectors agreed with the conclusions of the licensee's Apparent Cause Report, which attributes the performance deficiency to inadequate technical human performance, i.e., the failure to conduct first and second verifications, and independent peer checks, on risk-significant and/or TS-related activities. The licensee further concluded that inadequate technical rigor was applied when conducting reviews of the protected path list, which were, in part, recognized as a lack of a requirement for the use of the Paragon risk assessment program when performing a peer review of upcoming maintenance work windows.

Analysis: The inspectors determined that the licensee's failure to post protected equipment signs for Unit 2 systems whose unavailability would have taken the unit into a Red risk condition was contrary to the station's Protected Equipment Program procedure, OP-AA-108-117, and is a performance deficiency. The inspectors determined that this finding is more than minor because the licensee failed to implement prescribed compensatory measures of posting signs and barricades to protect the HPCS equipment during the RCIC work window. Noteworthy was that while the licensee probabilistic risk assessment staff did identify the HPCS system and components as needing to be protected during the work planning stage, plant operators later used information from a separate (older) risk evaluation that did not include the HPCS equipment as needing to be protected. Hence, workers started to perform RCIC work without the HPCS equipment being protected until it was identified by the inspectors.

This finding is similar to Example 7.g. provided in IMC 0612, Appendix E, because the HPCS safety function was degraded by failing to protect the system. The inspectors used IMC 0609, Appendix K, "Maintenance Risk Assessment and Risk Management Significance Determination Process," Flowchart 2, "Assessment of Risk Management Actions," to analyze the finding. A regional SRA reviewed the licensee's risk significance evaluation of this issue. The total exposure time for this deficiency was conservatively assumed to be 37 hours, the entire duration of the RCIC work window. The licensee determined the actual duration of the condition to be more than 12 hours, but had the NRC not identified the deficiency, it would have existed the entire duration of the work window.

The inspectors performed a Phase 1 screening using IMC 0609, Appendix K, "Maintenance Risk Assessment and Risk Management Significance Determination Process," with assistance from the Regional SRA. The SRA also evaluated the licensee's independent risk significance evaluation performed by the licensee's risk expert. In Appendix K, Flowchart 2, "Assessment of Risk Management Actions," the licensee calculated the ICDP, or actual increase in risk during this work window, as 5.7×10^{-9}, and the ILERP, as 3.3×10^{-10}. The total exposure time used when calculating these values was conservatively assumed to be equal to the entire RCIC work window time of 37 hours. The SRA agreed with the licensee's calculation method and conclusions for these values. In accordance with Flowchart 2, because the ICDP was less than 1×10^{-6} and the ILERP was less than 1×10^{-7}, the finding screened as Green.

This finding has a cross-cutting aspect in the area of Human Performance, Work Practices, because the licensee failed to use human error prevention techniques, such as self and peer checking, commensurate with the risk of the assigned task (H.4(a)). Specifically, the failure to conduct first and second verifications, and independent peer checks, or other human error prevention techniques on risk-significant and/or TS-related activities led to the missed postings for protected pathway equipment.

Enforcement: The requirements for monitoring the effectiveness of maintenance at nuclear power plants, as described in 10 CFR 50.65(a)(4), states, in part, that "before performing maintenance activities, the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities."

Contrary to the above, on January 11, 2011, the licensee failed to implement all necessary prescribed risk management actions during the Unit 2 RCIC system maintenance window. As a result, the increase in risk associated with making the

HPCS system inoperable while the RCIC system was unavailable was not adequately accounted for. The licensee entered this issue into their CAP as AR 1161438 and AR 1166596. Corrective actions planned and completed by the licensee included revising the LaSalle Operations Philosophy Handbook to include additional expectations for protected pathways, including an update to the protected pathway attachment for a peer review using the Paragon program; updating the protected pathway electronic files with peer reviews; and conducting training for the operators on using Paragon when determining what equipment to protect when a system or component is out-of-service. Because the licensee has entered the issue into their CAP and the finding is of very low safety significance, this violation of 10 CFR 50.65(a)(4) is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000373/2011002-01; 05000374/2011002-01, Failure to Post Protected Pathway Signs for a Red Risk Path System).

1R15 Operability Evaluations (71111.15)

.1 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following issues:

- Unit 1 low pressure core spray (LPCS) water leg pump degraded performance;
- Unit 1 Division 2 switchgear operability during failure of switchgear ventilation system dampers;
- Unit 1 ‘E’ safety relief valve (SRV) non-automatic depressurization system (non-ADS) accumulator check valve degraded performance;
- Unit 2 air void in ‘B’ RHR system piping;
- Unit 2 RCIC piping outside of the pressure drop test acceptance criteria; and
- Units 1 and 2 shared secondary containment inadvertent inoperability.

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

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

b. Findings

(1) Failure to Follow Plant Barrier Control Process Caused Secondary Containment to Become Inoperable

Introduction: A finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified by the inspectors for the licensee's failure to follow steps 3.6 and 3.7 of procedure CC-AA-201, Revision 8, "Plant Barrier Control Program." Specifically, evidence has shown that two adjacent airlock doors were opened simultaneously for a period of time sufficient to cause reactor building air pressure to surpass the Technical Specification (TS) allowed value for operability.

Description: Through a review of operator logs, the inspectors noted that on March 3, 2011 at 9:28 a.m. while Unit 1 was in Mode 1 and Unit 2 was in Refueling Mode in the process of loading fuel into the reactor, the reactor building-to-outside differential pressure was observed to be in excess of -0.25 inches of water column—the maximum value allowed by TS Surveillance Requirement 3.6.4.1.1. As a result, TS Limiting Condition for Operability (LCO) 3.6.4.1 Condition A was entered for both units, since the reactor building is a shared structure, and core alterations were immediately halted. The condition was immediately evaluated by the licensee and entered into the CAP as AR 1182255 entitled "Loss of Secondary Containment During Fuel Movement." Shortly thereafter, secondary containment pressure returned to the acceptable range, the licensee declared secondary containment operable at 9:42 a.m., and fuel moves recommenced by 10:00 a.m. on Unit 2.

The aforementioned AR documented the licensee's decision making process for troubleshooting the pressure anomaly. The licensee identified that there were two potential causes for the loss of differential pressure in the reactor building: a malfunctioning ventilation system, or a breach of the ventilation envelope of the reactor building. Instrument maintenance personnel were immediately dispatched to the operating reactor building ventilation panel and found no evidence that the ventilation system was malfunctioning. Additionally, due to previous operating experience at LaSalle which has shown that the simultaneous opening of two reactor building airlock doors causes similar effects as those seen on secondary containment pressure, the licensee promptly dispatched operations personnel to observe various interlocked door locations to check for openings. Although no doors were reportedly found open at the time, the licensee nevertheless concluded that since the reactor building ventilation system apparently functioned as designed, the loss of operability of the secondary containment was due to the opening of two interlock doors.

The inspectors conducted interviews with the site engineering staff responsible for secondary containment ventilation as well as for the design and operation of the door interlock systems. Through these discussions and the review of the reactor building ventilation system flow output traces for the period of time in question, the inspectors noted that the evidence pointed towards a cause of two interlocked airlock doors being held open by plant workers. This opening would have represented an unmonitored pathway to the environment from the reactor building in the case of a fuel handling accident on Unit 2 or a design basis accident on the operating reactor, Unit 1. Additionally, it was explained to the inspectors by the engineering staff that the open pathway existed for approximately 15 minutes before being closed. The period of time that the building pressure actually exceeded the TS limit was approximately 3 minutes of that 15 minute window.

The inspectors noted that the airlock doors are designed with an interlock system to prevent the inadvertent opening of two adjacent airlock doors at the same time. However, due to design limitations on the actuation speed of the locking mechanism, two interlocked doors can actually be opened simultaneously if they are opened at precisely the same moment. The inspectors further noted that, through discussions with engineering staff, an inadvertent simultaneous opening of two doors followed by their rapid closure would not have caused the secondary containment pressure to reach the threshold of inoperability; instead the step change in ventilation air flow was shown to be caused by an opening in the building envelope for approximately 15 minutes, as previously mentioned.

As a result, the inspectors concluded that plant workers failed to follow steps 3.6 and 3.7 of procedure CC-AA-201, Revision 8, "Plant Barrier Control Program," by defeating the reactor building door interlock mechanism and failing to close the open pathway immediately.

Step 3.6 of procedure CC-AA-201 states that "Doors SHOULD always be placed in their required position after use in order to be considered operable. Doors that are found not to be in their required position SHOULD be placed in their required position." The note for Step 3.6 states "…Primary and Secondary Containment access doors are subject to limitations imposed by Tech Specs. Provided one door in an interlock / airlock is closed when the interlocked door is opened for access / egress of personnel and equipment, and Tech Spec requirements are met, the barrier is **not** impaired, the provisions of Step 3.7 do **not** apply, and **no** PBI [plant barrier impairment] is required."

Step 3.7 of that same procedure states, in part, that "…doors designed for personnel ingress/egress MAY be opened without a PBI PERMIT provided all of the following conditions are met: the door is closed following the ingress/egress of personnel & equipment; the door is not blocked or tied open; … [and] site-specific restrictions including those from site-specific documents such as Tech Specs are met."

Analysis: The inspectors determined that two airlock doors being opened simultaneously for a period of time sufficient to allow reactor building air pressure to surpass the TS allowed value for operability was contrary to steps 3.6 and 3.7 of procedure CC-AA-201, Revision 8, "Plant Barrier Control Program," and was a performance deficiency.

The finding was determined to be more than minor because it was associated with the Barrier Integrity Cornerstone attribute of Configuration Control and affected the cornerstone objective of providing reasonable assurance that physical design barriers (secondary containment) protect the public from radionuclide releases caused by accidents or events. Specifically, opening two reactor building airlock doors at the same time did not maintain reasonable assurance that the secondary containment would be capable of performing its safety function in the event of a reactor accident or fuel handling accident.

The inspectors performed a Phase 1 SDP review of this finding using the guidance provided in IMC 0609, Attachment 4, "Initial Screening and Characterization of Findings." In accordance with Table 3b, "SDP Phase 1 Screening Worksheet for

Initiating Events, Mitigating Systems, and Barriers Cornerstones," the inspectors determined that the cornerstone best reflecting the dominant risk was the Containment Barrier Cornerstone. Per Question 3, the finding represented an actual open pathway in the physical integrity of reactor containment. A Region III SRA continued the risk assessment using IMC 0609, Appendix H, "Containment Integrity Significance Determination Process."

For both units, the SRA determined that this was a "Type B" finding (i.e., had no impact on the determination of delta core damage frequency). The SRA reviewed the SSCs listed in Table 6.1, "Phase 1 Screening-Type B Findings at Full Power," to determine if the finding was associated with an SSC(s) important to LERF. LaSalle is a Boiling Water Reactor-5 with a Mark II Containment. The SSCs listed in Table 6.1 for Mark II Containments included containment penetration seals, containment isolation valves (CIVs), and vent and purge systems, none of which applied in this case.

For Unit 1, since an open pathway existed to the environment from the secondary containment, the SRA performed a Phase 2 SDP analysis using the Appendix H guidance. Table 6.2, "Phase 2 Risk Significance -Type B Findings at Full Power," indicated that all findings less than 3-day duration are very low risk significance.

For Unit 2, the SRA performed a Phase 1 SDP analysis using Figure 6.2, "Road Map for LERF-based Risk Significance Evaluation for Type B Findings at Shutdown." This figure indicated that Type B shutdown findings, when the reactor vessel water level is greater than or equal to that required for movement of irradiated fuel assemblies, are of very low risk significance.

Based on the above, the SRA concluded that the total risk associated with this finding is very low and best characterized as Green.

This finding has a cross-cutting aspect in the area of Human Performance, Work Control because the licensee did not appropriately coordinate work activities by incorporating actions to address the impact of the work on different job activities, and the need for work groups to communicate, coordinate, and cooperate with each other during activities in which interdepartmental coordination is necessary to assure plant and human performance. Specifically, if the plant workers that opened the doors for 15 minutes would have communicated their desire to create the open pathway for equipment egress purposes, then the Plant Barrier Control Program would have been referenced and workers would not have been allowed to do so. This would have prevented the procedure violation (H.3(b)).

Enforcement: 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings.

Contrary to the above, on March 3, 2011, the licensee failed to accomplish configuration control activities that directly affected the ability of the secondary containment to perform its safety function—an activity affecting quality—in accordance with prescribed procedures. Specifically, plant workers' failure to follow steps 3.6 and 3.7 of procedure CC-AA-201, Revision 8, entitled "Plant Barrier Control Program," was considered a violation of Criterion V. Because this violation was of very low safety significance and it was entered into the licensee's CAP as ARs 1182255 and 1195987, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 05000373/2011002-02; 05000374/2011002-02, Failure to Follow Plant Barrier Control Process Caused Secondary Containment to Become Inoperable).

(2) Failure to Make a Required 10 CFR 50.72 Report for an Inoperable Secondary Containment

Introduction: A Severity Level IV NCV of 10 CFR 50.72 (b)(3)(v) was identified by the inspectors for the failure of the licensee to make an eight-hour report to the NRC for a condition that, at the time of discovery, could have prevented secondary containment from fulfilling its safety function. Specifically, when two adjacent reactor building airlock doors were opened concurrently, the ability of the system to perform its specified safety function could no longer be assured.

