March 5, 2002

Mr. John L. Skolds, President
Exelon Nuclear
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: LASALLE COUNTY STATION
NRC INSPECTION REPORT 50-373/01-19(DRP); 50-374/01-19(DRP)

Dear Mr. Skolds:

On February 16, 2002, the NRC completed an inspection at your LaSalle County Station. The enclosed report presents the results of that inspection. The results of this inspection were discussed on February 14, 2002, with Mr. G. Barnes and other members of your staff.

The inspection was an examination by the resident inspectors of activities conducted under your license as they relate to reactor safety and to compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of a selective examination of procedures and representative records, observations of activities, and interviews with personnel.

Immediately following the terrorist attacks on the World Trade Center and the Pentagon, the NRC issued an advisory recommending that nuclear power plant licensees go to the highest level of security, and all promptly did so. With continued uncertainty about the possibility of additional terrorist activities, the Nation's nuclear power plants remain at the highest level of security and the NRC continues to monitor the situation. This advisory was followed by additional advisories and although the specific actions are not releasable to the public, they generally include increased patrols, augmented security forces and capabilities, additional security posts, heightened coordination with law enforcement and military authorities, and more limited access of personnel and vehicles to the sites. The NRC has conducted various audits of your response to these advisories and your ability to respond to terrorist attacks with the capabilities of the current design basis threat (DBT). From these audits, the NRC has concluded that your security program is adequate at this time.

Based on the results of this inspection, the inspectors identified three issues of very low safety significance (Green) that were determined to involve violations of NRC requirements. However, because of their very low safety significance and because they were entered into your corrective action program, the NRC is treating these issues as a Non-Cited Violations in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny these Non-Cited Violations, you should provide a response with a basis for your denial, within 30 days of the date of this inspection report, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, Region III; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at LaSalle County Station.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC website at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Bruce Burgess, Chief
Branch 2
Division of Reactor Projects

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

Enclosure: Inspection Report 50-373/01-19(DRP);
50-374/01-19(DRP)

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REGION III

Docket Nos:	50-373, 50-374
License Nos:	NPF-11, NPF-18
Report Nos:	50-373/01-19(DRP); 50-374/01-19(DRP)
Licensee:	Exelon Generation Company
Facility:	LaSalle County Station, Units 1 and 2
Location:	2601 N. 21st Road Marseilles, IL 61341
Dates:	December 30, 2001 through February 16, 2002
Inspectors:	E. Duncan, Senior Resident Inspector G. Wilson, Resident Inspector W. Slawinski, Radiation Protection Specialist J. Yesinowski, Illinois Department of Nuclear Safety
Approved by:	Bruce Burgess, Chief Branch 2 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000373-01-19(DRP), IR 05000374-01-19(DRP), on 12/30/01-2/16/02; Exelon; LaSalle County Station, Units 1 & 2; Surveillance Testing.

This report covers a 7-week routine resident inspection. The inspection was conducted by resident inspectors and a regional radiation specialist inspector. Three Green findings were identified which were the subject of three Non-Cited Violations. 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 NRC’s program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/OVERSIGHT/index.html. Findings for which the SDP does not apply are indicated by “No Color” or by the severity level of the applicable violation.

A. Inspector Identified Findings

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

- Green. Licensee personnel failed to identify during work activities in March 2000, that a 2A Emergency Diesel Generator (EDG) governor guard clip was missing, which if installed, would have prevented a 2A EDG testing failure on November 7, 2001.

  The issue was of very low safety significance since the 2A EDG was restored to service within the Technical Specification Allowed Outage Time and the redundant EDG was available during the entire time that the 2A EDG was inoperable. (Section 1R22)

Cornerstone: Occupational Radiation Safety

- Green. A finding and associated Non-Cited Violation of Technical Specification 5.7.4 were identified for failure to adequately control access to a high-high radiation area, post and rope-off/barricade the area, and activate a flashing light as a warning device for entry into the area (Section 2OS1.2).

  This finding was determined to be of very low safety significance since radiological consequences of the access control problem were minimal and because area radiation levels, coupled with workers proper use of electronic dosimetry and response to dosimetry alarms, precluded a substantial potential for an overexposure.

- Green. A finding and associated Non-Cited Violation of Technical Specification 5.4.1 were identified for the failure to fully implement the radiological engineering controls required by the radiation work permit and the ALARA plan, during work on a reactor recirculation system flow control valve (Section 2OS2.7).

This finding was determined to be of very low safety significance since radiation exposures to involved workers were low relative to regulatory limits, and because radiological conditions were not of a magnitude sufficient to create a substantial potential for an overexposure.

B. Licensee Identified Violations

No violations of significance were identified.

Report Details

Summary of Plant Status: Unit 1 shut down for a planned refueling outage on January 10. The outage was completed and Unit 1 was restarted and synchronized to the grid on February 4. Following power ascension activities, Unit 1 operated at or near full power for the remainder of the inspection period. Unit 2 operated at or near full power until February 5 when a problem with the generator exciter required that power be reduced to 15 percent to allow the generator be taken offline for repairs. Repairs were completed and the Unit 2 generator was synchronized to the grid on February 6. Following power ascension activities, Unit 2 operated at full power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

On January 16, 2002, the inspectors performed a walkdown of accessible portions of the Unit 1 Spent Fuel Pool Cooling (FC) system to verify system operability during maintenance activities associated with the Unit 1 Residual Heat Removal (RHR), Low Pressure Core Spray System (LPCS), and High Pressure Core Spray (HPCS) systems.

On February 4, 2002, the inspectors performed a walkdown of accessible portions of the Unit 1 HPCS system to verify system operability during maintenance activities associated with the Unit 1 Reactor Core Isolation Cooling (RCIC) system.

The inspectors reviewed documentation to determine correct system lineup. These documents included plant procedures, such as abnormal and emergency operating procedures, as well as plant drawings and valve lineup sheets. The inspectors identified any discrepancies between the existing and correct lineup.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

The inspectors walked down the following risk significant areas to identify any fire protection degradations:

- Fire Zone 7A1 - Unit 1 Division 3 Diesel Ventilation Equipment Room
- Fire Zone 7A2 - Unit 1 Division 2 Diesel Ventilation Equipment Room
- Fire Zone 7A3 - Unit 1 Division 1 Diesel Ventilation Equipment Room

- Fire Zone 8A1 - Unit 2 Division 3 Diesel Ventilation Equipment Room
- Fire Zone 8A2 - Unit 2 Division 2 Diesel Ventilation Equipment Room
- Fire Zone 2C - Unit 1 Reactor Building General Elevation 807'
- Fire Zone 4C4 - Auxiliary Building Computer Room
- Fire Zone 4C5 - Auxiliary Building Security Control Center

Emphasis was placed on control of transient combustibles and ignition sources; the material condition, operational lineup, and operational effectiveness of the fire protection systems, equipment, and features; and the material condition and operational status of fire barriers used to prevent fire damage or fire propagation.

