April 23, 2001

Mr. Oliver D. Kingsley, President
Exelon Nuclear
Exelon Generation Company, LLC
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: LASALLE COUNTY STATION
NRC INSPECTION REPORT 50-373/01-03(DRP); 50-374/01-03(DRP)

Dear Mr. Kingsley:

On March 31, 2001, 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 March 30, 2001, with Mr. C. Pardee 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.

On March 31, 2001, the local International Brotherhood of Electrical Workers (IBEW) union contract with ComEd expired. Because negotiations between the union and Exelon (ComEd) management indicated that an agreement was not likely prior to expiration of the contract, the NRC conducted an inspection to evaluate the licensee's strike contingency plans. This inspection, conducted prior to the expiration of the contract at LaSalle, verified that the licensee's plans met all of the requirements of the Technical Specifications and Federal Regulations in the event that a strike were to occur.

Based on the results of this inspection, the inspectors identified one issue of very low safety significance (No Color) that was determined to involve a violation of NRC requirements. However, because of its very low safety significance and because it was entered into your corrective action program, the NRC is treating this issue as a Non-Cited Violation in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny this Non-Cited Violation, 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.

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Sincerely,

/RA/

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

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

Enclosure: Inspection Report 50-373/01-03(DRP); 50-374/01-03(DRP)

cc w/encl:
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REGION III

Docket Nos:	50-373, 50-374
License Nos:	NPF-11, NPF-18
Report No:	50-373/01-03(DRP); 50-374/01-03(DRP)
Licensee:	Exelon Generation Company
Facility:	LaSalle County Station, Units 1 and 2
Location:	2601 N. 21st Road Marseilles, IL 61341
Dates:	February 11 - March 31, 2001
Inspectors:	E. Duncan, Senior Resident Inspector G. Wilson, Resident Inspector G. Pirtle, Senior Security Specialist J. Yesinowski, Illinois Department of Nuclear Safety
Approved by:	Bruce L. Burgess, Chief Projects Branch 2 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000373-01-03; IR 05000374-01-03; on 02/11-03/31/2001; Exelon; LaSalle County Station, Units 1 & 2; Temporary Plant Modifications.

The inspection was conducted by the resident inspectors. One “No Color” finding was identified which was the subject of a Non-Cited Violation. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using 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.

Inspector Identified Findings

Cornerstones: Barrier Integrity

- No Color. The inspectors identified a Non-Cited Violation for the failure to identify that the solenoid valve associated with primary containment isolation valve 1RF012 had an air leak which could have rendered the system for measuring unidentified leakage inoperable. The finding was of very low safety significance since other diverse means were available to identify an increase in reactor coolant system leakage. (Section 1R23)

Report Details

Summary of Plant Status: Both units operated at or near full power for the entire inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R05 Fire Protection

a. Inspection Scope

The inspectors walked down the following risk significant areas looking for any fire protection degradations:

- Fire Zone 3H3: Unit 2 Residual Heat Removal (RHR) Heat Exchanger “B” Cubicle - Elevation 694'
- Fire Zone 2H1: Unit 1- Elevation 694' - Reactor Building General Floor Area
- Fire Zone 3B2: Unit 2 - Elevation 820' - Reactor Building General Floor Area

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 also observed and verified to be 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.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors reviewed the licensee 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 following systems were evaluated:

- Reactor Protection System (RPS)
- Nuclear Instrumentation

The Nuclear Instrumentation system was selected based upon performance problems and classification as a maintenance rule (a)(1) system. The Reactor Protection System was classified as a maintenance rule (a)(2) system and was chosen based upon its relatively high risk significance.

The inspectors independently verified the licensee's implementation of maintenance rule requirements 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 Prioritization

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 to address increased online risk during these periods, such as establishing compensatory actions, minimizing the duration of the activity, obtaining appropriate management approval, and informing appropriate plant staff, were accomplished when online risk was increased due to maintenance on risk-significant SSCs. The following specific activities were reviewed:

- The inspectors reviewed the maintenance risk assessment for work planned for the week of February 11, 2001. This included work associated with the 1A Emergency Diesel Generator (EDG) and 1A EDG Cooling Water Pump.