Description: On March 3, 2011, reactor building-to-outside differential pressure was observed to be in excess of -0.25 inches of water column—the maximum value allowed by TS Surveillance Requirement 3.6.4.1.1. As a result, TS LCO 3.6.4.1, Condition A, was entered for both units.

The inspectors agreed with the licensee's conclusion that since the ventilation system operated as expected, the condition could have only been caused by the concurrent opening of two adjacent reactor building airlock doors by plant workers. The open, unmonitored, unfiltered pathway that was created by this event essentially defeated the safety function of the secondary containment building by allowing a bypass pathway for radioactive effluent to escape in the event of a design basis accident on the operating unit or a fuel handling accident on the refueling unit. See Section 1R15.(1) above for more details.

LaSalle's TSs define OPERABLE-OPERABILITY by stating "A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s)…" Therefore, by applying LaSalle's definition of OPERABILITY, when a system is declared INOPERABLE it is incapable of performing its specified safety function(s). When secondary containment was declared inoperable for failing to maintain $\geq$ -0.25 inches of vacuum, it was, by definition, deemed incapable of performing its specified safety function.

During the review of this issue, the inspectors analyzed the challenge to the configuration of the reactor building ventilation envelope through a snapshot- in-time approach. Specifically, at the moment that both doors were opened simultaneously and not immediately shut, the operability of the secondary containment would have been brought into question, regardless of the building differential pressure being above the TS allowed value at the time. The reason for this is due to TS Surveillance Requirement (SR) 3.0.1, which states, in part, that, "…failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO." In the snapshot-in-time where an open pathway existed in the secondary containment boundary, SR 3.6.4.1.3, which requires that the secondary containment be drawn down to$\geq$ -0.25 inches of vacuum water gauge in $\leq$ 900 seconds using one standby gas treatment subsystem, may not have been able to have been satisfied. Therefore, in accordance with SR 3.0.1, the LCO may not have been met. This is further evidence that the operability of secondary containment and its ability to perform its safety function were not ensured when two redundant airlock doors were opened simultaneously.

In accordance with NUREG-1022, Event Reporting Guidelines – 10 CFR 50.72 and 50.73, Revision 2, an 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 control the release of radioactive material (the condition identified on March 3) should have been reported within eight hours to the NRC under 10 CFR 50.72(b)(3)(v). The licensee did not do so within the required timeframe.

Analysis: The inspectors determined that the failure to make a required eight-hour report to the NRC for the loss of the secondary containment safety function was not in accordance with 10 CFR 50.72(b)(3)(v) and was a performance deficiency. Because violations of 10 CFR 50.72 are considered to be violations that potentially impact the regulatory process, they are dispositioned using the traditional enforcement process instead of the Reactor Oversight Process (ROP) SDP. As such, a cross-cutting aspect was not assigned to this violation. Per the NRC Enforcement Policy, Section 6.0, "Violation Examples," a failure to make a report required by 10 CFR 50.72 is categorized as a Severity Level IV violation.

Enforcement: Title 10 CFR 50.72 (b)(3)(v)(C) and (D) requires, in part, that licensees report any event or condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to (C) control the release of radioactive material, and (D) mitigate the consequences of an accident.

Contrary to the above, on March 3, 2011, the licensee failed to report a condition that could have prevented the fulfillment of a safety function. Specifically, in the case of secondary containment, when it was declared inoperable for failing to maintain greater than or equal to 0.25 inches of vacuum due to an open pathway, the secondary containment, by definition, was deemed incapable of performing its specified safety function. Because this violation was not repetitive or willful, and was entered into the licensee's CAP as AR 01182255 and AR 01195987, this violation is being treated as a Severity Level IV NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 05000373/2011002-03; 05000374/2011002-03, Failure to Make a Required 10 CFR 50.72 Report for an Inoperable Secondary Containment).

1R18 Plant Modifications (71111.18)

.1 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the following modifications:

- Unit 2 rod control management system (RCMS) circuit modification to prevent unintended control rod motion (Permanent); and
- Unit 2 multiple spurious operations (MSO) modification installation (Permanent).

The inspectors reviewed the configuration changes and associated 10 CFR 50.59 safety evaluation screening against the design basis, the UFSAR, and the TS, as applicable, to verify that the modification did not affect the operability or availability of the affected systems. The inspectors, as applicable, observed ongoing and completed work activities to ensure that the modifications were installed as directed and consistent with the design control documents; the modifications operated as expected; post-modification testing adequately demonstrated continued system operability, availability, and reliability; and that operation of the modifications did not impact the operability of any interfacing systems. As applicable, the inspectors verified that relevant procedure, design, and licensing documents were properly updated. Lastly, the inspectors discussed the plant modification with operations, engineering, and training personnel to ensure that the individuals were aware of how the operation with the plant modification in place could impact overall plant performance. Documents reviewed in the course of this inspection are listed in the Attachment to this report.

This inspection constituted two permanent modification samples as defined in IP 71111.18-05.

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19)

.1 Post-Maintenance Testing

a. Inspection Scope

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

- Unit 2 HPCS valve test following MSO modification installation;
- Unit 2 Division 1 water leg pump run following replacement;
- Unit 1 B reactor recirculation (RR) pump run following seal replacement;
- Unit 1 E SRV test following replacement;
- Unit 1 RCIC steam inboard isolation bypass valve test following repairs; and
- Unit 2 Division 2 DG run following voltage regulator replacement.

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 TSs, 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 six post-maintenance testing samples as defined in IP 71111.19-05.

b. Findings

No findings were identified.

1R20 Outage Activities (71111.20)

.1 Refueling Outage Activities

a. Inspection Scope

The inspectors reviewed the Outage Safety Plan (OSP) and contingency plans for the Unit 2 RFO, conducted from February 14, 2011, to March 8, 2011, to confirm that the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth. During the RFO, the inspectors observed portions of the shutdown and cooldown processes and monitored licensee controls over the outage activities listed below. Documents reviewed during the inspection are listed in the Attachment to this report.

- Licensee configuration management, including maintenance of defense-in-depth commensurate with the OSP for key safety functions and compliance with the applicable TS when taking equipment out-of-service;
- Implementation of clearance activities and confirmation that tags were properly hung and equipment appropriately configured to safely support the work or testing;
- Installation and configuration of reactor coolant pressure, level, and temperature instruments to provide accurate indication, accounting for instrument error;
- Controls over the status and configuration of electrical systems to ensure that TS and OSP requirements were met, and controls over switchyard activities;
- Monitoring of decay heat removal processes, systems, and components;
- Controls to ensure that outage work was not impacting the ability of the operators to operate the spent fuel pool cooling system;
- Reactor water inventory controls including flow paths, configurations, and alternative means for inventory addition, and controls to prevent inventory loss;
- Controls over activities that could affect reactivity;
- Maintenance of secondary containment as required by TS;
- Refueling activities, including fuel handling and sipping to detect fuel assembly leakage;
- Startup and ascension to full power operation, tracking of startup prerequisites, walkdown of the drywell (DW) (primary containment) to verify that debris had not been left which could block emergency core cooling system (ECCS) suction strainers, and reactor physics testing;
- Licensee identification and resolution of problems related to RFO activities.

This inspection constituted one RFO sample as defined in IP 71111.20-05.

b. Findings

No findings were identified.

.2 Other Outage Activities

a. Inspection Scope

The inspectors evaluated outage activities for an unscheduled outage on Unit 1 that began on February 1, 2011, and continued through February 11, 2011. The inspectors reviewed activities to ensure that the licensee considered risk in developing, planning, and implementing the outage schedule. The inspectors observed or reviewed the reactor shutdown and cooldown, outage equipment configuration and risk management, electrical lineups, selected clearances, control and monitoring of decay heat removal, control of containment activities, startup and heat-up activities, and identification and resolution of problems associated with the outage. In addition, the inspectors observed maintenance activities accomplished during the outage and the Plant Operations Review Committee startup meeting to ensure event issues were addressed prior to restart.

The outage began on February 1 following an automatic scram due to a fault at the C phase bushing of the Unit 1 West Main Power Transformer (MPT).
The circumstances of the trip were summarized in NRC event notice 46582 and in Section 4OA3 of this inspection report. Following repairs of the 1W MPT, the licensee attempted to restart Unit 1 on February 9, but had to shut down once more after identifying a steam leak originating from a valve associated with the Reactor Core Isolation Cooling (RCIC) system. The circumstances of the steam leak were described in NRC event notice 46605. The licensee repaired the leak on February 10 and subsequently restarted the unit, returning to full power on February 13.

This inspection constituted one other outage sample as defined in IP 71111.20-05.

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22)

.1 Surveillance Testing

a. Inspection Scope

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

- 0 DG auxiliaries test (Routine);
- Unit 2 integrated Division II response time test (Routine);
- Unit 2 secondary containment leak rate test (Routine);
- Unit 1 A RHR and RHR service water pump and valve tests (IST);
- Unit 2 A RHR valves local leak rate testing and quick look testing (CIV);

- Unit 2 main steam isolation valve testing (CIV); and
- Unit 2 reactor vessel leakage (hydrostatic) test (RCS).

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

- did preconditioning occur;
- were the effects of the testing adequately addressed by control room personnel or engineers prior to the commencement of the testing;
- were acceptance criteria clearly stated, demonstrated operational readiness, and consistent with the system design basis;
- plant equipment calibration was correct, accurate, and properly documented;
- as-left setpoints were within required ranges; and the calibration frequency was in accordance with TSs, the UFSAR, procedures, and applicable commitments;
- measuring and test equipment calibration was current;
- test equipment was used within the required range and accuracy; applicable prerequisites described in the test procedures were satisfied;
- test frequencies met TS requirements to demonstrate operability and reliability; tests were performed in accordance with the test procedures and other applicable procedures; jumpers and lifted leads were controlled and restored where used;
- test data and results were accurate, complete, within limits, and valid;
- test equipment was removed after testing;
- where applicable for IST activities, testing was performed in accordance with the applicable version of Section XI, American Society of Mechanical Engineers (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 three routine surveillance testing samples, one IST sample, one RCS leak detection inspection sample, and two CIV samples as defined in IP 71111.22, Sections -02 and -05.

b. Findings

No findings were identified.

2. RADIATION SAFETY

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

This inspection constituted a partial sample as defined in IP 71124.02 05.

.1 Inspection Planning (02.01)

a. Inspection Scope

The inspectors reviewed all licensee performance indicators (PIs) for the occupational exposure cornerstone for followup. The inspectors reviewed the results of radiation protection (RP) program audits (e.g., licensee's quality assurance audits or other independent audits). The inspectors reviewed any reports of operational occurrences related to occupational radiation safety since the last inspection. The inspectors reviewed the results of the audit and operational report reviews to gain insights into overall licensee performance.

b. Findings

No findings were identified.

.2 Radiological Hazard Assessment (02.02)

a. Inspection Scope

The inspectors determined if there have been changes to plant operations since the last inspection that may result in a significant new radiological hazard for onsite workers or members of the public. The inspectors evaluated whether the licensee assessed the potential impact of these changes and has implemented periodic monitoring, as appropriate, to detect and quantify the radiological hazard.

The inspectors reviewed the last two radiological surveys from selected plant areas and evaluated whether the thoroughness and frequency of the surveys where appropriate for the given radiological hazard.

The inspectors conducted walkdowns of the facility, including radioactive waste processing, storage, and handling areas to evaluate material conditions and performed independent radiation measurements to verify conditions.

The inspectors selected the following radiologically risk-significant work activities that involved exposure to radiation.

- Radiation Work Permit (RWP) 1001653 L2R13 DW SRV activities;
- RWP 1001666 L2R13 Under-Vessel Nuclear Instrumentation;
- RWP 1001667 L2R13 Control Rod Drive (CRD) Pull/Put;
- RWP 1001703 L2R13 Reactor Vessel Disassembly/Reassembly; and
- RWP 1001726 L2R13 Feed Water Heater Replacements.

For these work activities, the inspectors assessed whether the pre-work surveys performed were appropriate to identify and quantify the radiological hazard and to establish adequate protective measures. The inspectors evaluated the radiological survey program to determine if hazards were properly identified, including the following:

- identification of hot particles;
- the presence of alpha emitters;
- the potential for airborne radioactive materials, including the potential presence of transuranics and/or other hard-to-detect radioactive materials (This evaluation may include licensee planned entry into non-routinely entered areas subject to previous contamination from failed fuel.);
- the hazards associated with work activities that could suddenly and severely increase radiological conditions and that the licensee has established a means to inform workers of changes that could significantly impact their occupational dose; and
- severe radiation field dose gradients that can result in non-uniform exposures of the body.

The inspectors observed work in potential airborne areas and evaluated whether the air samples were representative of the breathing air zone. The inspectors evaluated whether continuous air monitors were located in areas with low background to minimize false alarms and were representative of actual work areas. The inspectors evaluated the licensee's program for monitoring levels of loose surface contamination in areas of the plant with the potential for the contamination to become airborne.

b. Findings

No findings were identified.

.3 Instructions to Workers (02.03)

a. Inspection Scope

The inspectors selected various containers holding non-exempt licensed radioactive materials that may cause unplanned or inadvertent exposure of workers, and assessed whether the containers were labeled and controlled in accordance with 10 CFR 20.1904, "Labeling Containers," or met the requirements of 10 CFR 20.1905(g), "Exemptions to Labeling Requirements."

The inspectors reviewed the following RWPs used to access high radiation areas and evaluated the specified work control instructions or control barriers.