In particular, the inspectors verified that all observed transient combustibles were being controlled in accordance with the licensee's administrative control procedures. In addition, the inspectors observed the physical condition of fire detection devices, such as overhead sprinklers, and verified that any observed deficiencies did not impact the operational effectiveness of the system. The physical condition of portable fire fighting equipment, such as portable fire extinguishers, was observed. The inspectors also verified that extinguishers were located appropriately, and that access to the extinguishers was unobstructed. Fire hoses were verified to be installed at their designated locations and the physical condition of the hoses was verified to be satisfactory and access unobstructed. The physical condition of passive fire protection features such as fire doors, ventilation system fire dampers, fire barriers, fire zone penetration seals, and fire retardant structural steel coatings was inspected and verified to be properly installed and in good physical condition.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.06)

a. Inspection Scope

The inspectors observed the licensee visual inspection of the 1B Residual Heat Removal (RHR) heat exchanger directed by Work Order (WO) 99275462. In addition, the inspectors verified that the eddy current examination results for the 1A RHR heat exchanger were appropriately evaluated against pre-established acceptance criteria, and that the frequency of inspection was sufficient to detect degradation prior to the loss of heat removal capabilities below design values.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

On February 13, 2002, the inspectors observed licensed operator requalification training conducted in accordance with Scenario Evaluation Guide (SEG) 02C1-05, "Control Rod Drift In/Main Turbine Load Pressure Switch Failure/Increased Main Turbine Vibration/Main Turbine Fails to Trip/Drywell Steam Leak/1B RHR Clogged Suction Strainer."

The inspectors verified crew performance in terms of clarity and formality of communication; the ability to take timely action in the safe direction; the prioritizing, interpreting, and verifying of alarms; the correct use and implementation of procedures, including alarm response procedures; timely control board operation and manipulation, including high-risk operator actions; the oversight and direction by the shift manager, including the ability to identify and implement appropriate Technical Specification actions such as reporting and emergency plan actions and notifications; and the group dynamics.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12)

a. Inspection Scope

The inspectors reviewed the licensee's implementation of the maintenance rule requirements, including a review of scoping, goal-setting, and performance monitoring, short-term and long-term corrective actions, and current equipment performance status. The systems selected for inspection were all classified as risk significant by the licensee's maintenance rule program. The systems evaluated were:

- Safety-Related 125 Volt Direct Current (VDC)
- Safety-Related 250 VDC

The inspectors independently verified the licensee's implementation of maintenance rule requirements for these systems by verifying that these systems were properly scoped within the maintenance rule; that all failed structures, systems, or components (SSCs) were properly categorized and classified as (a)(1) or (a)(2); that performance criteria for SSCs classified as (a)(2) were appropriate; and that the goals and corrective actions for SSCs classified as (a)(1) were appropriate. The inspectors also verified that issues were identified at an appropriate threshold and entered in the corrective action program.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessment and Emergent Work Evaluation (71111.13)

a. Inspection Scope

The inspectors reviewed the licensee's evaluation of plant risk, scheduling, configuration control, and performance of maintenance associated with planned and emergent work activities and verified that scheduled and emergent work activities were adequately managed. In particular, the inspectors reviewed the licensee's program for conducting maintenance risk safety assessments and verified that the licensee's planning, risk management tools, and the assessment and management of online risk was adequate. The inspectors also verified that licensee actions, such as establishing compensatory actions, minimizing the duration of the activity, obtaining appropriate management approval, and informing appropriate plant staff, to address increased online risk during these periods, were accomplished when online risk was increased due to maintenance on risk-significant SSCs. The following specific activities were reviewed:

- Maintenance risk assessment for work planned during the week of December 31, 2001.
- Maintenance risk assessment for work planned during the weeks of February 4 and 11, 2002.

b. Findings

No findings of significance were identified.

1R14 Non-Routine Evolutions (71111.14)

.1 Unit 1 Reactor Water Level Control System (RWLCS) Modification Testing

a. Inspection Scope

The inspectors observed Unit 1 RWLCS power ascension modification testing prescribed by design change package (DCP) 9900072. The testing was conducted to verify that modifications which installed a digital RWLCS were effective.

b. Findings

No findings of significance were identified.

.2 (Closed) Unresolved Item (URI) 50-374/01017-01: Adequacy of Corrective Actions to Address Reactor Core Isolation Cooling (RCIC) Open Check Valve Indication.

This issue was reviewed and is discussed in NRC Inspection Report 50-373/01016; 50-374/01016. This Unresolved Item is closed.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed selected Operability Evaluations (OEs) and Engineering Changes (ECs) of degraded and non-conforming conditions affecting mitigating systems and barrier integrity to ensure that operability was properly justified and the component or system remained available, such that no unrecognized increase in risk had occurred. The following evaluations were reviewed:

- OE 01-21: Use of Non-Conservative Core Monitoring (Powerplex) Steam Tables
- EC 49520: Safety Relief Valve Tailpipe Thickness

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

.1 Individual Impact Assessment

a. Inspection Scope

The inspectors reviewed Operator Workarounds (OWAs) and Operator Challenges (OCs) to identify any potentially adverse impact on the function of mitigating systems or the ability to implement an abnormal or emergency operating procedure. The following items were reviewed:

- OC 321: Unit 1 RHR Service Water (RHRSW) Keepfill Failures
- OC 332: Unit 2 Main Stop Valves Fail to Cycle
- OWA 334/335: Condensate Storage Tank Ruptures

b. Findings

No findings of significance were identified.

.2 Operator Workarounds - Cumulative Effects Assessment

a. Inspection Scope

The inspectors reviewed the cumulative effects of all documented OWAs and OCs on reliability, availability, and potential for mis-operation of a system; the potential for increasing initiating event frequency or impact on multiple mitigating systems; and the ability of operators to respond in a correct and timely manner to plant transients and accidents.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

a. Inspection Scope

The inspectors reviewed Design Change Package 9900187 which removed and/or abandoned in place instruments and associated tubing originally installed to monitor Safety Relief Valves, downcomers, and suppression pool response during initial startup testing of Unit 1. The testing was to validate the calculated response of the SRVs, downcomers, and the suppression pool to SRV actuation. Various parameters were measured including pressure, temperature, acceleration, and strain using specially installed instrumentation.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed and observed the following post-maintenance testing activities involving risk significant equipment:

- WR 99007715 HPCS Post-Maintenance Operability Run
- WR 00344975 1B RHRSW Operability Run
- WR 99118105-02 1E51-F066 RCIC Inboard Check Valve Operability

During post-maintenance testing observations, the inspectors verified that the test was adequate for the scope of the maintenance work which had been performed, and that the testing acceptance criteria was clear and demonstrated operational readiness consistent with the design and licensing basis documents. The inspectors also verified that the impact of the testing had been properly characterized during the pre-job briefing; the test was performed as written and all testing prerequisites were satisfied; and that the test data was complete, appropriately verified, and met the requirements of the testing procedure. Following the completion of the test, the inspectors verified that the test equipment was removed, and that the equipment was returned to a condition in which it could perform its safety function.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

a. Inspection Scope

The inspectors observed the performance of LaSalle Unit 1 Refueling Outage L1R09 and evaluated licensee outage activities to ensure that the licensee considered risk in developing the outage schedule; adhered to administrative risk reduction methodologies developed to control plant configuration; developed mitigation strategies for losses of key safety functions; and adhered to the operating license and Technical Specification requirements that ensured defense-in-depth. The following specific outage-related activities were accomplished:

- Outage Plan Review

The inspectors reviewed the licensee's outage control plan and verified that the licensee had appropriately considered risk, industry experience, and previous site-specific problems. The inspectors also confirmed that contingency plans for losses of key safety functions had been established.