- The inspectors reviewed the maintenance risk assessment for work planned for the week of February 18, 2001. This included work associated with the Unit 0 EDG and Unit 0 EDG Cooling Water Pump. During the review, and following maintenance on the system, the Unit 0 EDG Cooling Water Pump motor failed during surveillance testing. As a result, the inspectors also reviewed the maintenance risk associated with this emergent work.

- The inspectors reviewed the maintenance risk assessment for work planned for the week of February 18, 2001. This included emergent work associated with the Unit 1 Reactor Core Isolation Cooling (RCIC) system concurrent with a motor-driven reactor feedwater pump outage.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed selected operability evaluations 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 operability evaluations were reviewed:

- OE 01-007 Revision 0: Reactor Core Isolation Cooling (RCIC) Barometric Condenser Condensate Pump Failure

This operability evaluation reviewed the impact of the failure of the RCIC Barometric Condenser Condensate Pump on RCIC system operability. The inspectors verified that in the event that the condensate pump failed, that the vacuum tank relief valve had an adequate relief setpoint and capacity to ensure that all inputs to the tank would be conducted into the Reactor Building Equipment Drain Tank, and therefore not substantially increase water level in the RCIC exhaust drain pot above normal to adversely impact RCIC system operation.

- OE 01-010 Revision 0: Drywell Floor Drain Fillup Rate Monitor

This operability evaluation reviewed the operability of the Unit 1 Drywell Floor Drain Fillup Rate monitor which had exhibited fluctuations in indicated leakage flow. The inspectors verified that the design and operation of the monitor met the requirements of Technical Specification 3.4.3.1, “Reactor Coolant System Leakage - Leakage Detection System,” and Regulatory Guide 1.45, “Reactor Coolant Boundary Leakage Detection Systems.” In particular, the inspectors verified that the monitor had the capability to collect and measure leakage to the primary containment from unidentified sources with an accuracy of 1 gallon per minute (gpm) or better, and that the sensitivity and response time of the monitor was adequate to detect a leakage rate of 1 gpm in less than 1 hour.

- OE 00-011 Revision 0: Reactor Core Isolation Cooling Spectacle Flange Weld

Modification M01-1-9900138 replaced RCIC Valve 1E51-F064 with a spectacle flange. The weld that attached the upstream spectacle flange to line 1RI01A-10 was within the American Society of Mechanical Engineers (ASME) Class 1 boundary. The ASME weld record for field weld #3 properly assigned a welding procedure qualified for impact testing, Welding Procedure Standard (WPS) 1-1-B1. However, the weld record incorrectly specified a maximum interpass temperature of 700 degrees fahrenheit although the actual maximum interpass temperature permitted by WPS 1-1-B1 was 550 degrees fahrenheit. Therefore, field weld #3 was welded within the heat input range specified by WPS 1-1-B1, but the interpass temperature exceeded the maximum allowable temperature by up to 150 degrees fahrenheit. This operability evaluation reviewed the impact of this error on the structural integrity of the weld and determined whether any adverse consequences of the error were likely. To confirm the conclusions regarding the acceptability of the subject weld, the inspectors reviewed non-destructive testing results as well as special testing which utilized test coupons welded under the same conditions as field weld #3 and then subjected to testing, such as Charpy V-notch, to ensure that all ASME code fracture toughness requirements were met.

b. <u>Findings</u>

No findings of significance were identified.

1R16 <u>Operator Workarounds</u>

.1 <u>Operator Workarounds - Individual Impact Assessment</u>

a. <u>Inspection Scope</u>

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 322: Reactor Water Level Control System Problems

This operator challenge identified that reactor water level control system problems had challenged operators and contributed to unplanned plant transients. The inspectors verified that although these problems have led to at least one initiating event, the function of accident mitigating systems and the ability to implement procedures to respond to the events had not been adversely impacted.

- OC 324: Safety Relief Valve Internal Seat Leakage

This operator challenge identified that multiple Unit 1 and Unit 2 Safety Relief Valves have internal seat leakage requiring operation of the Low Pressure Core Spray (LPCS) system to mix the suppression pool and operation in the suppression pool cooling mode of the Residual Heat Removal (RHR) system to ensure that suppression pool temperature is maintained within design basis limits. The subject of OC 324 has also been reviewed in several operability evaluations (OEs), including OE 00-12, which the inspectors documented their review of in NRC Inspection Report 50-373/01-02; 50-374/01-02. In addition, if an event occurs which requires that these systems inject into the reactor vessel, re-alignment of the RHR and LPCS systems into the injection mode would automatically occur.