- RWP 1001653 L2R13 DW SRV activities;
- RWP 1001666 L2R13 Under-Vessel Nuclear Instrumentation;
- RWP 1001667 L2R13 CRD Pull/Put;
- RWP 1001703 L2R13 Reactor Vessel Disassembly/Reassembly; and
- RWP 1001726 L2R13 Feed Water Heater Replacements.

For these RWPs, the inspectors assessed whether allowable stay times or permissible dose (including from the intake of radioactive material) for radiologically significant work under each RWP were clearly identified. The inspectors evaluated whether electronic personal dosimeter alarm set-points were in conformance with survey indications and plant policy.

The inspectors reviewed selected occurrences where a worker's electronic personal dosimeter noticeably malfunctioned or alarmed. The inspectors evaluated whether workers responded appropriately to the off-normal condition. The inspectors assessed whether the issue was included in the CAP and dose evaluations were conducted as appropriate.

For work activities that could suddenly and severely increase radiological conditions, the inspectors assessed the licensee's means to inform workers of changes that could significantly impact their occupational dose.

b. Findings

No findings were identified.

.4 Contamination and Radioactive Material Control (02.04)

a. Inspection Scope

The inspectors observed locations where the licensee monitors potentially contaminated material leaving the radiological control area and inspected the methods used for control, survey, and release from these areas. The inspectors observed the performance of personnel surveying and releasing material for unrestricted use and evaluated whether the work was performed in accordance with plant procedures and whether the procedures were sufficient to control the spread of contamination and prevent unintended release of radioactive materials from the site. The inspectors assessed whether the radiation monitoring instrumentation had appropriate sensitivity for the types of radiation present.

The inspectors reviewed the licensee's criteria for the survey and release of potentially contaminated material. The inspectors evaluated whether there was guidance on how to respond to an alarm that indicates the presence of licensed radioactive material.

The inspectors reviewed the licensee's procedures and records to verify that the radiation detection instrumentation was used at its typical sensitivity level based on appropriate counting parameters. The inspectors assessed whether or not the licensee has established a de facto "release limit" by altering the instrument's typical sensitivity through such methods as raising the energy discriminator level or locating the instrument in a high radiation background area.

The inspectors selected several sealed sources from the licensee's inventory records and assessed whether the sources were accounted for and verified to be intact.

The inspectors evaluated whether any transactions, since the last inspection, involving nationally tracked sources were reported in accordance with 10 CFR 20.2207.

b. Findings

No findings were identified.

.5 Radiological Hazards Control and Work Coverage (02.05)

a. Inspection Scope

The inspectors evaluated ambient radiological conditions (e.g., radiation levels or potential radiation levels) during tours of the facility. The inspectors assessed whether the conditions were consistent with applicable posted surveys, RWPs, and worker briefings.

The inspectors evaluated the adequacy of radiological controls, such as required surveys, RP job coverage (including audio and visual surveillance for remote job coverage), and contamination controls. The inspectors evaluated the licensee's use of electronic personal dosimeters in high noise areas as high radiation area monitoring devices.

The inspectors assessed whether radiation monitoring devices were placed on the individual's body consistent with licensee procedures. The inspectors assessed whether the dosimeter was placed in the location of highest expected dose or that the licensee properly employed an NRC-approved method of determining effective dose equivalent.

The inspectors reviewed the application of dosimetry to effectively monitor exposure to personnel in high radiation work areas with significant dose rate gradients.

The inspectors reviewed the following RWPs for work within airborne radioactivity areas with the potential for individual worker internal exposures.

- RWP 10011666; L2R13 Under-Vessel Nuclear Instrumentation As-Low-As-Is-Reasonably-Achievable (ALARA) Plan, Revision 0;
- RWP 10011667; L2R13 CRD Pull/Put ALARA Plan, Revision 0;
- RWP 10011703; L2R13 Reactor Vessel Disassembly/Reassembly ALARA Plan, Revision 3;
- RWP 10011726; L2R13 Feed Water Heater Replacements ALARA Plan, Revision 0; and
- RWP 10011666, L2R13 Under-Vessel Nuclear Instrumentation ALARA Plan, Revision 0.

For these RWPs, the inspectors evaluated airborne radioactive controls and monitoring, including potential for significant airborne levels (e.g., grinding, grit blasting, system breaches, entry into tanks, cubicles, and reactor cavities). The inspectors assessed barrier (e.g., tent or glove box) integrity and temporary high efficiency particulate air VENTILATION SYSTEM operation.

The inspectors examined the posting and physical controls for selected high radiation areas and very high radiation areas to verify conformance with the occupational Performance Indicator (PI).

b. Findings

No findings were identified.

.6 Radiation Worker Performance (02.07)

a. Inspection Scope

The inspectors observed radiation worker performance with respect to stated RP work requirements. The inspectors assessed whether workers were aware of the radiological conditions in their workplace and the RWP controls/limits in place, and whether their performance reflected the level of radiological hazards present.

b. Findings

No findings were identified.

.7 Radiation Protection Technician Proficiency (02.08)

a. Inspection Scope

The inspectors observed the performance of the RP technicians with respect to all RP work requirements. The inspectors evaluated whether technicians were aware of the radiological conditions in their workplace and the RWP controls/limits, and whether their performance was consistent with their training and qualifications with respect to the radiological hazards and work activities.

b. Findings

No findings were identified.

.8 Problem Identification and Resolution (02.09)

a. Inspection Scope

The inspectors evaluated whether problems associated with radiation monitoring and exposure control were being identified by the licensee at an appropriate threshold and were properly addressed for resolution in the licensee's CAP. The inspectors assessed the appropriateness of the corrective actions for a selected sample of problems documented by the licensee that involve radiation monitoring and exposure controls. The inspectors assessed the licensee's process for applying operating experience to their plant.

b. Findings

No findings were identified.

2RS2 Occupational ALARA Planning and Controls (71124.02)

These inspection activities supplement those documented in Inspection Report 05000373/2010002; 000374/2010002; and 05000373/2010004; 000374/2010004 and constitute one complete sample as defined in IP 71124.02-05.

.1 Inspection Planning (02.01)

a. Inspection Scope

The inspectors reviewed pertinent information regarding plant collective exposure history, current exposure trends, and ongoing or planned activities in order to assess current performance and exposure challenges. The inspectors reviewed the plant's three-year rolling average collective exposure.

The inspectors reviewed the site-specific trends in collective exposures (using NUREG-0713, "Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities," and plant historical data) and source term (average contact dose rate with reactor coolant piping) measurements (using Electric Power Research Institute TR-108737, "BWR Iron Control Monitoring Interim Report," issued December 1998, and/or plant historical data, when available).

The inspectors reviewed site-specific procedures associated with maintaining occupational exposures ALARA, which included a review of processes used to estimate and track exposures from specific work activities.

b. Findings

No findings were identified.

.2 Radiological Work Planning (02.02)

a. Inspection Scope

The inspectors selected the following work activities of the highest exposure significance.

- L2R13 DW SRV Activities ALARA Plan, Revision 2;
- L2R13 Under-Vessel Nuclear Instrumentation ALARA Plan, Revision 0;
- L2R13 CRD Pull/put ALARA Plan, Revision 0;
- L2R13 Reactor Vessel Disassembly/Reassembly ALARA Plan; and
- L2R13 Feed Water Heater Replacements ALARA Plan, Revision 0.

The inspectors reviewed the ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements. The inspectors determined whether the licensee reasonably grouped the radiological work into work activities, based on historical precedence, industry norms, and/or special circumstances.

The inspectors assessed whether the licensee's planning identified appropriate dose mitigation features; considered alternate mitigation features; and defined reasonable dose goals. The inspectors evaluated whether the licensee's ALARA assessment had taken into account decreased worker efficiency from use of respiratory protective devices and/or heat stress mitigation equipment (e.g., ice vests). The inspectors determined whether the licensee's work planning considered the use of remote technologies (e.g., teledosimetry, remote visual monitoring, and robotics) as a means to reduce dose and the use of dose reduction insights from industry operating experience and plant specific lessons learned. The inspectors assessed the integration of ALARA requirements into work procedure and RWP documents.

The inspectors compared the results achieved (dose rate reductions, person-rem used) with the intended dose established in the licensee's ALARA planning for these work activities. The inspectors compared the person-hour estimates provided by maintenance planning and other groups to the RP group with the actual work activity time requirements, and evaluated the accuracy of these time estimates. The inspectors assessed the reasons (e.g., failure to adequately plan the activity, failure to provide sufficient work controls) for any inconsistencies between intended and actual work activity doses.

b. Findings

No findings were identified.

.3 Verification of Dose Estimates and Exposure Tracking Systems (02.03)

a. Inspection Scope

The inspectors reviewed the assumptions and basis (including dose rate and person-hour estimates) for the current annual collective exposure estimate for reasonable accuracy for select ALARA work packages. The inspectors reviewed applicable procedures to determine the methodology for estimating exposures from specific work activities and the intended dose outcome.

The inspectors evaluated whether the licensee had established measures to track, trend, and, if necessary, to reduce occupational doses for ongoing work activities. The inspectors assessed whether trigger points or criteria were established to prompt additional reviews and/or additional ALARA planning and controls.

The inspectors evaluated the licensee's method of adjusting exposure estimates, or re-planning work, when unexpected changes in scope or emergent work were encountered. The inspectors assessed whether adjustments to exposure estimates (intended dose) were based on sound RP and ALARA principles or if they were just adjusted to account for failures to control the work. The inspectors evaluated whether the frequency of these adjustments called into question the adequacy of the original ALARA planning process.

b. Findings

No findings were identified.

.4 Radiation Worker Performance (02.05)

a. Inspection Scope

The inspectors observed radiation worker and RP technician performance during work activities being performed in radiation areas, airborne radioactivity areas, or high radiation areas. The inspectors evaluated whether workers demonstrated the ALARA philosophy in practice (e.g., workers are familiar with the work activity scope and tools to be used, workers used ALARA low dose waiting areas) and whether there were any procedure compliance issues (e.g., workers are not complying with work activity controls). The inspectors observed radiation worker performance to assess whether the training and skill level was sufficient with respect to the radiological hazards and the work involved.

b. Findings

No findings were identified.

.5 Problem Identification and Resolution (02.06)

a. Inspection Scope

The inspectors evaluated whether problems associated with ALARA planning and controls are being identified by the licensee at an appropriate threshold and were properly addressed for resolution in the licensee's CAP.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

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

4OA1 Performance Indicator Verification (71151)

.1 Unplanned Scrams per 7000 Critical Hours

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Scrams per 7000 Critical Hours PI for Unit 1 and Unit 2 for the period from the first quarter 2010 through the fourth quarter 2010. 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 6, dated October 2009, were used. The inspectors reviewed the licensee's operator narrative logs, ARs, event reports, and NRC Inspection Reports for the period of the first quarter 2010 through the fourth quarter 2010 to validate the accuracy of the submittals. The inspectors also reviewed the licensee's CAP 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 unplanned scrams per 7000 critical hours samples as defined in IP 71151-05.

b. Findings

No findings were identified.

.2 Unplanned Scrams with Complications

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Scrams with Complications PI for Unit 1 and Unit 2 for the period from the first quarter 2010 through the fourth quarter 2010. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the NEI Document 99-02, “Regulatory Assessment Performance Indicator Guideline,” Revision 6, dated October 2009, were used. The inspectors reviewed the licensee’s operator narrative logs, ARs, event reports, and NRC Integrated Inspection Reports for the period of the first quarter 2010 through the fourth quarter 2010 to validate the accuracy of the submittals. The inspectors also reviewed the licensee’s CAP 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 unplanned scrams with complications samples as defined in IP 71151-05.

b. Findings

No findings were identified.

.3 Unplanned Transients per 7000 Critical Hours

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Transients per 7000 Critical Hours PI for Unit 1 and Unit 2 for the period from the first quarter 2010 through the fourth quarter 2010. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the NEI Document 99-02, “Regulatory Assessment Performance Indicator Guideline,” Revision 6, dated October 2009, were used. The inspectors reviewed the licensee’s operator narrative logs, ARs, maintenance rule records, event reports, and NRC Integrated Inspection Reports for the period of the first quarter 2010 through the fourth quarter 2010 to validate the accuracy of the submittals. The inspectors also reviewed the licensee’s CAP 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 unplanned transients per 7000 critical hours samples as defined in IP 71151-05.

b. Findings

No findings 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. Inspection Scope

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

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

b. Findings

No findings were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

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

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

b. Findings

No findings were identified.

.3 <u>Selected Issue Followup Inspection: Unit 1 Shutdown Cooling Isolation and Reportability</u>

a. <u>Inspection Scope</u>

During a review of items entered in the licensee's CAP, the inspectors recognized a CAP item documenting the unexpected closure of the shutdown cooling common suction valve during the performance of procedure LOP-RH-07, "Shutdown Cooling System Startup, Operation and Transfer." The inspectors verified that the plant was returned to a stable, safe, shutdown condition and that the normal method of decay heat removal through the RHR system was restored.

The inspectors' review focused on the implementation of NRC reportability regulations and guidelines and in the appropriateness of the corrective actions implemented as a result of a previous Unit 1 loss of shutdown cooling event in July 2009. In addition to interviewing licensee staff, the inspectors reviewed various documents such as control room logs, ARs and operating procedures. Additional documents reviewed are listed in the Attachment to this report.