- Monitoring of Shutdown Activities

The inspectors observed the Unit 1 shutdown to Refueling Outage L1R09 and verified that the plant was operated in accordance with regulatory requirements and plant procedures. In particular, the inspectors verified that cooldown restrictions were followed.

- Licensee Control of Outage Activities

The inspectors verified that the licensee appropriately managed the configuration of equipment during the outage to ensure that a defense-in-depth commensurate with the outage risk plan for key safety functions and applicable Technical Specifications was maintained. The inspectors also verified that outage activities were appropriately managed. In particular, out-of-service activities were reviewed to ensure that tags were properly hung to support the out-of-service. Reactor coolant system instrumentation was verified to be configured to provide adequate indication of reactor vessel pressure, temperature, and level. In addition, the inspectors routinely observed decay heat removal system parameters and verified that decay heat removal systems were functioning properly. The inspectors verified that the status and configuration of electrical systems met Technical Specification requirements and the licensee's outage risk plan. Switchyard activities were verified to be controlled appropriately. The inspectors verified that flow paths, configurations, and alternative means for inventory addition and decay heat removal were consistent with the outage risk plan. The inspectors verified that the licensee controlled reactivity and maintained secondary containment in accordance with Technical Specifications.

- Refueling Activities

The inspectors verified that fuel handling operations were conducted in accordance with Technical Specifications and approved procedures. The inspectors also verified that the location of fuel assemblies was tracked from core offload through core reload.

- Monitoring of Heatup and Startup Activities

The inspectors verified that Technical Specifications, license conditions, and other prerequisites, commitments, and administrative procedure prerequisites for mode changes were met prior to changing modes or plant configurations. The inspectors conducted a walkdown of containment prior to restart and verified that debris had not been left which could adversely impact the Emergency Core Cooling System (ECCS) suction strainers.

- Identification and Resolution of Problems

The inspectors verified that the licensee identified problems related to refueling outage activities at an appropriate threshold and entered them into the corrective action program.

b. <u>Findings</u>

No findings of significance were identified.

1R22 <u>Surveillance Testing</u>

a. <u>Inspection Scope</u>

The inspectors observed surveillance testing on risk-significant equipment and verified that the SSCs selected were capable of performing their intended safety function and that the surveillance tests satisfied the requirements contained in Technical Specifications, the Updated Final Safety Analysis Report (UFSAR), and licensee procedures. During surveillance testing observations, the inspectors verified that the test was adequate to demonstrate operational readiness consistent with design and licensing basis documents, and that the testing acceptance criteria was clear. The inspectors also verified that the impact of the testing had been properly characterized during the pre-job briefing; the test was performed as written and all testing prerequisites were satisfied; the test data was complete, appropriately verified, and met the requirements of the testing procedure; and that the test equipment range and accuracy was consistent with the application, and the calibration was current. Following the completion of the test, the inspectors verified that the test equipment was removed, and that the equipment was returned to a condition in which it could perform its safety function.

The following surveillance testing activities were observed:

- LaSalle Technical Surveillance (LTS) 100-21, “Primary Containment Chill Water Isolation Valves Local Leak Rate Test”

- LTS-800-103, “1B Emergency Diesel Generator (EDG) 1E22-S001 Start and Load Acceptance Test”

- LTS-600-41, “Primary Containment Inspections for ECCS [Emergency Core Cooling System] Suction Strainer Debris Sources”

- LOS-RH-Q1, “2B RHR System Operability and Inservice Test”

- LOS-RI-R3, “Unit 1 Reactor Core Isolation Cooling System Pump Operability Test”

In addition, the inspectors reviewed the surveillance testing results following an identified failure of the routine monthly 2A EDG surveillance test on November 7, 2001, accomplished in accordance with LOS-DG-M2, Attachment 2A, “2A Emergency Diesel Generator Operability Test.”

b. Findings

One Green finding and an associated Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI, “Corrective Actions,” was identified for failure to recognize, during work activities conducted in March 2000, that a governor guard clip was missing, which if installed, would have prevented a sequence of events which led to a 2A EDG failure.

Description of Issue

On November 7, 2001, during the performance of routine monthly surveillance LOS-DG-M2, Attachment 2A, “2A Emergency Diesel Generator Operability Test,” operators identified that the governor synchronizer (speed setting) knob, which was normally operated by hand, was difficult to turn and chose to use a crescent wrench to accomplish the task. The 2A EDG was successfully started at an initial speed of about 500 revolutions per minute (rpm). However, attempts to increase engine speed to 900 rpm, as specified by the surveillance test, were unsuccessful. The EDG was shutdown to conduct troubleshooting in accordance with Work Request (WR) 00378515, “2A Emergency Diesel Generator (EDG) Governor Repair.”

During the troubleshooting activities, electrical maintenance department (EMD) personnel identified that the stiffness identified during initial attempts to set the governor synchronizer (speed setting) knob was caused by the governor friction drive assembly being out of tolerance. In addition, EMD personnel identified that the speed indicating gear high speed stop was jammed into the governor intermediate gear, preventing the speed synchronizing motor from increasing engine speed. Also, EMD personnel identified that a guard clip designed to prevent direct contact between the high speed stop and the intermediate gear was missing. The presence of this clip would have prevented the direct contact between the high speed stop and the intermediate gear teeth, which would have allowed the 2A EDG to start and run properly.

As part of the licensee’s immediate corrective actions, the missing guard clip was replaced and the speed indicating gear was re-aligned to prevent the high speed stop from contacting the intermediate gear during the surveillance. In addition, LOS-DG-M2 was revised to reduce the number of turns required to set initial engine speed and thereby prevent challenging the integrity of the guard clip. Also, the governor friction drive was adjusted to within specified tolerance limits to allow manipulation of the speed setting knob without the aid of a crescent wrench.

Inspector Review

The inspectors reviewed all of the information identified above and concluded that operators failed to adequately question the difficulty experienced during attempts to initially manipulate the speed setting knob and, as a result, rotated the speed setting knob to a point that the speed indicating gear high speed stop jammed into the intermediate gear, which rendered the EDG unable to increase speed and therefore inoperable. In addition, the difficulty in rotating the speed setting knob was identified in CR L2001-05819, but was incorrectly attributed to being due solely to increased governor temperature during operation of the EDG. Also, a prior opportunity to identify that the guard clip was missing existed during the performance of Work Order (WO) 990117039, which removed the governor low speed stop from the intermediate gear in March 2000.