- OC 321: Residual Heat Removal Service Water Keepfill System Failures

This operator challenge identified that the Division 2 Residual Heat Removal Service Water keepfill systems were subject to repeated failures which have required compensatory and contingency actions.

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 operator workarounds and operator challenges 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.

The majority of documented OWAs and OCs reviewed had only a negligible potential impact on initiating event frequency, the functional capability of a mitigating system, or the potential to impact human reliability in responding to an event. Operator Challenge 322 regarding reactor water level control problems which have led to a number of plant transients, and OC 324 regarding multiple Unit 1 and Unit 2 safety relief valves which leak excessively were recently identified by licensee personnel and were reviewed for an aggregate impact since in the first case an initiating event frequency impact existed, and in the second case, the reliability of the RHR and LPCS systems could be impacted by the additional required operating time of these systems.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed and observed post-maintenance testing activities involving risk significant equipment. 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. The following work requests (WRs) involving risk significant equipment were reviewed:

- WR 990156949-01 Unit 1 Pressure Switch 1PS-DG047 1A

The inspectors observed the performance of LaSalle Operating Surveillance (LOS) DG-M2 Attachment 1A, "1A Diesel Generator Operability Test - Fast Start," which verified that the 1A Emergency Diesel Generator (EDG) crankcase overpressure switch replacement had been adequately accomplished with no leakage present.

- WR 99023789301 LOS-DG-Q1, Attachment A5, "'0' Diesel Generator Cooling Water Pump ASME [American Society of Mechanical Engineers] Section XI Test"

The inspectors observed the performance of LOS-DG-Q1, Attachment A5, on February 21, 2001, which verified that the Unit 0 Emergency Diesel Generator (EDG) Cooling Water Pump was operable following the replacement of a faulted pump motor. The failure occurred during the performance of LaSalle Technical Surveillance 800-104, "0 Diesel Generator 0DG01K Twenty-Four Hour Run Surveillance," on February 20, 2001. In particular, the inspectors observed the collection of pump and motor vibration data and verified that the data was within established acceptance criteria.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

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 the 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 800-104, “0 Diesel Generator 0DG01K Twenty-Four Hour Run Surveillance”

During the performance of LTS-800-104 on February 20, 2001, the Unit 0 EDG Cooling Water Pump failed unexpectedly. The Unit 0 EDG was shutdown and LTS-800-104 was suspended. The cooling water pump was replaced and the surveillance was re-started on February 21. The inspectors observed portions of the surveillance. In particular, the inspectors verified through a review of strip chart data, that the EDG attained required voltage and frequency within the time frames specified in the surveillance acceptance criteria and Technical Specifications.

- LaSalle Instrument Surveillance RI-101, “Unit 1 RCIC [Reactor Core Isolation Cooling] Steam Line High Flow Isolation Calibration”

On February 27, 2001, the inspectors observed the performance of LIS-RI-101. In particular, the inspectors verified through a review and observation of the maintenance activity that the pressure switch actuation, the alarm actuation, and associated relay actuation were obtained within the pressure bands and the time frames specified in the surveillance acceptance criteria and Technical Specifications. During the surveillance, the inspectors also verified that the RCIC system remained available as defined by the NRC unavailability performance indicators.

b. <u>Findings</u>

No findings of significance were identified.

1R23 <u>Temporary Plant Modifications</u>

a. <u>Inspection Scope</u>

The inspectors reviewed Temporary Modification 9900469 which installed an alternate method of monitoring the Unit 1 drywell floor drain sump input used to measure unidentified leakage. The inspectors reviewed the associated 10 CFR 50.59 safety evaluation against the system design basis documentation, including the Updated Final Safety Analysis Report, Technical Specifications, and Regulatory Guide 1.45, “Reactor Coolant Pressure Boundary Leakage Detection Systems,” and verified that the temporary modification had not adversely impacted leakage detection system operability. The inspectors also conducted a walkdown of the temporary modification and compared the installed configuration against the configuration prescribed in design drawings.

b. Findings

The inspectors identified a Non-Cited Violation for the failure to identify that a solenoid valve had an air leak which could have rendered the system for measuring unidentified leakage inoperable. This “No Color” finding was of very low safety significance.