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

b. <u>Findings</u>

<u>(URI) Potential Failure to Make a Non-Emergency Event Notification to the NRC Following a Loss of Shutdown Cooling Safety Function on Unit 1</u>

<u>Introduction</u>: The inspectors identified an Unresolved Item (URI) associated with the potential failure to make a non-emergency eight-hour notification to the NRC in accordance with 10 CFR 50.72 (b)(3)(v) for a loss of safety function of a system which was required to remove residual heat from the reactor. This item remains unresolved pending further review by the NRC staff.

<u>Description</u>: On February 2, 2011, Unit 1 was in hot shutdown (Mode 3) following an unexpected scram that occurred the previous day. The control room operators were in the process of placing the "B" train of shutdown cooling in operation. When the "B" RHR pump was started, the initial flow conditions created a spurious closure of the common pump suction valve, CIV 1E12-F009, on a sensed high-flow condition. This containment isolation valve 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 the1E12-F009 valve to stop the potential interfacing system loss-of-coolant accident. The closure of this common suction valve resulted in the "B" RHR pump tripping and a complete isolation of shutdown cooling. The licensee declared both trains of shutdown cooling inoperable and entered Technical Specification (TS) 3.4.9, "RHR Shutdown Cooling – Hot Shutdown", which requires, in part, for the immediate initiation of actions to restore the RHR shutdown cooling subsystem to operable status.

In order to prevent spurious containment isolations, the licensee had proceduralized the installation of jumpers that bypass the relays that would cause the containment isolation to occur, based upon previous operating experience. Following the closure of valve 1E12-F009 and the isolation of shutdown cooling, equipment operators installed jumpers to bypass the containment isolation. The control room operators reset the containment isolation logic, re-opened the 1E12-F009 valve, started the “B” RHR pump, and placed that train in operation. Once the RHR system was operating in shutdown cooling mode, the jumpers to defeat the high flow isolation were removed, shutdown cooling was declared operable, and the associated TSs were exited.

The licensee reviewed the event for 10CFR 50.72 reportability, but determined that it was not reportable since the closure of 1E12-F009 was considered spurious and it occurred during system startup, hence, shutdown cooling was never established. As such, this event was not characterized as an event or condition that could have prevented the fulfillment of a safety function. The inspectors consulted NUREG-1022, Revision 2, “Event Report Guidelines 10 CFR 50.72 and 50.73,” which is considered the NRC staff’s position on the reporting of nuclear events. The NUREG-1022 guidance states, 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 scenario. In addition, NUREG-1022 states that the event must be reported regardless of whether or not an alternate system could have been used to perform the system’s safety function. In addition to the reportability aspect, the inspector is reviewing the appropriateness of the corrective actions completed by the licensee as a result of this event.

The inspectors plan to engage personnel in the NRC Office of Nuclear Reactor Regulation to ensure that the licensee is implementing the reportability guidelines and regulations appropriately and to ensure that the loss of shutdown cooling by all licensees is treated consistently. An URI is opened pending further review by the NRC staff. (URI 05000374/2011002-04).

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

.1 Unit 1 Automatic Reactor Scram

a. Inspection Scope

The inspectors reviewed the plant’s response to a Unit 1 reactor scram on February 1, 2011. With the reactor operating at 100 percent rated thermal power and the mode switch in Run, a fault in one of the phases of the Unit 1 West Main Power Transformer (MPT) resulted in a voltage perturbation on the grid, which caused a main generator lockout and subsequent reactor scram. The 1W MPT fire deluge system immediately initiated as a result of the transient, even though there was no fire. Additional complications due to the voltage perturbation included a reactor water cleanup system isolation on both units, a reactor building ventilation partial isolation on Unit 1, and one circulating water pump trip. Unit 1 experienced an expected reactor pressure spike when its main turbine stop and control valves shut on the generator/turbine trip resulting in the ‘U’ SRV lifting and reseating once reactor pressure was lowered. The C-phase high voltage bushing on the 1W MPT that caused the generator trip was replaced and the unit attempted to restart on February 9, 2011. During startup operations, the licensee had to shut down once more after identifying a steam leak originating from a valve associated with the Reactor Core Isolation Cooling (RCIC) system. The licensee repaired the leak on February 10 and completed a restart of the unit, returning to full power on February 13, 2011.

The inspectors reviewed the operators' response to the scram and associated equipment issues. The inspectors noted the use of emergency and plant operating procedures and evaluated the performance of the safety systems during the scram. The inspectors also observed the licensee's review of the emergency action level matrix for applicability and ensured that NRC notifications were made in a timely manner. Documents reviewed in this inspection are listed in the Attachment to this report.

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

b. Findings

No findings were identified.

4OA5 Other Activities

.1 Operation of an Independent Spent Fuel Storage Installation at Operating Plants (60855.1)

a. Inspection Scope

On January 26, 2011, during LaSalle County Station's independent spent fuel storage installation (ISFSI) loading campaign, dry storage cask (HI-STORM) number four became immobilized while being transported through the reactor building equipment access trackway. The HI-STORM was loaded with spent fuel, supported underneath by a low profile transporter (LPT), and was in transit to the ISFSI haul path being pulled by a rail car mover. There are six roller assemblies underneath the LPT that provide support and allow the HI-STORM to traverse the railway tracks in the reactor building. Upon closer inspection, the LPT did not appear centered on the rails and one of the chain links on a roller assembly appeared to have failed. The rail car mover was unable to overcome the additional force of friction created by the broken roller. The licensee initiated a recovery plan. The site's initial attempts to free the stuck HI-STORM were monitored by the resident inspectors with support from NRC Region III ISFSI inspectors.

To free the HI-STORM, the licensee first attempted to align the LPT to the center of the rails and applied grease to the rails to reduce friction in an attempt to allow the LPT to slide on the rails. A jacking mechanism was created, supported against the access building wall, and applied a horizontal force to the front of the LPT in an effort to realign the LPT. The effect of the additional horizontal force was evaluated in engineering change (EC) 383020, "Evaluate Side-Jacking of Low Profile Transporter (LPT) with Loaded HI-STORM," and reviewed by the inspectors. The EC limited the horizontal load to 6811 pounds force. The site's first attempt to free the HI-STORM occurred, using the methods described, on January 27, 2011, and then again on January 28, 2011. These attempts were not successful in freeing the HI-STORM. The inspectors were onsite to verify these operations were carried out safely and in accordance with the approved procedures.

Next, the licensee decided an alternate large prime mover would be used to move the LPT and HI-STORM, instead of the original rail car mover. This was due to the large prime mover's ability to provide better traction and force to overcome friction caused by the binding roller. The licensee completed EC 383045, "Evaluate Pulling the LPT with a Large Prime Mover with a Loaded HI-STORM and Rollers Not Fully Functional." The evaluation determined the limitations associated with using a large prime mover to transport the LPT and HI-STORM. Specifically, the towing connections and LPT were analyzed to determine the maximum force that could be applied. The licensee determined that the maximum force applied by the large prime mover shall not exceed approximately 22,622 pounds force, and that limiting the force to this value would ensure that the LPT would not be overstressed.

Engineering Change (EC) 383045 also discussed that the broken rollers would not affect the roller capacity during a seismic event, because even though the rolling function of the individual roller was lost, the structural capacity was still available to resist the seismic loads and therefore calculation L-003495, “Stability Evaluations for the HI-STORM and HI-TRAC Casks in the Reactor Building,” Revision 1 bounded the degraded condition. The inspectors reviewed EC 383045 prior to the move.

The licensee attempted to move the LPT and HI-STORM using the large prime mover on January 29, 2011. The LPT failed to move and the licensee pursued a revision to EC 383045 to increase the allowable towing force. Upon NRC review of the revision, the inspectors noted an incorrect value was used in a bolt capacity evaluation. This issue was documented in AR 1169293, “EC 383045 Load Change Required During the Review Process,” and resulted in a reduction to the allowable towing force. The final allowable towing force was determined to be 27,692 pounds force and the movement of the HI-STORM was attempted again on January 30, 2011. However, when the allowable force was applied, the LPT and HI-STORM failed to move. The NRC inspectors were onsite to observe that the operation was carried out safely and in accordance with approved procedures.

In response to the circumstances of January 26, 2011, the NRC increased oversight of the attempts to free the HI-STORM at the LaSalle County Station and began a reactive inspection on February 1, 2011, to monitor further licensee actions. The reactive inspection plan instructed the inspectors to assess the licensee’s actions in safely moving the loaded HI-STORM out of the rail car area and into the haul path. The focus was on safety and that the actions taken were within the design and licensing basis of the affected equipment and facilities. This reactive inspection was performed in accordance with IMC 2690, “Inspection Program for Dry Storage of Spent Reactor Fuel at Independent Spent Fuel Storage Installations and for Title 10 of the Code of Federal Regulation Part 71 Transportation Packagings,” and IMC 2515, “Light-Water Reactor Inspection Program – Operations Phase.” Specifically the reactive inspection instructions included, but were not limited to, the following items:

- assessment of the licensee’s proposed actions for safely moving the loaded cask out of the access building;
- assessment of margins to safety of licensee ECs required for operations; and
- observations confirming that actions taken by the licensee ensured safe movement of the cask.

The licensee’s recovery plan was revised and the licensee determined that the damaged LPT would require repair. This required lifting the loaded HI-STORM off the damaged LPT using a set of hydraulic lifting jacks. The licensee determined that the HI-STORM must be supported seismically to prevent any potential adverse impact to the adjacent safety-related structure, and prevent damage to the fuel, while the HI-STORM was not supported by the LPT during a hypothetical seismic event. Consequently, a HI-STORM restraint was installed and attached to the reactor building structure inside the equipment access trackway. The seismic restraint impact on the reactor building was evaluated, in EC 383085, “Dry Cask Storage Reactor Building Equipment Access Trackway Restraint on Elevation 710’-6,” and was reviewed by the inspectors.

The loaded HI-STORM weighed approximately 180 tons. To facilitate lifting the HI-STORM, two jacking assemblies were constructed and placed on either side of the LPT. The jacking assemblies were constructed of parallel plates approximately 8.5 feet by 2 feet by 1 inch and made contact with the bottom of the HI-STORM where the HI-STORM overhangs the LPT by approximately 2 feet on each side. Between the parallel plates, lateral guides were installed to withstand any lateral loading. The bottom plate rested on the floor of the equipment access trackway, and the top plate made contact with the bottom of the HI-STORM. To raise the top plate, eight hydraulic cylinders (four per side) with capacity of 100 tons each, were placed between the plates. Two hydraulic pumps supplied the eight cylinders. Each pump provided hydraulic fluid to two cylinders in each jacking assembly (four cylinders total). This arrangement was chosen such that should one pump fail, the remaining four jacks would be capable of supporting the weight of the HI-STORM. This configuration was evaluated, in EC 383060 Revision 02, “Evaluate Lifting Dry Cask Off of LPT,” and reviewed by the NRC inspectors for adequate margins to safety prior to the lift.

The effects of the jacking assembly on the HI-STORM were evaluated as was the ability of the HI-STORM to resist the lateral loads imparted by the restraints in calculation L-003612 Revision 0, “Evaluation of Loads Being Imposed upon a Loaded HI-STORM during Lift & Set Activities to Correct Problems with a Stuck LPT.” Similarly, the licensee evaluated the effects a seismic event would have on the loaded jacking assemblies and the seismic restraints in calculation L-003613 Revision 0, “Structural Evaluations Associated with the Cask Temporary Jacking Configuration in the Reactor Building Equipment Access Airlock.” Both calculation L-003612 and L-003613 were reviewed by NRC inspectors.

On February 8, 2011, the licensee conducted the first lift of the HI-STORM off the damaged LPT. The WO 01405514-04, limited the lift height of the HI-STORM to one inch off of the LPT and to maintain the HI-STORM within one degree off of vertical. The NRC inspectors were onsite to observe the lift. Once the HI-STORM was off the LPT by approximately 1/2 inch, the damaged LPT was pulled out of the Equipment Access Airlock by the Trackmobile for repair. A replacement LPT was moved into position under the HI-STORM and the HI-STORM was set back down on the replacement LPT while the damaged LPT was repaired. The replacement LPT was not used in an attempt to remove the HI-STORM, as the LaSalle LPT had been modified to the specific needs of the site.

Once the damaged LPT was accessible for full inspection, it was confirmed that one of the rollers had failed. Chain links on each side of a roller had parted, which is believed to be the cause of the stuck LPT. In addition, bolts that held the roller to the LPT appeared to have noticeable deformation indicating the possibility that they were beginning to shear. It is unclear if this was the cause of the stuck HI-STORM. The exact failure mechanism is unknown and will be evaluated by a root cause analysis as documented in AR 1166922. All six of the effected rollers were replaced and the LPT was brought back to the equipment access trackway.

The second lift was conducted on February 9, 2011. The temporary LPT was removed from under the HI-STORM and the LaSalle LPT was positioned underneath the loaded HI-STORM. Once the HI-STORM was set down, the licensee removed the jacking assemblies and dismantled the seismic restraints. On the afternoon of February 9, 2011, the rail car mover was again connected to the repaired LPT in another attempt to pull the HI-STORM out of the building. The NRC inspectors were onsite to observe the second lift as well as the attempt to remove the HI-STORM from the building. The repairs to the LPT were successful and the HI-STORM was safely removed from the building. In the evening of February 9, 2011, HI-STORM number four was placed on the ISFSI pad.

b. Findings

No findings were identified.