Significance Evaluation

The inspectors reviewed this issue against the guidance contained in Appendix B, "Thresholds of Documentation," of Inspection Manual Chapter (IMC) 0610*, "Power Reactor Inspection Reports." In accordance with the Group 1 questions, the inspectors determined that the issue did have a credible impact on safety since the 2A EDG was rendered inoperable and maintenance was required to repair the EDG. As a result, the inspectors reviewed this issue against the Group 2 questions and determined that since the 2A EDG was a train in an accident mitigation system, the issue warranted further review in accordance with IMC 0609 "Significance Determination Process" (SDP). The inspectors conducted this review utilizing "SDP Phase 1 Screening Worksheet For IE [Initiating Events], MS [Mitigating Systems], and BI [Barrier Integrity] Cornerstones." The inspectors determined that although the operability of the 2A EDG was affected, because the loss of the 2A EDG did not exceed the Technical Specification Allowed Outage Time (AOT), that the Unit 2 Division 1 EDG was available, and that no weather-related impact existed, that the finding screened out as Green.

Enforcement Actions

10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," requires that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and non-conformances are promptly identified and corrected. The failure to identify that a governor guard clip was missing, which led in part to a sequence of events which rendered the 2A EDG inoperable, was an example where the requirements of 10 CFR 50, Appendix B, Criterion XVI, were not met and was a violation. However, because of the very low safety significance of the item and because the licensee has included this item in their corrective action program (Condition Report 00082092), this corrective action violation is being treated as a Non-Cited Violation (NCV 50-373/01019-01(DRP); 50-374/01019-01(DRP)).

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed Temporary Modification 334065 which installed an alternate method of Shutdown Range Vessel Level Indication. The inspectors reviewed the associated 10 CFR 50.59 safety evaluation against the system design basis documentation, including the Updated Final Safety Analysis Report (UFSAR) and verified that the temporary modification had not adversely impacted reactor vessel level indication. The inspectors also conducted a walkdown of the temporary modification and compared the installed configuration against the configuration prescribed in design drawings.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

.1 Plant Walkdowns and Radiological Boundary Verification

a. Inspection Scope

The inspector conducted walkdowns of selected radiologically controlled areas to verify the adequacy of radiological boundaries and postings. The inspector reviewed both the administrative controls specified in radiation work permits (RWPs) and the physical controls for access to these areas, and assessed worker adherence to these controls through direct observation. Specifically, the inspector walked down several radiologically significant work area boundaries (high and high-high radiation areas) in the Turbine Building and Unit 1 Reactor Building including the drywell, and performed confirmatory radiation measurements to verify that these areas and selected radiation areas were properly posted and controlled in accordance with 10 CFR Part 20 and licensee Technical Specifications. Additionally, the inspector reviewed a high-high radiation area access control incident that occurred in the Unit 1 drywell on January 11, 2002, and assessed performance indicator applicability for the incident and the adequacy of the licensee’s problem identification, extent of condition review and corrective actions (Section 20S1.2).

b. Findings

No findings of significance were identified.

.2 Review of High-High Radiation Area Access Control Problem

a. Inspection Scope

The inspector reviewed a high-high (locked high) radiation area access control incident that occurred during the Unit 1 refueling outage on January 11, 2002, associated with elevated dose rates in the drywell. Specifically, the inspector reviewed the licensee's prompt investigation of the incident, performed a parallel review to corroborate certain information, and discussed the incident with radiation protection (RP) management.

b. Findings

A Green finding and an associated Non-Cited Violation (NCV) were identified for the failure to maintain positive control over entry to a high-high radiation area located in an upper elevation of the Unit-1 drywell.

Shortly after midnight on January 11, 2002, two workers involved with safety relief valve (SRV) removal on the 777' elevation of the drywell experienced high dose rate alarms on their electronic dosimetry (ED), which were set to alarm at 300 mrem/hour. The highest radiation levels recorded by the workers' dosimetry was 520 mrem/hour. The workers immediately left the work area and reported the problem to the RP staff, as required by station procedure. The two workers were involved in other SRV work earlier in the shift, and their total integrated doses for the work that evening were 78 and 76 mrem.

Surveys performed by the RP staff after the workers experienced the ED alarms showed that general work area radiation levels had increased about five-fold on the 777' drywell elevation from the time the area was previously surveyed by RP staff about 20 hours earlier. Also, an area accessible to the workers adjacent to the elbow of the flange of the "C" low pressure coolant injection (LPCI) line showed radiation levels up to 2000 mrem/hour at a 30 centimeter distance. The identically configured "A" and "B" LPCI elbows did not exhibit similarly elevated radiation levels.

Inspector and licensee review of outage activities that took place during the time the drywell radiation levels increased, as well as a review of historical outage drywell survey data, could not conclusively determine the cause for the increased radiation levels. However, the licensee speculated that full flow injection tests of the "A" and "B" LPCI systems (which inject at a high flow rate into the vessel annulus region) completed about 10 hours after drywell area radiation levels were initially established on January 10, may have caused high activity particulate material to migrate into the open nozzle of the "C" LPCI system and produced the elevated dose rates on that elbow.

While the main drywell entrance was posted as a high radiation area, the high-high radiation area identified near the flange elbow of the "C" LPCI system was not posted accordingly, the flange area was not roped-off or barricaded, and a flashing light was not activated as a warning device for entry into the area. Consequently, access to the high-high radiation area was not properly controlled for up to approximately 20 hours while SRV removal activities intermittently took place.

This issue, if not corrected, would become a more significant concern should high-high radiation areas not be identified and access to them properly maintained. Also, the issue involved unintended dose to those workers that encountered the elevated dose rate areas which resulted from conditions contrary to technical specifications. Therefore, the issue represented a finding which was evaluated using the significance determination process (SDP) for the occupational radiation safety cornerstone. Since the inspector concluded that area radiation levels, coupled with workers proper use of electronic dosimetry and response to dosimetry alarms precluded a substantial potential for an overexposure, the issue was determined to be of very low safety significance.

Technical Specification 5.7.4 requires that high-high radiation areas (areas accessible to personnel with radiation levels greater than 1000 mrem/hour) that are located within large areas where no enclosure exists for purposes of locking be roped off, conspicuously posted, a flashing light be activated as a warning device, and positive control over each individual entry into the area be maintained. The failure to maintain positive control over entry to the high-high radiation area located on the 777' elevation of the Unit-1 drywell on January 11, 2002, is a violation of Technical Specification 5.7.4. However, because the licensee documented this issue in its corrective action program (CR # 90153) and because the violation is of very low safety significance, the violation is being treated as a NCV (NCV 50-373/01019-02;50-374/01019-02).