Background

The drywell floor drain sump is the collection point for all unidentified leakage into the drywell. The rate at which the drywell floor drain sump fills up is monitored as part of the leak detection system since a change in this fillup rate is an indicator of a leak. The allowed unidentified leakage is 5 gallons per minute (gpm). The leak detection instrumentation is designed to detect an unidentified leakage increase of 1 gpm in 1 hour to meet the requirements of Regulatory Guide 1.45, “Reactor Coolant Pressure Boundary Leakage Detection Systems,” committed to in the Technical Specification Bases for Technical Specification 3.4.3.1, “Leak Detection Systems.” This is accomplished by directing unidentified leakage to a collection compartment where it flows through a V-notch weir to the main sump. Measuring the level behind the weir provides a direct indication of flow rate. Problems had been previously encountered with this flow rate measuring instrumentation.

Temporary Modification 9900469 Discussion

The purpose of Temporary Modification 9900469 was to install a redundant means of measuring unidentified leakage to meet the requirements of Technical Specifications 3.4.3.1 and 3.4.3.2 which prescribed required leakage detection systems and operational leakage limits, respectively. This was accomplished through the installation of non-intrusive magnetic flow instrumentation outside of containment. Under the revised configuration, unidentified leakage flow from the drywell floor drain sumps pass through two existing Primary Containment Isolation System (PCIS) valves to newly installed piping and an added flow instrument which bypass the drywell floor drain sump pumps. The two PCIS valves discussed above, which were previously normally closed and only opened during drywell floor drain sump pump operation, were altered to be normally open to allow a continual measurement of unidentified leakage and to allow a redundant means to meet the leakage detection requirements prescribed by Technical Specification 3.4.3.1.

Inspector Review

The inspectors reviewed the associated 10 CFR 50.59 safety evaluation against the system design basis documentation, including the Updated Final Safety Analysis Report, Technical Specifications, and Regulatory Guide 1.45. On March 28, the inspectors conducted a walkdown of the temporary modification and compared the installed configuration against the configuration prescribed in design drawings. During the walkdown, the inspectors identified air leakage from the air solenoid valve associated with 1RF012, the inboard primary containment isolation valve. The inspectors informed the licensee of this material condition issue which was captured under Action Request 990139997. On March 29, due to potential operability concerns, the air solenoid valve was immediately replaced and the system was returned to operation.

Significance Evaluation

The inspectors reviewed this issue against the guidance contained in Appendix B, “Thresholds for Documentation,” of Inspection Manual Chapter (IMC) 0610*, “Power Reactor Inspection Reports.” The inspectors determined that the issue did have a credible impact on safety since the air leak, if uncorrected, could have resulted in the closure of isolation valve 1RF012 which would have rendered the detection of unidentified leakage through sump flow monitoring inoperable and adversely impacted the ability to meet the leakage detection and measurement requirements prescribed in Technical Specifications 3.4.3.1 and 3.4.3.2. In addition, the inspectors concluded since numerous operations, engineering, and other licensee personnel were present during modification installation and post-modification testing, that sufficient opportunity to identify this issue existed prior to discovery by the inspectors. However, since the leak detection system is not a mitigating system, this finding did not require evaluation utilizing the Significance Determination Process. As a result, a “No Color” finding was identified. In addition, since other diverse means to identify containment leakage such as particulate and gaseous radiation monitors, drywell temperature detectors, and drywell pressure instruments existed to identify increased reactor coolant system leakage, the finding was of very low safety significance.

Enforcement Actions

This finding did have a credible impact on safety; however, since other diverse means to identify reactor coolant system leakage was available, the finding was considered to be of very low safety significance (No Color). 10 CFR 50, Appendix B, Criterion XVI, “Corrective Actions,” 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 air leakage from the pilot solenoid valve associated with 1RF012, which under a more degraded condition could have rendered unidentified leakage measurement system 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 the corrective action program (Condition Report L2001-01928), this corrective action violation is being treated as a Non-Cited Violation (NCV 50-373/01003-01).

3. SAFEGUARDS

Cornerstone: Physical Protection (PP)

3PP4 Security Plan Changes

a. Inspection Scope

The inspector reviewed Revisions 64, 65, and 66 of the LaSalle Nuclear Station Security Plans, which were submitted by licensee letters dated January 2, 2001, February 8, 2001, and March 2, 2001, respectively. The review was completed to confirm that the changes did not decrease the effectiveness of the security plan and were submitted in accordance with 10 CFR 50.54(p).