.2 (Closed) Temporary Instruction 2515/179, "Verification of Licensee Responses to NRC Requirement for Inventories of Materials Tracked in the National Source Tracking System Pursuant to Title 10, Code of Federal Regulations, Part 20.2207 (10 CFR 20.2207)"

a. Inspection Scope

The inspectors confirmed that the licensee has reported the initial inventories of sealed sources pursuant to 10 CFR 20.2207 and verified that the National Source Tracking System database correctly reflects the Category 1 and 2 sealed sources in custody of the licensee. Inspectors interviewed personnel and performed the following:

- reviewed the licensee's source inventory;
- verified the presence of any Category 1 or 2 sources;
- reviewed procedures for and evaluated the effectiveness of storage and handling of sources;
- reviewed documents involving transactions of sources; and
- reviewed adequacy of licensee maintenance, posting, and labeling of nationally tracked sources.

b. Findings

No findings were identified.

4OA6 Management Meetings

.1 Exit Meeting Summary

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

.2 Interim Exit Meetings

Interim exits were conducted for:

- the results of the operation of an ISFSI at operating plants inspection with Mr. H. Vinyard, Engineering Director and members of his staff on February 11, 2011; and
- the results of the Radiological Hazard Assessment and Exposure Controls and ALARA Controls inspection with the Site Vice President, Mr. D. Rhodes, on March 2, 2011.

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.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

D. Rhoades, Site Vice President
P. Karaba, Plant Manager
J. Houston, Regulatory Assurance
J. Kutches, Manager of Projects
K. Hedgspeth, RP Manager
B. Maze, ISFSI Project Manager
B. Rash, Maintenance Director
S. Shields, Regulatory Assurance
T. Simpkin, Regulatory Assurance Manager
K. Taber, Operations Director
J. Vergara, Regulatory Assurance
H. Vinyard, Work Management Director
J. Washko, Outage Manager
J. White, Site Training Director
K. Lyons, Chemistry Manager
C. Wilson, Station Security Manager

Nuclear Regulatory Commission

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

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000373/2011002-01 05000374/2011002-01	NCV	Failure to Post Protected Pathway Signs for a Red Risk Path System
05000373/2011002-02 05000374/2011002-02	NCV	Failure to Follow Plant Barrier Control Process Caused Secondary Containment to Become Inoperable
05000373/2011002-03 05000374/2011002-03	NCV	Failure to Make a Required 10 CFR 50.72 Report for an Inoperable Secondary Containment
05000374/2011002-04	URI	Potential Failure to Make a Non-Emergency Event Notification to the NRC Following a Loss of Shutdown Cooling Safety Function on Unit 1

Closed

05000373/2011002-01 05000374/2011002-01	NCV	Failure to Post Protected Pathway Signs for a Red Risk Path System
05000373/2011002-02 05000374/2011002-02	NCV	Failure to Follow Plant Barrier Control Process Caused Secondary Containment to Become Inoperable
05000373/2011002-03 05000374/2011002-03	NCV	Failure to Make a Required 10 CFR 50.72 Report for an Inoperable Secondary Containment

Discussed

None

LIST OF DOCUMENTS REVIEWED

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

1R01 Adverse Weather Protection

Procedures:
- LAP-100-44; Inclement Weather Guidance; Rev. 1
- LOA-TORN-001; High Winds / Tornado; Rev. 11

Action Requests:
- 1167989; RP ERO Staffing Question; 1/28/11

Miscellaneous:
- LSCS P.G. # 128; Snow Policy; 12/14/2007

1R04 Equipment Alignment

Action Requests:
- 1161129; Actuator Housing is Leaking Grease; 1/11/2011
- 1161140; NRC Identification of Potential Issues; 1/11/2011
- 1161168; Air Operated Valve 2E51-F025 Has Packing Leak; 1/11/2011

Drawings:
- M-2116; P & ID / C & I Main Steam System “MS”; Rev. J

Miscellaneous:
- LOP-HP-02E; Unit 2 High Pressure Core Spray Electrical Checklist; Rev 5
- LOP-HP-02M; Unit 2 High Pressure Core Spray Mechanical Checklist; Rev 17
- LOP-IN-02M; Unit 2 Drywell Pneumatic System Mechanical Checklist; Rev 19
- LOP-RI-02E; Unit 2 Reactor Core Isolation Cooling System Electrical Checklist; Rev. 14
- LOP-RI-02M; Unit 2 Reactor Core Isolation Cooling System Mechanical Checklist; Rev. 19
- LSCS-UFSAR 9.3; Drywell Pneumatic System; Rev. 14

1R05 Fire Protection

Action Requests:
- 1174654; Reactor Level 4 Alarm when Securing 2A TDRFP; 2/13/2011
- 1177086; NOS ID: Four Foot Puddle of Oil on Floor; 2/18/2011
- 1178242; 2FW01PA Oil Leak Found Leaking Through Floor to 731’ Again; 2/17/2011
- 1180317; STD Team – Housekeeping; 2/25/2011
- 1180598; B TDRFP Exh Duct Exp Joint Hole Found in the Rib #1 Bellows; 2/26/2011
- 1180980; Oil Running Down Near TDRFP Duct and Bellows; 2/1/2011
- 1181361; STD Team – Housekeeping; 2/28/2011
- 1181916; STD Team – Housekeeping; 3/1/2011
- 1184580; Oil Leak on 2B TDRFP Turbine Outboard Bearing; 3/8/2011
- 1184831; U2 MDRFP Small Outboard Seal Leak; 3/8/2011
- 1185329; NRC Id’d: Housekeeping Issues in TDRFP Rooms; 3/9/2011

Miscellaneous:
- LaSalle County Generating Station Pre-Fire Plan; Fire Zone 2D, Unit 1, Elevation 786' 6"
- LaSalle County Generating Station Pre-Fire Plan; Fire Zone 3D, Unit 2, Elevation 786' 0"
- LaSalle County Generating Station Pre-Fire Plan; Fire Zone 5A, Unit 2, Elevation 768' 0"
- LaSalle County Generating Station Pre-Fire Plan; Fire Zone 5B, Unit 2, Elevation 731' 0"
- LaSalle County Generating Station Pre-Fire Plan; Fire Zone 5B2, Unit 2, Elevation 663' 0" to 768' 0"
- LSCS-FPR; H.3-2.5; Unit 1 – Elevation 786' 6", Fire Zone 2D
- LSCS-FPR; H.3-3.5; Unit 2 – Elevation 786' 6", Fire Zone 3D
- LSCS-FPR; H.3.5.2; Unit 2 Turbine Driven Reactor Feed Pump Area – Fire Zone 5A2; Rev. 4
- LSCS-FPR; H.3.5.14; Unit 2 Motor-Driven Reactor Feed Pump Room – Fire Zone 5B10; Rev. 4
- LSCS-FPR; H.4.2.7-8; Safe Shutdown for Fire Zones 2D and 3D, Units 1 and 2 Reactor Buildings

1R07 Annual Heat Sink Performance

Procedures:
- EC 364950; ASME Section XI Code Repair Plan for RHR Heat Exhanger 1(2)E12-B001A/B End Cover Plate; Rev. 001
- EC 365168; Evaluation of Impact of Unit 2B RHR Heat Exchanger bypass Flow on Thermal Performance; Rev. 0
- EC 383287; Evaluation of the 2B RHR Heat Exchanger Eddy Current Testing; Rev. 0

Action Requests:
- 1176833; 2E12-B001B 2B RHR HX Channel Cover Needs Repair; 2/18/2011

Miscellaneous:
- HX 2E12-B001B; 2B RHR Heat Exchanger Inspection Report; 2/18/2011

1R11 Licensed Operator Requalification Program

Miscellaneous:
- Simulator training scenario 1st quarter 2011

1R12 Maintenance Effectiveness

Procedures:
- ER-AA-310; Implementation of the Maintenance Rule; Rev. 8
- ER-AA-310-1004; Maintenance Rule – Performance Monitoring; Rev. 8
- ER-AA-310-1005; Maintenance Rule – Dispositioning Between (a)(1) and (a)(2)
- NF-AA-430; Failed Fuel Action Plan; Rev. 11

Action Requests:
- 1011466; Div 2 Visual Annunciator Power Supply Failure Alarm; 1/2/2010
- 1013228; Annunciator Point Power Supplies; 1/6/2010
- 1039681; Div 1 Ground U-1; 3/7/2010
- 1041120; NSO Sequence of Events Computer Not Working; 3/10/2010
- 1055680; Annunciator Did Not Alarm During OPRM Surveillance; 4/12/2010
- 1059499; Spurious Alarms with No R-Point on SER; 4/21/2010
- 1077616; Annunciator 0PL01J-E203 Will Not Light Up; 6/6/2010
- 1083433; 2H13-P603 Annunciator Horn Alarming Intermittently; 6/23/2010

- 1084991; U1 MCR Visual Annunciator Issue; 6/28/2010
- 1108220; Three Annunciator Alarms Not Working Correctly; 8/31/2010
- 1149959; WCI – Annunciator Power Supplies Not Replaced; 12/8/2010
- 1155164; Completion of the Annunciator System Maintenance Rule Action Plan Delayed; 12/272010
- 1155164; Annunciator Maintenance Rule Action Plan Delayed; 12/22/2010
- 1155759; +28 VDC Failure on Annunciator Power Supply #4; 12/25/2010
- 1155759; 1PM02J-B103, Div. 1 Visual Annunciator Trouble Alarm; 12/28/2010
- 1158119; Loss of Div 1 Visual Annunciator Power; 1/3/2011
- 1158619; Annunciator Tile 1PM13J-A503 Lit w/No Alarm Input Actuated; 1/4/2011
- 1158749; Annunciator Point Power Supply 1 and 2; 1/5/2011
- 1174725; Increase in XE-133 Values During U-2 Reactor S/D; 2/13/2011
- 1176714; RM-L2C13 Suspect Fuel Defect 29-16 Location; 2/17/2011
- 1180642; RM – Fuel – MR Performance Criteria Exceeded on Unit 2; 2/17/2011
- 1181509; Spurious Actuation of Division I Annunciators; 3/1/2011
- 1184005; 2A RT Pump Tripped on Hi Motor Cavity Temperature; 3/11/2011

Work Orders:
- 1301023-01; Replace Point Power Supply #4 (PS-12); 1/3/2011

Miscellaneous:
- AN-01; LaSalle Maintenance Rule Evaluation: Provide Control Room Operators with Audible / Visual Notification of Abnormal Conditions; 12/2010
- AN-01; LaSalle Maintenance Rule Scoping/Performance Criteria, Annunciator; undated
- AR 1181388; Maintenance Rule Expert Panel Determination for Maintenance Rule AP-09; 2/28/2011
- AR 930416; LaSalle Maintenance Rule (a)(1) Action Plan for Annunciator System; 7/16/2010
- AR 1176714; Functional Failure cause Determination Evaluation of Unit 2 Nuclear Fuel Higher than Anticipated Xe-133 Concentration; 3/11/2011
- LaSalle Maintenance Rule Evaluation: AN System; 11/2010
- System Health Report; Unit 1 AN – Annunciators; 7/1/2010 – 9/30/2010
- System Health Report; Unit 2 AN – Annunciators; 7/1/2010 – 9/30/2010

<u>1R13 Maintenance Risk Assessments and Emergent Work Control</u>

Procedures:
- OU-LA-104; Shutdown Safety Management Program; Rev. 13
- OP-AA-108-117; Protected Equipment Program; Rev. 1
- OP-LA-101-111-1002; LaSalle Operations Philosophy Handbook; Rev. 33
- WC-AA-101; Online Work Control Process; Rev. 18

Action Requests:
- 1161438; NRC Identified Concern on Protective Pathway Determination; 1/11/2011
- 1166596; ACE for Protected Pathways; 1/25/2011

Miscellaneous:
- LaSalle Operations Logs; 1/11/2011
- PRA-Meeting Agenda; 1/5/2011
- Risk Significance of Failure to Establish Thorough Protected Equipment during RCIC Work Window; 2/17/2011

1R15 Operability Evaluations

Procedures:
- CC-AA-201; Plant Barrier Control Program; Rev. 8
- CC-LA-201-1001; Plant Barrier Control Program Implementation; Rev. 2
- EC 332271; Evaluation to Allow Division III Switchgear Ventilation to be Out of Service (OOS) While the Unit is Online; Rev. 0
- EC 372452; GL2008-01 Void Calculation and Acceptance Criteria; Rev. 00
- EC 378126; RCMS Power Module Trip on Unintended Control Rod Motion; 9/22/2010
- ECR 363533; Provide Guidance for Acceptance Criteria for Acceptable Pressure Decay during Underground RCIC Piping Test (LTS-900-14); undated
- LOA-VX-101; Unit 1 Switchgear Heat Removal System Abnormal; Rev. 7
- LOP-RH-01; Filling and Venting the Residual Heat Removal System; Rev. 45
- LOR-2H13-P601-C308; LPCS Pump Discharge Press Low; Rev. 3
- LOS-CS-M1; Secondary Containment Integrity; Rev. 24
- LOA-PC-101; Primary/Secondary Containment Trouble; Rev. 14
- LOS-LP-Q1; LPCS System Inservice Test; Rev. 52
- LOS-MS-R7; Main Steam Safety Relief Valve Operability; Rev. 1
- LTS 900-14; Underground RCIC Piping Test; Rev. 7
- OP-AA-108-106; Equipment Return to Service; Rev. 4
- OP-AA-108-115; Operability Determinations (CM-1); Rev. 9
- OP-AA-108-115-1002; Supplemental Consideration for On-Shift Immediate Operability Determinations; Rev. 2