2OS2 As-Low-As-Is-Reasonably-Achievable (ALARA) Planning and Controls (71121.02)

.1 Radiation Dose Goals and Trending

a. Inspection Scope

The inspector reviewed the station’s outage exposure data for the last several refueling outages to establish its prior performance relative to the industry. Job specific and cumulative exposure performance and exposure trends for the first week of the scheduled three week Unit 1 refueling outage (L1R09) were reviewed to assess the licensee’s current dose performance compared to pre-outage exposure goals and projections. The inspector also reviewed the licensee’s dose forecasting practices for radiologically significant jobs scheduled to take place during the outage, to determine if adequate technical bases for outage dose estimates existed and to determine if outage experiences, craft work group defined job scope, resource estimates and industry operating experiences were used to establish reasonable dose estimates. Additionally, the inspector reviewed the effectiveness of the RP organization’s exposure tracking for the outage, to verify that the licensee could identify problems with its exposure performance and take actions to address identified deficiencies.

b. Findings

No findings of significance were identified.

.2 Radiological Work Planning

a. Inspection Scope

The inspector reviewed the licensee's procedure for ALARA Plan development, and evaluated several L1R09 ALARA plans to verify consistency with the procedure and to assess their overall adequacy relative to both licensee and industry practices. Specifically, the inspector selected the following outage jobs that were projected to accrue in excess of 5 rem, and assessed the adequacy of the radiological controls and the work planning for each:

- Safety Relief Valve (SRV) Modification, Removal and Replacement
- Source Range Monitor/Intermediate Range Monitor Replacement, Connector and Cable Work
- Control Rod Drive (CRD) Replacements
- Drywell Nozzle In-Service-Inspection (ISI) and Reactor Vessel Welds
- 1B33-F060A Valve Repair/Rebuild
- Disassemble and Reassemble Reactor Vessel, In-Vessel-Inspections, Fuel Moves, and Reactor Cavity and Dryer Separator Pit Decontamination
- Drywell Permanent Shielding Modification
- Drywell Scaffolding Installation and Removal

The inspector reviewed the RWP and the ALARA plan developed for each job, and assessed the radiological engineering controls and other dose mitigation techniques specified in these documents to verify that plans were completed in compliance with procedure and included appropriate controls to reduce dose. These documents were also reviewed to determine if job history files, licensee lessons learned, and industry operating experiences were adequately integrated into each work package. Additionally, the inspector discussed ALARA planning with several RP staff, to verify that adequate interfaces between contractors, station work groups, and ALARA staff occurred during job planning.

b. Findings

No findings of significance were identified.

.3 Implementation of ALARA Controls and Radiological Oversight of Work

b. Inspection Scope

The inspector selected the following high exposure or high radiation area jobs conducted during the outage and reviewed the execution of the ALARA program:

- Drywell In-Service-Inspection Activities
- Drywell Permanent Shield Modification
- Control Rod Drive Replacement

The inspector discussed job performance with RP staff. Also, total effective dose equivalent (TEDE) ALARA evaluations completed for a variety of under-vessel work activities, including CRD replacement, were assessed for technical adequacy. Work in progress reports and radiological survey data for these and other selected jobs, as applicable, were also reviewed to assess their adequacy and consistency with licensee procedures. The inspector attended the pre-job ALARA brief for permanently shielding the bottom head drain and ring headers. The inspector selectively reviewed briefing information for other work to verify that radiological and work execution information were exchanged effectively. The inspector evaluated the licensee's radiological engineering controls utilized at selected work locations to determine if the controls were consistent with those specified in the ALARA plans. The inspector also observed and questioned both RP staff that provided job coverage for various outage activities, and radiation workers (radworkers) involved in CRD removal and other drywell work, to verify that they had adequate knowledge of radiological work conditions and ALARA controls.

b. Findings

No findings of significance were identified.

.4 Verification of Exposure Estimates and Exposure Tracking Systems

a. Inspection Scope

The inspector reviewed the methodology and assumptions used by the ALARA group to develop L1R09 dose estimates, and compared collective outage and individual job dose performance during the initial week of the outage to assess dose performance and determine the accuracy of pre-outage projections. The inspector reviewed job dose history files and dose reductions anticipated through lessons learned, to verify that they were appropriately used to forecast outage doses. In particular, the inspector discussed with the ALARA staff several jobs anticipated to expend greater than 25 rem and that would each exceed original dose projections, to determine whether the licensee had identified those factors that contributed to additional dose and/or inaccurate dose estimates. The inspector also reviewed the process used to revise dose estimates and capture lessons learned, to verify compliance with the licensee's ALARA procedure. As of January 17, 2002, the licensee had recorded an outage exposure of approximately 176 rem compared to its estimate of about 154 rem for that stage of the outage, and projected that its original outage dose estimate would be exceeded by approximately 20 percent. Selected work in progress reports were examined to evaluate the licensee's ability to assess the effectiveness of a job, to execute its ALARA plan, and to institute changes in work plans, if warranted. The licensee's exposure tracking system was also reviewed to determine if the level of exposure tracking detail, exposure report timeliness, and report distribution were sufficient to support the control of outage exposures.

b. Findings

No findings of significance were identified.

.5 Source Term Reduction and Control

a. Inspection Scope

The inspector reviewed the status of the licensee's source term reduction program, focusing on those initiatives taken for the outage such as hydrolazing, flushing, de-sludging, and installation of permanent and temporary shielding. The inspector also evaluated aspects of the licensee's water chemistry control program and its impact on source term reduction, to determine whether the program was implemented consistent with industry initiatives and fuel vendor recommendations specified in recent Service Information Letters. Initiatives such as feedwater iron reduction and optimization of reactor water zinc were reviewed to verify that the licensee continued to pursue opportunities for source term control. Noble metal injection was initiated for both operating units during the prior run cycles and currently both units utilized hydrogen injection, depleted zinc oxide addition and the noble metals coating as initiatives to mitigate inter-granular stress corrosion cracking and to reduce the plant's source term. The licensee's overall source term reduction program was assessed to verify it included other initiatives such as cobalt reduction through stellite control and to verify that a viable source term control program was in place and progressing.

b. Findings

No findings of significance were identified.

.6 Identification and Resolution of Problems

a. Inspection Scope

The inspector reviewed the results of an RP self-assessment completed as part of an outage ALARA readiness review and condition reports (CRs) generated by the RP staff during the outage, to evaluate the effectiveness of the RP organization's ability to identify and correct problems. The inspector also reviewed outage related Nuclear Oversight Department field observations, CRs generated by station departments other than RP and two prompt investigation reports related to outage issues, to verify that the licensee adequately identified individual problems and trends, determined contributing causes and extent of condition, and developed appropriate corrective actions.

b. Findings

No findings of significance were identified.