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems

As discussed in Section 1R23, "Temporary Plant Modifications," the inspectors identified a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Actions," for the failure to identify that the solenoid valve associated with primary containment isolation valve 1RF012 had an air leak which could have rendered the system for measuring unidentified leakage inoperable. The inspectors concluded that since numerous operations, engineering, and other licensee personnel were present during temporary modification installation and post-modification testing, sufficient opportunity to identify this issue existed prior to discovery by the inspectors. However, since only the leak detection system was impacted, which was not a mitigating system, this issue did not require evaluation utilizing the Significance Determination Process. As a result, a "No Color" finding was identified.

4OA5 Other

.1 IP 92709 Licensee Strike Contingency Plans

a. Inspection Scope

On March 31, 2001, the local International Brotherhood of Electrical Workers (IBEW) union contract with ComEd expired. Because negotiations between the union and Exelon (ComEd) management indicated that an agreement was not likely prior to expiration of the contract, the NRC conducted an inspection to evaluate the licensee's strike contingency plans. This inspection, conducted prior to the expiration of the contract at LaSalle, verified that the licensee's plans met all of the requirements of the Technical Specifications and Federal Regulations in the event that a strike were to occur.

The inspectors evaluated the licensee's strike contingency plan and verified that all Technical Specifications and Code of Federal Regulation requirements were met. In particular, the inspectors verified that in the unlikely event of a strike, the licensee's strike contingency plan ensured that personnel were sufficient in number and qualifications to maintain the safe operation of the facility, including implementation of the site emergency plan. Specifically, the inspectors verified that in the areas of plant management, operations, maintenance, security, chemistry, radiation protection, surveillance and calibrations, and administrative controls, strike contingency personnel met all qualification requirements.

The inspectors reviewed the licensee's safeguards contingency plan and verified that the equipment and personnel required by the plan were available and sufficient to ensure that reactor operation and facility security would be maintained.

The inspectors verified that support from local agencies if needed was adequate to ensure unimpeded access of strike contingency workers, medical care services, local fire department services, and support goods. Emergency communication equipment and the Emergency Notification System were verified to be available.

A discussion was held between the LaSalle Site Vice-President and the NRC Branch Chief responsible for LaSalle to ensure that in the unlikely event of a strike, remaining LaSalle County Station personnel were prepared to continue the safe operation of the facility.

b. Findings

No findings of significance were identified.

.2 Review of Annual Institute of Nuclear Power Operations (INPO) Evaluation Report

The inspectors reviewed the INPO Evaluation Report issued in February 2001 for the INPO evaluation conducted from April 17 through April 28, 2000. The inspectors verified that the results of the INPO evaluation were generally consistent with those identified during NRC inspections. No issues were identified which could substantially affect nuclear safety.

4OA6 Meetings

Exit Meeting Summary

The inspectors presented the inspection results to Mr. C. Pardee and other members of licensee management on March 31, 2001. 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.

PARTIAL LIST OF PERSONS CONTACTED

Exelon

D. Bost, Site Engineering Manager
D. Enright, Operations Manager
F. Gogliotti, Design Engineering Supervisor
M. Karney, Manager, Nuclear Security, Midwest Regional Operating Group
C. Pardee, Site Vice President
J. Pollock, System Engineering Manager
W. Riffer, Regulatory Assurance Manager
M. Schiavoni, Station Manager
C. Wilson, Station Security Manager

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-373/01003-01	NCV	1RF012 Primary Containment Isolation Valve Leakage

Closed

50-373/01003-01	NCV	1RF012 Primary Containment Isolation Valve Leakage

Discussed

None

LIST OF ACRONYMS USED

ASME	American Society of Mechanical Engineers
DRP	Division of Reactor Projects
EDG	Emergency Diesel Generator
gpm	Gallons Per Minute
IMC	[NRC] Inspection Manual Chapter
INPO	Institute of Nuclear Power Operations
LPCS	Low Pressure Core Spray
NCV	Non-Cited Violation
OC	Operator Challenge
OE	Operability Evaluation
OWA	Operator Workaround
PCIS	Primary Containment Isolation System
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
RPS	Reactor Protection System
SDP	Significance Determination Process
SSC	System, Structure, and/or Component
UFSAR	Updated Final Safety Analysis Report
WPS	Welding Procedure Standard
WR	Work Request