Action Requests:
- 1164614; Loss of Div. 2 SWGR Ventilation Due to Blowing Fuses; 1/20/2011
- 1171555; E SRV Non- ADS Accumulator Check Valve Needs Repair; 2/6/2011
- 1171578; E SRV ADS Accumulator Check Valve Needs Repair; 2/6/2011
- 1171686; Inspection Results of the Check Valve 1B21-F040E; 2/6/2011
- 1171688; Inspection Results of the Check Valve 1B21-F036E; 2/6/2011
- 1171718; E SRV Non-ADS Accumulator Check Valve Needs Repair; 2/6/2011
- 1171788; 1B 21-A004E Non- ADS Accumulator will Pass LOS-MS-07; 2/7/2011
- 1177566; RCIC Piping Exceeded Pressure Drop Acceptance Criteria; 2/20/2011
- 1179224; Small Void upstream of 2E12-F016B Following LOP-RH-01; 2/24/2011
- 1181963; NRC ID: Piping Void IR 1179224 Incomplete Description; 3/1/2011
- 1182255; Loss of Secondary Containment During Fuel Movement; 3/2/2011
- 1183511; OTDM Input for Possible Loss Rod or ATWS from Degraded Pips; 3/11/2011
- 1195987; Secondary Containment Pressure Anomaly; 4/1/2011

Work Orders:
- 1363966-29; 1E SRV Tailpipe Temperature Step Change; 2/9/2011
- 1372971-01; LOS-LP-Q1 U2 LPCS System Att. 2A; 12/24/2010

Drawings:
- 1E-1-4083AL; Schematic Diagram Reactor Building Ventilation System “VR”
- 1E-1-4500CQ; External Wiring Diagram Reactor Building Ventilation System “VR” Local Control Panels; Rev. C
- 1E-1-4083BA; Schematic Diagram Reactor Building Ventilation System “VR” Part 25; Rev. E
- 1E-1-4500CT; External Wiring Diagram Reactor Building Ventilation System “VR” Local Control Panels; Rev. E
- 1E-1-4083AN; Schematic Diagram Reactor Building Ventilation System “VR” Part 13; Rev. D

- 1E-1-4083AP; Schematic Diagram Reactor Building Ventilation System "VR" Part 14; Rev. E
- 1E-0-452AA; Internal/External Wiring Diagram Railroad Equipment Access Airlock Door D-19 System "VV"; Rev. D
- 1E-0-4525AB; Internal/External Wiring Diagram Railroad Equipment Access Airlock Door D-20 System "VV"; Rev. D
- 1E-1-4499AE; External Wiring Diagram Steam Tunnel Airlock Door Cont. PNLS Sys. "VV" (Sargent & Lundy); Rev E
- 1E-1-4083AK; Schematic Diagram Reactor Building Ventilation System "VR" Part 10; Rev. E
- 1E-1-4500CP; External Wiring Diagram Reactor Building ventilation System "VR" Local Control Panels; Rev. H
- 1E-1-4499AF; Internal Wiring Diagram Steam Tunnel Airlock Door Cont. PNLS Sys. "VV" (Sargent & Lundy); Rev. A
- 1E2-4499AE; External Wiring Diagram Steam Tunnel Airlock door cont. PNLS Sys. "VV" (Sargent & Lundy); Rev. C
- 1E2-4499AF; Internal Wiring Diagram Steam Tunnel Airlock Door Cont. PNLS Sys. "VV" (Sargent & Lundy); Rev. A
- Fig. 62-1; Pressure Relief System; 4/2001
- Fig. 62-2; Automatic Depressurization Logic; 4/2001
- Fig. 070-01; Main Steam System Overview; 4/2009
- Fig. 070-04; Safety Relief Valve Operator; 9/2005
- Fig. 070-05; Low-Low Set Logic 'A'; 3/2001
- Fig. 070-06; Low-Low Set Logic 'B'; 3/2001
- Fig. 070-07; SRV Discharge Locations and Relief/Safety Setpoints (psig); 3/2001
- Fig. 070-030; Main Steam Safety/Relief Valve; 10/1999
- Fig. 128-4; Switchgear Heat Removal System; 2/2003
- M-55; P & ID Main Steam; Rev. Y
- M-66; P & ID Drywell Pneumatic System; Rev. AE
- M-939; Residual Heat Removal Piping; Rev. AE
- Unit 1/2 'B' Residual Heat Removal Elevation Drawing

Calculations:
- L-001649; Acceptance Criteria for Leakage testing of ADS SRVs and Accumulators and SR Portion of IN System (EC 364700); Rev. 4
- L-003263;Safety Relief Valve Functions, Low-Low-Set and Automatic Depressurization System; Rev. 2

Miscellaneous:
- EC 383151; Evaluation of Non-ADS Accumulator 1B21-A004E Leakage Rate; Rev. 0
- LaSalle Operations Log – LPCS; 12/20/2010 – 12/25/2010
- LaSalle Operations Log; 9/25/2010
- LaSalle Operations Log; 1/20/2011 – 1/21/2011
- LaSalle Operations Log; 3/2/2011 – 3/3/2011
- LER 05000220-05-00; Nine Mile Point Unit 1 Licensee Event report: Loss of Secondary Containment due to Both Reactor Building Track Bay Inner and Outer Doors being Opened Simultaneously; 11/30/2000
- LSCS-UFSAR 5.2.2.4; Safety/Relief Valves; Rev. 13
- LSCS-UFSAR 9.4-38; Switchgear Heat Removal System; Rev. 15
- Switchgear Room (SWGR) Temperature with Associated SWGR Ventilation System (VX) OOS; 2/3/2000

1R18 Plant Modifications

Procedures:
- EC 378126; 50.59 Screening L10-133 Add Two Programmable Controllers to Monitor CRD Drive Water Flow Loop; Rev. 0
- EC 378126; Design Considerations Summary for RCMS Power Module Trip on Unintended Control Rod Motion; Rev 000
- EC 378126; Work Planning Instructions for RCMS Power Module Trip on Unintended Control Rod Motion; Rev 000
- EC 380550/380789; MSO Mods for U-2 “The 5000 ft view”; 2/2011
- EC 380550; Design Consideration Summary for MSO – Modification to Rewire MOV’s Control Circuitry to Prevent the Valves from Spuriously Opening during MCR Fire; Rev. 4
- EC 380550; Work Planning Instructions for Modification to Rewire MOVs Control Circuitry to Prevent the Valves from Spuriously Opening during a Main Control Room Fire; Rev. 5
- EC 380552/380550; 50.59 Screening L10-195; Shorting Circuit Control Added to Reduce Possibility of Spurious Valve Opening due to MCR fire; Rev. 0
- EC 380789; Design Consideration Summary for MSO Mod FPR Synch Relays for ACBS 2412, 2415, 2422, and 2425; Rev. 1
- LST-2010-020; MSO Mod Test HR C Valves EC 380550; Special Test / Procedure Approval; 1/11/2011
- LST-2010-029; MSO Mod Test Synchrocheck Relay Div 1 EC 380789; 2/21/2011

Action Requests:
- 1177956; LaSalle review of Clinton HPCS Overfill Scenario; 2/17/2011
- 1178678; RCMS Problem Found During Replacement of MCR A Controller; 2/23/2011

Drawings:
- 1E-1-4206CF; Schematic Diagram RCMS, System “RD” Panel 1H13-P603 RCMS Power Distribution; Rev. A
- RCMS Power module Trip Circuit

1R19 Post-Maintenance Testing

Procedures:
- LOS-DG-202; 2A Diesel Generator, 2DG01K, Start and Load Acceptance Surveillance; Rev. 7
- LOS-HP-Q1; HPCS System Inservice Test; Rev. 63
- LOS-RI-Q2; Reactor Core Isolation Cooling System Valve Inservice Test for Refuel and Cold Shutdown Conditions; Rev. 38
- LST-2010-011; 2A Diesel Generator Voltage Regulator Test; Rev. 0

Action Requests:
- 1154709; U-2 Div 1 LPCS/LPCI Waterleg Pump Disch Press Lo; 12/21/2010
- 1158150; Existing Switch does not Match Schematic or Wiring Drawing; 1/3/2011
- 1164044; U-2 Div.1 RHR WTR Leg Pump Degraded; 1/19/2011
- 1173113; Leak Identified on 1E51-F076; 2/9/2011
- 1178718; LST 2010-001 Step Missed; 2/23/2011
- 1179040; Typo Found during LST-2011-001; 2/23/2011

Work Orders:
- 1363259-02; Contingency to Replace Min Flow Orifice Plate 2E21-D003; 1/20/2011
- 1365532-05; 1B RR Pump Seal Pressure Trends; 2/9/2011

- 1409247-01 Doc 1A; Perform Repairs on Valve 1E51-F076; 2/2011
- 1409247-01 / 1E51-F076; Perform Weld Repair to Valve; 2/10/2011

Drawings:
- CN04-4048; Anderson, Greenwood & Co., Valve Assy, Motor OPR (Less Motor) G12A15 Series; Sargent & Lundy approved on 5/2/1979
- M-101; P&ID Reactor Core Isolation Coolant (RCIC); Rev. BH

Miscellaneous:
- U2 Div. 1 H_20 Leg Pump Day Shift Operator Log; 1/17/2011
- U2 LPCS Maintenance Window; 1/19/2011 - 1/21/2011

<u>1R20 Outage Activities</u>

Procedures:
- CRD-007; GE Hitachi Nuclear Energy, CRDM Exchange Processes; Rev. 11
- LGP-1-1; Normal Unit Startup; Rev. 94
- LOP-DW-01; Drywell Close Out (After Outage); Rev. 48
- LOP-NB-02; Operations with the Potential to Drain the Reactor Vessel; Rev. 10
- LOP-RH-07; Shutdown Cooling System Startup, Operation and Transfer; Rev. 61
- OP-AA-108-108; Unit 1 Forced Outage Restart Review; 2/7/2011
- OP-AA-108-108-1001; Drywell/Containment Closeout; Rev. 1
- OP-AA-117-101; Operations with the Potential to Drain the Reactor Vessel; Rev. 0
- OP-LA-117-101-1001; Operations with the Potential to Drain the Reactor Vessel; Rev. 2

Action Requests:
- 1169879; U-2 Response from the U-1 Scram; 2/1/2011
- 1169881; Various U1 Control Rod Scram Valves Indicate Open; 2/1/2011
- 1169944; Packing Leak on 1HD189D; 2/2/2011
- 1169946; LaSalle Unit 1 Scram 2-1-11; 2/1/2011
- 1169948; After U-1 Scram the 1G CP Prefilter Leaked Water from Seam; 2/1/2011
- 1169952; Found Likely Cause of Generator Lock-Out Trip; 2/2/2011
- 1169952; U1 Scram due to MPT Bushing Failure; 2/4/2011
- 1169954; 1A CW Pump tripped During U-1 MPT Trip and Scram; 2/1/2011
- 1169959; 1A RR FCV Motion Inhibit When Trying to Open after Scram; 2/1/2011
- 1170016; RWCU Isolation Occurred with U1 MPT Trip; 2/1/2011
- 1170241; 1A Circ Water Pump Trip; 2/1/2011
- 1170243; U1 Rod 18-23 Data Faults Following 2-1-11 Auto Scram; 2/1/2011
- 1170371; Additional Damage Discovered on 1W MPT; 2/3/2011
- 1170567; Decreased RT Flow with Both RR Pumps Isolated; 2/3/2011
- 1170737; NOS ID Issues/Enhancements from Unit 1 Scram; 2/3/2011
- 1170877; More External Damage Found on 1W MPT; 2/4/2011
- 1170913; U1 2-1-11 Auto Scram Time Data Review; 2/1/2011
- 1171318; Delta-Unibus 1E MPT Inspection Results; 2/4/2011
- 1171328; Delta-Unibus 1W MPT Inspection Results; 2/4/2011
- 1171331; Startup Issue – Doghouse Repair; 2/4/2011
- 1171486; U1 VR Trip During 2-11-11 Scram; 2/1/2011
- 1171601; Log Entry Deficiencies During Unit 1 Scram; 2/1/2011
- 1171718; E SRV Non-Ads Accumulator Check Valve Needs Repair; 2/6/2011
- 1171760; 1B 33-F338B Closed without an Isolation Signal; 2/7/2011
- 1171788; 1B 21-A004E Non-Ads Accumulator did not Pass LOS-MS-R7; 2/7/2011