.7 Review of a Radiological Intake Incident During Work on a Reactor Recirculation System Flow Control Valve

a. Inspection Scope

The inspector reviewed the circumstances associated with a radiological incident that occurred during the Unit-1 refueling outage on January 17, 2002, while a highly contaminated valve assembly was being dismantled in the mechanical maintenance hot shop. Specifically, the inspector reviewed the licensee's prompt investigation and exposure evaluation reports, the ALARA plan and RWP that governed the work activity, and discussed the incident with RP staff. The inspector also independently calculated the committed effective dose equivalent (CEDE) assigned to the workers and evaluated the potential for an exposure in excess of regulatory limits, to verify the accuracy of the licensee's assessments.

b. <u>Findings</u>

A Green finding and an associated NCV were identified for the failure to follow the RWP and fully implement the radiological engineering controls required by the ALARA plan, during work on a Unit 1 reactor recirculation system flow control valve (1B33-FO60A).

On January 17, 2002, portions of the 60A valve, including the stuffing box and actuator cartridge/shaft assembly, were transported from the drywell to the mechanical maintenance hot shop. The parts were to be disassembled and decontaminated in the hot shop, and subsequently transported back to the drywell and the valve reassembled. Valve internals were highly contaminated (20 mRad/hour removable contamination or approximately 1 million dpm/100 square cm), so specific contamination control criteria were provided in the ALARA plan relative to the disassembly and decontamination of the valve parts in the maintenance shop.

After the valve's stuffing box was disassembled by contract pipe fitters and valve bolts and other parts were decontaminated, the job foreman noted that the actuator cartridge and valve stem had not been broken-down and decontaminated as originally planned. The ALARA analyst for the job was contacted and it was agreed that the valve shaft was to be removed from the actuator and these parts decontaminated. A come-along was to be attached to the cartridge, and the weight of the shaft was expected to separate the cartridge from the shaft. The three person pipe fitter work crew were instructed in the method to disassemble the components using the come-along, but were told not to remove the packing material from the assembly as this would be performed later. Despite these instructions and because the method discussed was not successful, the work crew used a sledgehammer to dislodge the shaft from the cartridge. This caused some of the packing material to be dislodged from the cartridge, which the workers failed to recognize as a radiological control problem. The problem was exacerbated when the workers pried out the remainder of the packing material from the cartridge using a screwdriver, placed the packing material in a bag, then squeezed the bag to allow air to escape so the bag could be closed. As a result of these actions, the workers were contaminated and small intakes to all three members of the work crew occurred.

Although the ALARA plan and RWP specifically addressed the contamination controls that were to be employed during valve disassembly, few of the controls were actually implemented because the work crew couldn't implement some of them and didn't understand others. The ALARA plan also required that RP personnel be continuously present during the work activity; however, RPT coverage was only provided during initiation of the work and after most of the packing material was dislodged. The RPT allowed the removal of the packing material to continue, even though it was recognized that work took place beyond the scope of what was planned. The RPT believed it was in the best interest of the ALARA concept to continue the work, since most of the packing material had already been removed when he responded to the job site after the hammering was overheard.

The work was completed and the crew left the mechanical maintenance shop not recognizing the radiological problem that had been created. The workers alarmed the personnel contamination monitors as they attempted to exit the RCA. Positive nasal smears prompted whole body count analyses of all three workers, each showing small intakes of radioactive material. Further evaluation disclosed intakes through the ingestion pathway, and the maximum dose was calculated at 26 mrem CEDE.

The licensee's evaluation of the incident identified several aspects of the ALARA plan and RWP (#01010858) for the work which were not implemented, as follows:

- Work was not stopped and RP notified immediately when the work plan failed.
- RP was not contacted to conduct a survey before the packing material was removed.
- Continuous RPT coverage was not provided during valve disassembly.
- The stuffing box actuator cartridge assembly was not unpacked underwater or otherwise kept coated with a liberal application of ultra-gel.
- A high efficiency particulate air equipped ventilation system was not used during disassembly and decontamination of the valve components.

Other contributing factors to the incident as identified by the licensee included:

- Work conducted outside the scope of what was briefed.
- The RPT that provided intermittent coverage was not aware of the engineering controls specified in the ALARA plan and did not participate in the pre-job brief.
- Poor decisions were made by the RPT and work was allowed to continue despite potential radiological problems.
- No contingency plans were developed as part of the ALARA plan should the packing material be dislodged.
- Several communication problems occurred between the work crew, work crew supervision and the RP staff.

This issue, if not corrected, would become a more significant concern should radiological engineering controls specified in the RWP or ALARA plan not be fully implemented. Also, the issue involved unintended dose (intakes) to those workers that were involved in the valve disassembly which resulted from actions and conditions contrary to the RWP and the ALARA plan that governed the work. Therefore, the issue represents a finding which was evaluated using the SDP for the occupational radiation safety cornerstone. Since radiation exposures to involved workers were low relative to regulatory limits and because radiological conditions (removable contamination levels) were not of a magnitude sufficient to create a substantial potential for an overexposure (as determined by the licensee and verified by the inspector), the issue was determined to be of very low safety significance.

Technical Specification 5.4.1 requires, in part, that procedures be established, implemented and maintained that cover the activities recommended in Regulatory Guide 1.33, Revision 2, Appendix A, which includes procedures for ALARA program implementation. Procedure RP-AA-401, “Operational ALARA Planning and Controls,” requires in Section 3 that individual workers adhere to the ALARA plan and RWP requirements including in-field application of the plan, and in Section 4 that the ALARA plan be reviewed by all involved workers prior to the work. The failure to fully implement and adhere to the ALARA plan and RWP requirements and the failure of the RPT to review the ALARA plan prior to the work is a violation of Technical Specification 5.4.1. However, because the licensee documented this issue in its corrective action program (CR # 91224) and because the violation is of very low safety significance, the violation is being treated as a NCV (NCV 50-373/01019-03 and 50-374/01019-03).

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

Cornerstone: Mitigating Systems

a. Inspection Scope

The inspectors reviewed reported 4th quarter data for the Unit 1 and Unit 2 Residual Heat Removal (RHR) System Unavailability performance indicator and the Unplanned Power Changes Per 7000 Critical Hours performance indicator. The inspectors utilized the performance indicator definitions and guidance contained in Nuclear Energy Institute (NEI) 99-02, “Regulatory Assessment Indicator Guideline,” Revision 2.

b. Findings

No findings of significance were identified.

4OA6 Meetings

Exit Meeting Summary

A preliminary exit meeting was held with Mr. G. Barnes and other members of licensee management on January 18, 2002 and followup telephone conversations were held on January 30 and February 13, 2002 with the Radiation Protection Manager to discuss Access Controls for Radiologically Significant Areas and ALARA Planning/Controls.

The inspectors presented the final inspection results to Mr. G. Barnes and other members of licensee management on February 14, 2002. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

KEY POINTS OF CONTACT

Exelon

G. Barnes, Site Vice President
D. Czufin, Site Engineering Manager
D. Enright, Operations Manager
J. Estes, Radiological Engineering Manager
F. Gogliotti, Design Engineering Supervisor
J. Henry, System Engineering Manager
K. Hobbs, Radiation Protection Manager
W. Riffer, Regulatory Assurance Manager
M. Schiavoni, Station Manager
C. Wilson, Station Security Manager

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-373/01019-01;50-374/01019-01	NCV	Inoperable 2A EDG
50-373/01019-02;50-374/01019-02	NCV	Failure to adequately control access to a high-high radiation area, post and rope-off/barricade the area, and activate a flashing light as a warning device for entry into the area (Section 2OS1.2).
50-373/01019-03;50-374/01019-03	NCV	Failure to fully implement the radiological engineering controls required by the ALARA plan during work on a reactor recirculation system flow control valve (Section 2OS2.7)

Closed

50-373/01019-01;50-374/01019-01	NCV	Inoperable 2A EDG
50-373/01019-02;50-374/01019-02	NCV	Failure to adequately control access to a high-high radiation area, post and rope-off/barricade the area, and activate a flashing light as a warning device for entry into the area (Section 2OS1.2).
50-373/01019-03;50-374/01019-03	NCV	Failure to fully implement the radiological engineering controls required by the ALARA plan during work on a reactor recirculation system flow control valve (Section 2OS2.7).
50-374/01017-01	URI	RCIC Check Valve Indication Corrective Actions

Discussed

None

LIST OF ACRONYMS USED

ADHR	Alternate Decay Heat Removal
ALARA	As Low As Is Reasonably Achievable
ASME	American Society of Mechanical Engineers
AOT	Allowed Outage Time
BI	Barrier Integrity
CEDE	Committed Effective Dose Equivalent
CR	Condition Report
CRD	Control Rod Drive
DBT	Design Basis Threat
DCP	Design Change Package
DRP	Division of Reactor Projects
EC	Engineering Change
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EMD	Electrical Maintenance Department
IE	Initiating Events
IMC	Inspection Manual Chapter
ISI	In-Service-Inspection
L1R09	LaSalle Station Unit 1- Ninth Refueling Outage
LER	Licensee Event Report
LES	LaSalle Electrical Surveillance
LGP	LaSalle General Procedure
LOP	LaSalle Operating Procedure
LOS	LaSalle Operating Surveillance
LTP	LaSalle Technical Procedure
LTS	LaSalle Technical Surveillance
MS	Mitigating Systems
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
OC	Operator Challenge
OE	Operability Evaluation
OWA	Operator Workaround
PARS	Publicly Available Records
Radworker	Radiation Worker
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RP	Radiation Protection
rpm	revolutions per minute
RWLCS	Reactor Water Level Control System
RWP	Radiation Work Permit
SDP	Significance Determination Process
SRV	Safety Relief Valve
SSC	Structure, System, or Component
TEDE	Total Effective Dose Equivalent
UFSAR	Updated Final Safety Analysis Report

LIST OF ACRONYMS USED (con't)

VDC	Volt Direct Current
WO	Work Order
WR	Work Request

LIST OF DOCUMENTS REVIEWED

<u>1R04 Equipment Alignment</u>

LOP-FC-01E	“Unit 1 Fuel Pool Cooling System Electrical Checklist.
LOP-FC-01M	“Unit 1 Fuel Pool Cooling System Mechanical Checklist.
LOP-RI-01E	“Unit 1 Reactor Core Isolation Cooling System Electrical Checklist.
LOP-RI-01M	“Unit 1 Reactor Core Isolation Cooling System Mechanical Checklist.

<u>1R05 Fire Protection</u>

Appendix H	Updated Final Safety Analysis Report, Revision 13.

<u>1R07 Heat Sink Performance</u>

LaSalle Technical Surveillance (LTS) 200-17, “RHR Heat Exchanger Thermal Performance Monitoring,” Revision 5, dated May 5, 2000.

LaSalle Technical Procedure (LTP) 100-5, “Service Water Component Inspection Guideline,” Revision 3, dated February 15, 2000.

<u>1R11 Operator Licensing Requalification</u>

SEG 02C1-05	Control Rod Drift In/Main Turbine Load Pressure Switch Failure/Increased Main Turbine Vibration/Main Turbine Fails to Trip/Drywell Steam Leak/1B RHR Clogged Suction Strainer

<u>1R12 Maintenance Rule Implementation</u>

Functional Failure and Availability Data Sheets

<u>1R13 Maintenance Risk Assessment and Emergent Work Evaluation</u>

LaSalle 7-Day Look-Ahead Schedules (Various)

<u>1R14 Personnel Performance During Nonroutine Plant Evolutions</u>

DCP 9900072	Unit 1 RWLCS Power Ascension Modification Testing

<u>1R15 Operability Evaluations</u>

OE01-21	Use of Non-Conservative Core Monitoring (Powerplex) Steam Tables, dated December 21, 2001.

Grand Gulf Condition Report CR-GGN-2001-1899 dated December 4, 2001.

Grand Gulf LER 50-416/01-004, Revision 1, “Violation of Operating License Condition 2.C(1) Maximum Power Level, dated December 20, 2001.

American Society of Mechanical Engineers (ASME) Steam Tables, 3rd Edition.

EC 49520: Safety Relief Valves Tailpipe Thickness

1R16 Operator Workarounds

Operator Workaround List dated November 13, 2001.

1R17 Permanent Plant Modifications

DCP 99000187 Removal of Safety Relief Valve (SRV) Test Instrumentation and Cables.

10 CFR 50.59 Evaluation L99-1335 Removal of SRV Test Instrumentation and Cables.

Calculation 032893 Subsystem 1MS-73 Piping and Valve Removal.

1R19 Post-Maintenance Testing

WR 99007715 High Pressure Core Spray Pump Post Maintenance Operability Run.

LOS-HP-Q1 High Pressure Core Spray (HPCS) System Inservice Test.

WR 00344975 1B Residual Heat Removal (RHR) Service Water Operability.

LOS-RHR-Q1 Low Pressure Core Injection (LPCI) and RHR Service Water Pump and Valve Inservice Test for Modes 1, 2, 3, 4, and 5.

WR 99118105-02 1E51-F066 RCIC Inboard Check Valve Operability

1R20 Refueling and Outage Activities

LaSalle County Station L1R09 Refuel Outage - Revision 0, dated August 24, 2001.

LaSalle L1R09 2-Month Readiness Review

LaSalle Station Outage Operating Experience - L1R09 Edition

Exempt Change 0000334581, “Technical Requirement Manual 3.7.1 72-Hour Provision Acceptability Evaluation,” dated December 31, 2001.

Memorandum From M. Jordan, Senior Resident Inspector, LaSalle Nuclear Station, to G. Wright, Chief, Reactor Projects Branch 2C, “Technical Specifications on System Operability For Snubber Testing,” dated February 28, 1996.

Memorandum From H. Denton, Director, Office of Nuclear Reactor Regulation, to C. Norelius, Director, Division of Reactor Projects, Region III, “Technical Specification Interpretation on Snubbers,” dated May 27, 1986.