- 1172125; NOS ID – Issues with Unit 1 Post Transient Review; 2/7/2011
- 1172125; NOS-ID – Issues with Unit 1 Post Transient Review; 2/7/2011
- 1172171; ‘A’ RPS Breaker 1C71-S003C Not Tripping as Required; 2/7/2011
- 1172368; Unit 1 ‘A’ RR Pump Failed to Start in Slow Speed; 2/8/2011
- 1172684; LGP-1-1 Enhancement PCRA Required; 2/8/2011
- 1172689; Drywell Walkdown with NRC Items; 2/8/2011
- 1172746; Issues With Moving Rod 10-23 Off 00 Position – L1F40 Startup; 2/8/2011
- 1172780; 1CB018A Indicates Dual and Should be Full Closed; 2/9/2011
- 1172792; ‘A’ RPS Half Scram due to LPRM Spike; 2/9/2011
- 1172802; Rod Drift/Data Fault Alarms Due to Slow Withdraw Times-L1F40; 2/9/2011
- 1174257; Lessons Learned: Power Transformer Work; 2/12/2011
- 1174392; L1F40 Lessons Learned; 2/12/2011
- 1176382; Security – Unsecured Vital Door 877; 2/17/2011
- 1176408; NRC Question on Argon Cylinder; 2/15/2011
- 1176633; INR 11-03; Steam Dryer Tie Bars 5 and 28; 2/17/2011
- 1176699; RPV Head O-Ring Indication; 2/17/2011
- 1176714; L2C13 Suspect Fuel Defect; 2/17/2011
- 1176770; INR 11-04; Steam Dryer Tie Rod 21 Assembly; 2/17/2011
- 1177136; 2B Condenser Hood Inspection Results; 2/18/2011
- 1177192; Wear Observed on the 24C FW HTR Shell during FAC Inspection 2/18/2011
- 1177254; Steam Separator Shroud Head Bolt Indications; 2/18/2011
- 1177389; 2A MSR Inspection results; 2/19/2011
- 1178659; L2R13 – Foreign Material on Fuel Assembly – Update to IR 1177416
- 1180642; RM – Fuel – MR Performance Criteria Exceeded on Unit 2; 3/8/2011
- 1180739; RM-Trip 1 & Trip 2 Lights Lit when a No Trip Cond Exists; 3/4/2011
- 1176830; FME – Rivet Found Trapped in 1st Stage Nozzle Plate; 2/18/2011
- 1176927; As-Found Coupling Alignment for the 2FW01PB Not as Expected; 2/18/2011
- 1177200; U2 UAT HV Bushings Tests Indicate Degradation; 2/18/2011
- 1177246; 2E12-F005 Failed Seat Leakage test; 2/19/2011
- 1177278; Bolt Found in 2B Off-Gas Condenser; 2/19/2011
- 1177391; 2A Condenser Hood Inspection results; 2/19/2011
- 1177394; FME: Found Black material and Yellow Paint chips in #4 Contr; 2/19/2011
- 1177401; L2R13 FME – Identified FME on Assembly 43B030; 2/19/2011
- 1177416; Foreign Material Identified on Fuel Assembly in Core; 2/19/2011
- 1177435; FME Found in CV 4; 2/19/2011
- 1177479; Bottom Head Drain Line Replacement; 2/20/2011
- 1177509; Energy Injection into OOS System with Work in Progress; 2/20/2011
- 1175050; Actuator EHC Leakage; 2/14/2011
- 1175114; L2R13 Packing Leak Identified on MSIV 2B21-F028C; 2/14/2011
- 1175046; Packing Leak on BPV2; 2/14/2011
- 1175049; Packing Leak on BPV5; 2/14/2011
- 1174636; Steam leak on Valve 2B21-F422B; 2/13/2011
- 1177552; 2C Condenser Hood Inspection Results; 2/20/2011
- 1177565; FME Discovered During Disassembly of the RCIC Turbine; 2/20/2011
- 1175577; 2B33-C001B Pump/Motor Drywell Discovery Walkdown; 2/15/2011
- 1174794; Lost Position Indication on Rod 42-51; 2/14/2011
- 1174654; Reactor Level 4 Alarm when Securing 2A TDRFP; 2/13/2011
- 1174647; “A” IRM Indication Not changing During U2 Shutdown; 2/14/2011
- 1174638; Loss of 24C Heater on RR Downshift; 2/13/2011
- 1177586; Potential FME Noted During Disassembly of RCIC Turbine; 2/20/2011
- 1177657; Potential FME on Bottom of Top Guide at SRM “B” 40-45; 2/21/2011

- 1177660; 2C SRM Will Not Insert Back into Core; 2/21/2011
- 1176613; INR 11-01; 0 Degree Dryer Lower Guide Bracket; 2/17/2011
- 1176632; INR 11-02; 180 Degree Dryer Lower Guide; 2/17/2011
- 1176521; Document Findings from Generator Crawl Through; 2/16/2011
- 1176385; 2E51-F084 Failed IST Closure Test; 2/17/2011
- 1176386; 2E51-F082 Failed IST Closure Test; 2/17/2011
- 1176216; L2R13 2B21-F032A, As-Found LLRT Exceeded Admin Alarm Limit; 2/17/2011
- 1176200; L2R13 As-Found LLRT 2B33-F020 Exceeded Admin Alarm Limit; 2/17/2011
- 1175738; Small Bonnet Pressure Seal Leak on 2E51-F063 INDB STM ISOL; 2/16/2011
- 1175842; 2B21-F019 Valve with Packing Leakage; 2/16/2011
- 1178023; 2E12-F041C Fails Water Pressure Isolation Test; 2/21/2011
- 1178210; L2R13 Lessons Learned – RPV Cooldown Limited to <50 F / Hr; 2/14/2011
- 1178354; FME at base of Jet Pump 10; 2/22/2011
- 1178326; L2R13 FME paint Chip CRD Drive Tube Loc 42-39 – Retrieved; 2/22/2011
- 1178537; L2R13 CRB Exchange Mispositioning Event; 2/22/2011
- 1178573; Div 1 ARI Reset Pushbutton Contacts Stick; 2/22/2011
- 1174628; U2 Voltage Regulator Failed to Transfer to Manual; 2/14/2011
- 1174582; A SRM Indication Not Changing During Down Power; 2/13/2011
- 1178646; L2R13 – Foreign Material (3 Items) Found in U2 Reactor; 2/23/2011
- 1178651; L2R13-RX Vessel FME Retrieval Documentation; 2/23/2011
- 1176981; L2R13 FME - Foam Earpiece Floating in DSP; 2/18/2011
- 1179158; Tie-Wrap Discovered on Fuel in U2 SFP; 2/23/2011
- 1178659; L2R13 – Foreign material on Fuel Assembly – Update to IR 1177416; 2/23/2011
- 1179208; Discharge CRB May Have Dragged on Cattle chute; 2/23/2011
- 1179604; Minor Changes in Jet Pump Exam Results Since Last Outage; 2/24/2011
- 1180785; Hard Hat Cover was Retrieved from ‘C’ Condenser; 2/27/2011
- 1180823; “C” SRV Actuator Also Closed when “U” C/S Taken to Auto; 2/27/2011
- 1180924; “P” Non-Ads SRV Check Valve Failed Drop Test; 3/5/2011
- 1180927; Non Ads “F” SRV Failed Pressure Drop Test; 2/28/2011
- 1180952; U2 RFB Stopped Fuel Moves Due Noise from Speed Switch/Brake; 2/28/2011
- 1181399; Fuel Move Sheet Update Not Provided; 2/28/2011
- 1181495; L2R13 FME Identified in IR 1176109 Has Been Retrieved; 2/25/2011
- 1181509; Spurious Actuation of Division I Annunciators; 3/1/2011
- 1181537; 2E51-F046 Expected results not Obtained During LES-RI-201A; 3/1/2011
- 1181542; NCTL Issues Noted During Areva XM Fuel Assembly Inspection; 3/1/2011
- 1182539; L2R13LL: Cavity Vacuuming Post-Fuel moves Ineffective- Dirt 3/2/2011
- 1182841; Personnel Contamination Event; 3/3/2011
- 1182856; Numerous MCR Annun Windows Flashing When RCIC Init P/B Press; 3/3/2011
- 1182445; Security – Unsecured Vital Door 234; 3/2/2011
- 1183032; Single Blade guide Storage Interferes with Pool Gate Install; 3/3/2011
- 1183053; 1W MPT Cooling Bank 8-1 Excessive Noise-Excessive Vibration; 3/3/2011
- 1183161; Emergency Response Data System Down; 3/3/2011
- 1186274; Lessons Learned During L2R13 (Transformer Testing); 3/11/2011
- 1B RR Seal Injection Line Small Leak on INBD Most Piping Con; 2/5/2011

Event Reports:
- IR 1172821; Event/Issues Report of Human Performance Issue Verbal Report: Unexpected Half Scram during IRM Ranging for L1F40; 2/9/2011
- IR 1173113; Event/Issues Report: Leak Identified on 1E51-F076; 2/9/2011

Drawings:
- 29-3; Refueling Bellows Seal; 10/1999
- 29-2; Fuel Pool Cooling and Cleanup; 10/1999

Miscellaneous:
- AR 1169948; Post Transient Review of Unit 1 Trip due to Main Generator Load Reject; 2/1/2011
- AR 1178537; LaSalle Station Human Performance Alert L2R13 Control Rod Blade Mispositioning Event
- 1A CW trip – 1CW01PA; Evaluation of 1A Circulating Water Pump Trip after U1 Scram; 2/1/2011
- 1B21-F013E PORC Review; 1ESRV Tail Pipe Temperature Change from 180 to 212 degrees F. (IR 1101005); 2/1/2011
- EC 379205/NF1100048; Nuclear Fuels Transmittal of Design Information for U1 Cycle 14 Notch Worth Values with Group 3 Rod Left at 00; 2/7/2011
- L1F40 Testing of the 1E SRV; 2/1/2011
- LaSalle County Generating Station Pre-Fire Plan; Fire Zone 5B1, Unit 2, Elevation 728' 0"
- LSCS-FPR H.3-33; Unit 1 Primary Containment – Fire Zone 2J; Rev. 4
- LSCS-FPR H.4-28; Fire Zone 3I5 (Unit 2 Reactor Building); Rev. 4
- LSCS-FPR H.3-52; Unit 2 Reactor building Drain Tank Room – Fire Zone 3I6; Rev. 4
- LSCS-FPR Table H.3-2; Combustible Loading and Extinguishing Capability; Rev. 4
- LSCS-UFSAR 5.4.7.2.4; System Reliability Considerations; Rev. 15
- LSCS-UFSAR 6.3-9; ECCS Discharge Line Fill System; Rev. 18
- LSCS-UFSAR 15.B-3; Implementation Requirements; Rev. 14
- OP-AA-108-114; Post Transient Review Timeline of 2/1/2011 Scram
- Operator Logs; 2/23/2011
- Outage Control Center Log from 2/1/2011 to 2/3/2011
- QHPI - CR1172821; Quick Human Performance Investigation: Unexpected Half-Scram while Ranging Intermediate Range Monitor Switch during Startup; 2/9/2011

1R22 Surveillance Testing

Procedures:
- LOS-CS-SR1; Secondary containment Leak Rate test; Rev. 3
- LOS-DG-210; Unit 2 Integrated Division II Response Time Surveillance; Rev. 10
- LOS-DG-Q1; 0 Diesel Generator Auxiliaries Inservice Test; Rev. 53
- LOS-NB-R2; Reactor Vessel Leakage Test; Rev. 7
- LOS-PC-Q1; Primary Containment Isolation Valves Operability Test and Inservice Inspection; Rev. 43
- LOS-PC-R2; Group 1 Isolation Actuation Logic System Functional Test (With Air); Rev. 2
- LOS-RH-Q1; RHR (LPCI) and RHR Service Water Pump and Valve Inservice Test for Modes 1,2,3,4 and 5; Rev. 75
- LTS-900-6; RHR Shutdown Cooling Return Pressure Isolation Valve Water Leak Test 1(2) E12-F050A/B, 1(2) E12-F053A/B and 1(2)E12-F099A/B; Rev. 20

Action Requests:
- 1159682; Dryer Drain Valve Cycles Excessively; 1/7/2011
- 1164458; PCR LOS-DG-Q1 Att A4; 1/19/2011
- 1164421; LOS-DG-Q1 Att A4 Failure; 1/19/2011
- 1161863; 1B DG A Dryer Testing Requirements; 1/12/2011

- 1191148; MSL Tunnel Temperatures During LOS-CS-SR1; 3/23/2011
- 1191190; Hole in VR Duct Work Requires Patching; 3/23/2011

Work Orders:
- 1206236-01; RHR SDC PIV 2E12-F053A High Pressure Water Leak Test; 10/12/2010
- 1207456-01; Integrated Division II ECCS Response Time; 2/19/2011
- 1207459-01; LOS-PC-Q1 Att 2A: U-2 MSIV Operability and IST Inspections; 2/27/2011

Drawings:
- M-83; P&ID Diesel Generator Auxiliary System; Rev. AC
- M-142; P & ID Residual Heat Removal System; Rev. AX

Calculations:
- L-003068; LSCS Design Basis LOCA Analysis using RG 1.183 "Alternative Radiological Source Terms for Evaluation Design Basis Accidents at Nuclear Power Reactors"; Rev. 1

Miscellaneous:
- IPA Brief for Reactor Vessel Leakage Test per LOS-NB-R2 Rev 7 (Modified for L2R13 on 3/3/11)
- IPA Briefing Worksheet; Attachment 3; undated
- LaSalle Operations Log; 1/13/2011 – 1/14/2011
- LOCA AST Analysis Evaluation of Proposed Change Table 3.6-1; undated
- LSCS-UFSAR, 6.5-11; Standby Gas Treatment System; Rev. 18
- LSCS-UFSAR, Table 8.3-1; Loading on 4160 – Volt Buses; Rev. 17
- LSCS-UFSAR, Table 8.3-3; Diesel Generator Ratings; Rev. 0
- LSCS-UFSAR, Table 8.3-1; Loading on 4160 Volt Buses; Rev. 17
- LUCR-216; UFSAR Change Request for Alternate Source Term Implementation (EC 352505; Rev. 0); 11/9/2010
- QHPI – CR 1161863; Quick Human Performance Investigation of Missed Quarterly Surveillance for Unit 1 DG Air Dryer; 1/12/2011
- U1 1A RHR System LOS-RH-Q1 Att. 1A; Tech Spec Surveillance, Unit 1; 1/21/2011
- Unit 2 6 Trend Graph; 2/24/2011 16:20:47 – 16:21:17
- WO 1235216-01; MOV Post-Test Data Review Worksheet - LAS-2E12-F017A; 2/22/2011