LaSalle General Procedure (LGP) 2-1, “Normal Unit Shutdown,” Revision 59, dated April 24, 2001.

Calculation BSA-L-01-02,	“Alternate Decay Heat Removal (ADHR) System Qualification For L1R09 Refueling Outage,” Revision 0.
LES-PC-113A,	“Unit 1 Group 4 Outboard Isolation Logic System Functional Test,” Revision 3.
Design Analysis BSA-L-01-002	Alternate Decay Heat Removal (ADHR) System Qualification for L1R09 Refueling Outage, Revision 0.
LOP-DW-01	Drywell Closeout (After Outage), Revision 32, dated October 1, 2001.

<u>1R22 Surveillance Testing</u>

LOS-DG-M2, Attachment 2A	2A Emergency Diesel Generator Operability Test.
Work Request 00378515	2A Emergency Diesel Generator (EDG) Governor Repair.
CR 00082092	2A EDG Governor Failed to Respond During Monthly Run; November 7, 2001.
CR L2001-05819	Speed Adjustment Knob on 2A EDG Hard to Turn; November 7, 2001.
LTS-800-103	“1B Emergency Diesel Generator (EDG) IE22-S001 Start and Load Acceptance Test”
LTS-100-21	Primary Containment Chilled Water Isolation Valves Local Leak Rate Tests.
LTS-600-41	Primary Containment Inspections for ECCS Suction Strainer Debris Sources.

Unit 1 Suppression Pool As-Found Underwater Inspection Videotape.

Underwater Construction Corporation (UCC) Report - L1R09 Suppression Pool Inspection, dated January 18, 2002.

LOS-RH-Q1, Attachment 2B	2B Residual Heat Removal System Operability and Inservice Test.

LOS-RI-R3, Attachment 1A	Reactor Core Isolation Cooling System Pump Operability Test.
Offsite Review 82-18, Supplement 1	Surveillance Change to RCIC Pump Operability Test.
Onsite Review 82-27	Change to RCIC 18-Month Technical Specification Surveillance Requirement 4.7.3.c.2 to Account for Flow Differences Between the Test Flow Path and the Normal Flow Path.
Onsite Review 82-31	LOS-RI-R3 - 150# Operability Test Required Flow
Engineering Change 335171	Evaluation of SRV Test Instruments/Cables in the Unit 1 Suppression Pool
Engineering Change 335179	Evaluation of SRV Test Instruments/Cables in the Unit 1 Suppression Pool

1R23 Temporary Plant Modifications

Temporary Modification 334065	Shutdown Range Vessel Level Indication.
Work Order 388494-01	Shutdown Range Vessel Level Indication.

2OS1 Access Controls For Radiologically Significant Areas

CR # 90153 and Prompt Investigation Report (Draft) “C” LPCI Locked High Radiation Area Event January 16, 2002

RP-AA-460 Controls for High and Very High Radiation Areas Revision 1

Technical Specification 5.7 High Radiation Area Amendment 147/133

Drywell Survey Data @ 777' and 796' Elevations January 10 and 11, 2002, and Historical Outage Data

2OS2 ALARA Planning and Controls

L1R09 RWP	Dose Reports, Dose Trending Data and ALARA Dose Estimates January 15 - 18, 2002

Listing of Outage Generated CRs Coded to RP Issues January 10 - 17, 2002

RP-AA-401	Operational ALARA Planning and Controls Revision 1
RP AA-400	ALARA Program Revision 1
RWP # 01010837 and Associated ALARA Plan	SRV Work and Associated Activities for L1R09Revision 2

RWP # 01010852 and Associated ALARA Plan	SRM/IRM Detector Replacement Revision 2
RWP # 01010853 and Associated ALARA Plan	Control Rod Drive Replacements Revision1
RWP # 01010855 and Associated ALARA Plan	Reactor Vessel and Nozzle ISI and Support Activities Revision 1
RWP # 01010856 and Associated ALARA Plan	Drywell ISI and Support Activities Revision1
RWP # 01010857 and Associated ALARA Plan	Low Power Range Monitor Replacements and Testing Revision 1
RWP # 01010858 and Associated ALARA Plan	Repair of the 1B33-F060A Valve Revision 2
RWP # 01010859 and Associated ALARA Plan	Disassemble and Reassemble Reactor Vessel, Fuel Moves and Cavity Dryer Separator Pit Decon Revision 0
RWP # 10000630 and Associated ALARA Plan	Install Permanent Lead Shielding in the DrywellRevision 1
RWP # 01010842 and Associated ALARA Plan	Drywell Scaffolding Installation/Removal Revision 1

2001 and L1R09 Area-Based Source Term Reduction Project Matrix Undated

Source Term Reduction Subcommittee Action Plan Summary December 2001

Feedwater Iron and Zinc and Reactor Water Cobalt/Zinc Concentration Data November 1999 - December 2001

RP-AA-401	Attachment 7 ALARA Work In Progress Review for Drywell Scaffolding January 11, 2002
RP-AA-401	Attachment 7 ALARA Work In Progress Review for SRV Replacement January 15, 2002
TEDE ALARA Evaluations	For RWP # 01010853, 01010852 and 01010854 CRD Replacement, SRM/IRM & Connector Work and Under Vessel Sump Activities January 11, 2000 and January 12, 2002
CR # 00090141	Reactor Flood Up Specification - Shutdown Chemistry January 11, 2002

CR #s 89574, 89588, 89985, 90247, 90324, 90302, 90381, 90491, 90493, 90413, 90155, 90981, 91001, 90274, 90965, 90711, 90983, and 90986 Radworker Performance Issues January 1 - 17, 2002

CR # 00090534 L1R09 Scaffold RWP Exceeds Estimate January 12, 2002

CR # 00090319 Higher Than Anticipated Drywell Dose Rates January 12, 2002

Nuclear Oversight Assessment NOA-LS-01-4Q Plant Support Functional Area Assessment Agenda Plan October 2001

Focus Area Self-Assessment Report # 2001-036 Outage Readiness and Preparation December 3 - 5, 2001

CR # 91224 Prompt Investigation of Multiple Uptakes During 1B33-FO60A Valve Parts Decontamination and Potential Exposure Evaluation January 25, 2002

RP-AA-441 Evaluation and Selection Process for Respirator Use Revision 1

Performance Indicator Verification

Performance Indicator Data Sheets - RHR System, January 2000 through December 2001.

Performance Indicator Data Sheets - Unplanned Power Changes, January 2000 through December 2001.

Plant Status

Engineering Change 335129 Provide Engineering Recommendations Regarding the Increased Vibration Levels on the Alterex Brushes Observed During LES-GM-126 on Unit 2 (January 2002).

WO 403866-01 Unit 2 Exciter Inboard Collector Ring Cleaning.

Engineering Change 335333 Justification to Ramp LaSalle Unit 1 to Full Power.