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- 1113026; Hot Particle on Hi-Trac Pool Lid, 9/14/2010
- 1119041; Personnel Electronic Dose Rate Alarm, 9/28/2010
- 1126570; Increased Dose Rates from 2WE04T to General Area of Plant, 10/14/20101161225; Radiation Protection Procedure Not Implemented; 1/11/2011
- 1171077; Personnel Contamination Turbine Worker Alarmed Monitor; 2/4/2011
- 1171410; 50 Percent Work-In-Progress Review for Radiation Work Permit 10012684; Unit 1 Safety Relief Valve Work L1F40; 2/4/2011
- 1772730; Personnel Contamination Auxiliary Building Clean Area; February 9, 2011
- 1173829; Pre-Outage Clean Area Personnel Contamination; February 11, 2011
- 1174230; Personnel Contamination During Quality Control Inspection in Clean Area; 2/12/2011
- 1174679; Steam Tunnel Door Not Labeled; 2/14/2011
- 1775095; L2R13 Contamination Found on the Refuel Floor; 2/14/2011
- 1175820; Diver Left Hand was Contaminated While Working in Clean Area; 2/15/2011
- 1177619; Untimely Secured High Radiation Area Swing Gate Communications; 2/21/2011

- 1776250; Electronic Dose Rate Alarm; 2/17/2011
- 1176564; Personnel Contamination of Shoe, 2/15/2011
- 1178274; L2R13 Personnel Contamination Events; 2/22/2011
- 1178603; Personnel Contamination Events Not Communicated Timely; 2/22/2011
- 1179576; NRC Identified: Issue with Step-Off Pads and Air Sampler; 2/24/2011
- 1179705; Air Sample Head Not Loaded Properly; 2/24/2011

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- 1148883; Potential Noncompliance with NRC SRP 11.4; 12/6/2010
- 1175411; Unanticipated Electronic Dose Rate Alarm; 2/15/2011
- 1175716; Elevated Dose Rates Under-Vessel; February 16, 2011
- 1178122; Nuclear Oversight Identified: Radiation Work Permit Not Followed Resulting in Additional Dose; 2/22/2011
- 1182163; L2R13 ALARA Committee Recommendation Dose Savings Initiative; 2/21/2011
- 1179920; 2R13 Lessons Learned: Unit 2; 740' Power Lost Affected Drywell Remote Monitoring; 2/24/2011
- 1181904; Radiation Protection Correction Specialist Observations; 3/1/2011
- 1180748; L2R13 Lessons Learned on "B" Inboard Main Steam Isolation Valve Work; 2/27/2011

Miscellaneous:
- L2R13 Daily Exposure Trending Report; 2/24 to 3/1/2011
- RWP 10011653; L2R13 Drywell Safety Relief Valve Activities ALARA Plan; Rev. 2
- RWP 10011666; L2R13 Under-Vessel Nuclear Instrumentation ALARA Plan; Rev. 0
- RWP 10011667; L2R13 Control Rod Drive Pull/Put ALARA Plan; Rev. 0
- RWP 10011703; L2R13 Reactor Vessel Disassembly/Reassembly ALARA Plan; Rev. 3
- RWP 10011726; L2R13 Feed Water Heater Replacements ALARA Plan; Rev. 0
- RWP 10011683WIP; Work-In-Progress for L2R13 MUR Project; 2/24/2011
- RWP 10011653WIP; Work-In-Progress for L2R13 Drywell Safety Relief Valve Activities; 2/22/2011
- RWP 10011658WIP; Work-In-Progress for L2R13 Drywell Scaffolding Activities; 2/19/2011

4OA1 Performance Indicator Verification

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- Basis Summary Sheet for Unplanned Power Changes per 7000 Critical Hours; 11/8/2007
- LAS01V_C302 (MWT); Trending Graph of Instantaneous PPLXCTP 3483.89; 8/12 – 8/13/2010
- LAS01V_G401 (MW); Trending Graph of Generator Gross MW; 8/2010
- LaSalle Unit 1 4Q / 2010 Performance Indicators Unplanned Scrams per 7000 Critical Hours; 1/2009 – 12/2010
- Monthly Data Elements for NRC Unplanned Power Changes per 7000 Critical Hours; 2/2010 – 12/2010
- Monthly Data Elements for NRC/WANO Unit/Reactor Shutdown Occurrences;1/2010 – 12/2010
- P.8.1.2; Performance Indicator Modification/ Review Request: Unplanned Power Reduction; 9/13/2010
- Unit 1 2010 Trend Graph Generator Gross Megawatts
- Unit 2 2010 Trend Graph Generator Gross Megawatts

4OA2 Identification and Resolution of Problems

Procedures:
- EC 380607; Design Consideration Summary: RHR SDC HI Flow ISOL Time Delay Setpoint Change; Rev. 000
- EC 380607; Work Planning Instructions: RHR SDC HI Flow ISOL Time Delay Setpoint Change; 2/11/2011
- EC 380608; Design Consideration Summary: RHR SDC HI Flow ISOL Time Delay Setpoint Change; Rev. 000
- LOP-RH-07; Shutdown Cooling System Startup, Operation and Transfer; Rev. 61
- LOP-RH-08; Shutdown Cooling System Shutdown; Rev. 37
- LS-AA-120; Attachment 2, “Issue Report Level and Class Criteria”; Rev. 12
- LS-AA-125; Attachment 3, Frequently Used Condition Report Assignment Types; Rev. 15
- LS-AA-125; Attachment 4, Investigation/CR Quick Reference; Rev. 15

Action Requests:
- 120099; Primary Systems Internal Tracking; 8/21/2002
- 880069; NER NC-09-007 Yellow, Loss of SDC (Dresden); 2/12/2009
- 943883; Spurious Isolation of RHR SDC Inboard Isolation; 7/20/2009
- 1170495; 1E12-F009 Closure During LOP-RH-07; 2/3/2011

Work Orders:
- 943883-04; Document Prompt Investigation (PINV); 7/23/2009
- 943883-22; Generate IRs to create work orders to replace the remaining three relays; 9/23/2009
- 943883-26; Review/Evaluate Replacement Relay with Better Contact Reliability; 5/13/2010
- 943883-28; Revise Procedures and Administrative Controls to Defeat the Isolation Function of the 1B21H-K77 Relay while in Mode 4 for Unit 1. Complete prior to L1R13; 2/4/2010
- 943883-44; Document CA & ACIT Assignment Numbers from the RCR; 9/25/2009

Drawings:
- 1E-1-4232AM; Schematic Diagram PRI Containment and Reactor Vessel ISOL Sys. PC B2HI PT-12; Rev. R
- 1E-1-422OBD; Schematic Diagram Residual Heat Removal System RH (E12) Pt 28; Rev. N
- 1E-1-4232AF; Schematic Diagram Primary Containment and Reactor Vessel Isolation System “PC” (B21H) Part 6; Rev. R
- 1E-1-4232AG; Schematic Diagram Primary Containment and Reactor Vessel Isolation System “PC” (B21H) Part 7; Rev. P
- 1E-1-4232AK; Schematic Diagram Primary Containment and Reactor Vessel Isolation System “PC” (B21H) Part 10; Rev. U

Action Requests Resulting from NRC/IEMA Inspection:
- 1176408; NRC Question on Argon Cylinder; 2/17/2011
- 1172689; Drywell Walkdown with NRC Items; 2/8/2011
- 1161140; NRC Identification of Potential Issues; 1/11/2011
- 1169700; NRC Identified Materials in Vicinity of Transformers; 2/1/2011
- 1161129; Actuator Housing is Leaking Grease; 1/11/2011
- 1179576; NRC ID: Issue with Step-Off Pads Dirty and Air Sampler; 2/24/2011
- 1181902; IEMA Identified Compressor Gas Cylinder; 3/1/2011
- 1183997; NRC Identified Issues During DW Walkdown; 3/5/2011
- 1183814; U-2 Drywell Close-Out Inspection, 740 Elevation Only; 3/5/2011

- 1183699; NRC ID: Packing Leak on 2E12-F011A During LOS-NB-R2; 3/5/2011
- 1183700; NRC ID: Packing Leak on 2E12-F026A During LOS-NB-R2; 3/5/2011
- 1183701; NRC ID: Packing Leak on 2E12-F098A During LOS-NB-R2; 3/5/2011
- 1161287; IEMA Id: Liquid Argon Container Not Stored per SA-AA-122; 1/11/2011
- 1161438; NRC Identified Concern on Protected Pathway Determination; 1/11/2011
- 1162253; NRC Identified – Valve Gag not Attached to Valve 1C41-F029B
- 1181963; NRC ID: Piping Void IR 1179224 Incomplete Description; 3/1/2011
- 1183703; NRC ID: Packing Leak on 2B21-N039PP-SV During LOS-NB-R2; 3/5/2011
- 1183705; NRC ID: Packing Leak on 2B21-N061KK-SV During LOS-NB-R2; 3/5/2011
- 1183706; NRC ID: Packing Leak on 2B21-N061DD-SV During LOS-NB-R2; 3/5/2011
- 1183707; NRC ID: Packing Leak on 2B21-R452D-EQV During LOS-NB-R2; 3/5/2011
- 1183708; NRC ID: Packing Leak on 2B21-N408A-EQ During LOS-NB-R2; 3/5/2011
- 1183710; NRC ID: Packing Leak on 2B21-N026BA-EQ During LOS-NB-R2; 3/5/2011
- 1183722; NRC Inspections Id’d Packing Leaks Not Id’d by Exelon; 3/5/2011
- 1183894; U-2 Drywell Close-Out Inspection, 777, 796 and 807 Elevations; 3/5/2011
- 1185329; NRC Id’d: Housekeeping Issues in TDRFP Rooms; 3/9/2011
- 1190782; NRC Identified Discrepancy in Unplanned PWR Change Indicator; 3/22/2011
- 1193349; IEMA Inspector Identified: Chewing Gum in the RCA; 3/28/2011
- 1197603; NRC: 4^{th} Qtr Report 05000373(374)/2010005, 7200070/201001

Miscellaneous
- Figure 64-13: RHR System Loop B; 3/2001
- L10-122; 50.59 Screening for EC 380607 and EC 380608 – RHR Shutdown Cooling (SDC) and RPV Head Spray Operability; Rev. 000/000
- LaSalle Operations Log; 2/2/2011 – 2/4/2011
- LER 2000-005-01; PSLTR: #01-0126; Letter from Exelon Nuclear to U.S. NRC re Supplemental Licensee Event Report Recirculation Loop Temperature Failure Causes Shutdown Cooling Inoperability; 12/14/2001
- Loss of Shutdown Cooling Actions Timeline for July 20, 2009 and March 2009
- LSCS-UFSAR 15.2-9.3; Core and System Performance; Rev. 13

LIST OF ACRONYMS USED

ADAMS	Agencywide Document Access Management System
ADS	Automatic Depressurization System
ALARA	As-Low-As-Is-Reasonably-Achievable
AR	Action Request
ASME	American Society of Mechanical Engineers
CAP	Corrective Action Program
CFR	Code of Federal Regulations
CIV	Containment Isolation Valve
CRD	Control Rod Drive
DG	Diesel Generator
DRP	Division of Reactor Projects
DW	Drywell
EC	Engineering Change
ECCS	Emergency Core Cooling System
HI-STORM	Dry Storage Cask
HPCS	High Pressure Core Spray
ICDP	Incremental Core Damage Probability
ILERP	Incremental Large Early Release Probability
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IR	Inspection Report
ISFSI	Independent Spent Fuel Storage Installation
IST	Inservice Testing
LCO	Limiting Condition for Operation
LERF	Large Early Release Frequency
LPT	Low Profile Transporter
MPT	Main Power Transformer
MSO	Multiple Spurious Operations
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
NUREG	Nuclear Regulatory Guide
OSP	Outage Safety Plan
PARS	Publicly Available Records System
PBI	Plant Barrier Impairment
PI	Performance Indicator
RCIC	Reactor Core Isolation Cooling
RCMS	Reactor Control Management System
RFO	Refueling Outage
RHR	Residual Heat Removal
ROP	Reactor Oversight Process
RP	Radiation Protection
RR	Reactor Recirculation
RWP	Radiation Work Permit
SDP	Significance Determination Process
SRA	Senior Reactor Analyst
SRO	Senior Reactor Operator
SRV	Safety Relief Valve
SSC	Structure, System, and Component

TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
WO	Work Order

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Sincerely,

/RA/

Kenneth Riemer, Chief
Branch 2
Division of Reactor Projects

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

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Letter to M. Pacilio from K. Riemer dated May 11, 2011

SUBJECT: LASALLE COUNTY STATION, UNITS 1 AND 2,
NRC INTEGRATED INSPECTION REPORT
05000373/2011002; 05000374/2011002; 07200070/2011001

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