November 28, 2000

Mr. Oliver D. Kingsley
President, Nuclear Generation Group
Exelon Nuclear Generation
ATTN: Regulatory Services
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: LASALLE COUNTY STATION - NRC INSPECTION REPORT 50-373/00-18(DRP); 50-374/00-18(DRP)

Dear Mr. Kingsley:

On November 9, 2000, 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 November 16, 2000, 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.

Based on the results of this inspection, the inspectors identified one issue of very low safety significance 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 20555-001, with copies to the Regional Administrator, Region III; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-001; 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 Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Bruce Burgess, Chief
Reactor Projects Branch 2

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

Enclosure: Inspection Report 50-373/00-18(DRP); 50-374/00-18(DRP)

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REGION III

Docket Nos:	50-373, 50-374
License Nos:	NPF-11, NPF-18
Report Nos:	50-373/00-18 (DRP); 50-374/00-18 (DRP)
Licensee:	Commonwealth Edison Company
Facility:	LaSalle County Station, Units 1 and 2
Location:	2601 N. 21st Road Marseilles, IL 61341
Dates:	October 1 - November 9, 2000
Inspectors:	E. Duncan, Senior Resident Inspector P. Krohn, Resident Inspector
Approved by:	Bruce Burgess, Chief Reactor Projects Branch 2 Division of Reactor Projects

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC-licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety	**Radiation Safety**	**Safeguards**
• Initiating Events • Mitigating Systems • Barrier Integrity • Emergency Preparedness	• Occupational • Public	• Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW, or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE indicators correspond to performance that may result in increased NRC oversight. A YELLOW indicator represents performance that minimally reduces safety margin and requires even more NRC oversight. A RED indicator is indicative of performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.

SUMMARY OF FINDINGS

IR 05000373-00-18, IR 05000374-00-18, on 10/1-11/9/2000, Exelon, LaSalle County Station, Units 1 & 2. Non-Routine Plant Evolutions.

The inspection was conducted by the resident inspectors. The inspection identified one “No Color” finding The significance of issues is indicated by their color (Green, White, Yellow, Red) and was determined by using IMC 0609 “Significance Determination Process” (SDP). 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, Barrier Integrity, and Emergency Preparedness

No Color. The inspectors reviewed a loss of feedwater heaters which occurred during a load drop on October 9 and concluded that due to inadequate preparation, operators were challenged with an unanticipated condition for which they had not been specifically trained. The inspectors identified a Non-Cited Violation for failure to have an adequate procedure for directing operator actions in the event of a loss of a large portion of feedwater heaters and thereby ensuring that the plant was operated within analyzed boundaries. (Section 1R14).

B. Licensee Identified Violations

No violations of significance were identified.

Report Details

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

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather

a. Inspection Scope

The inspectors verified that the design features and licensee procedures protecting systems from the effects of low temperature during the winter season were adequate. In particular, the inspectors focused on emergency diesel generator (EDG) room heating, ice melt circulating water line operation, and auxiliary electrical equipment room (AEER) heating. For these areas, the inspectors reviewed the Updated Final Safety Analysis Report (UFSAR), LaSalle Operating Surveillance (LOS) ZZ-A2, "Preparation for Winter/Summer Operation," Revision 21, and Design Change Package (DCP) 9900405, "Reactivate 0VE03AA and 0VE03AB." In each case, the inspectors verified that the plant was adequately protected for cold weather. The inspectors walked down portions of the EDG and AEER ventilation systems and verified that the systems had been properly aligned for cold weather operation.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

.1 Partial Walkdown

a. Inspection Scope

The inspectors performed a walkdown of accessible portions of the Unit 1 High Pressure Core Spray (HPCS) system to verify system operability during maintenance on the Unit 1 Reactor Core Isolation Cooling (RCIC) system. The inspectors reviewed appropriate documentation to determine the correct system lineup. These documents included plant procedures, such as abnormal and emergency operating procedures, as well as plant drawings. The inspectors verified the correct valve position of all critical valves in the primary system flowpath using the system piping and instrumentation drawings (P&IDs) and the HPCS system mechanical checklist, and verified breaker alignment using the HPCS system electrical checklist.

b. Findings

No findings of significance were identified.

.2 Complete Walkdown

a. Inspection Scope

The inspectors performed a complete walkdown of accessible portions of the Unit 1 and Unit 2 Emergency Diesel Generators (EDGs) to verify system operability. The inspectors verified the correct valve position of all valves in the primary system flowpath using the system piping and instrumentation drawings (P&IDs) and the EDG system mechanical checklists and verified breaker alignment using the EDG system electrical checklists. Instrumentation valve configurations and appropriate meter indications were also observed. Lubrication and cooling of major components were verified by direct observation. Proper installation of hangers and supports was periodically observed during the walkdown, and operational status of support systems was directly verified by observation of various parameters. Control room and EDG room switch positions for the EDG system were observed. The inspectors also evaluated other conditions such as adequacy of housekeeping, the absence of ignition sources, and proper component labeling.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

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

Fire Zone 2I4: Unit 1 - Low Pressure Core Spray (LPCS)/RCIC Pump Cubicle
Fire Zone 3I4: Unit 2 - Low Pressure Core Spray (LPCS)/RCIC Pump Cubicle
Fire Zone 2E: Unit 1 Reactor Building Elevation 761 feet 0 inches
Fire Zone 3E: Unit 2 Reactor Building Elevation 761 feet 0 inches
Fire Zone 3F: Unit 2 Reactor Building Elevation 740 feet 0 inches
Fire Zone 3I2: Unit 2 High Pressure Core Spray (HPCS) Cubicle

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 mitigation 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, that access to the hoses were unobstructed, and the physical condition of the hoses were verified to be satisfactory. 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 were inspected and verified to be properly installed and in good physical condition.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspectors observed licensed reactor operator simulator training on LaSalle General Procedure 1-1, “Reactor Startup,” in anticipation of restart from LaSalle Refueling Outage L2R08. The inspectors also observed classroom training associated with the new fuel loading. The inspectors verified crew performance in the simulator 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 (TS) actions; and operator group dynamics. The inspectors also verified that classroom training adequately discussed expected core response changes with the new fuel loading.

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 systems evaluated were:

- 250 Volt Direct Current (VDC) Distribution System
- Reactor Core Isolation Cooling (RCIC)
- Auxiliary Electrical Equipment Room Ventilation (VE)

The RCIC and VE systems were selected based on performance problems and an (a)(1) maintenance rule classification. The 250 VDC distribution system was classified as (a)(2) and was chosen based on its relatively high risk significance. 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; that equipment failures had been properly categorized as functional failures; 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 October 1, 2000. This included surveillance activities associated with the Unit 1 250 VDC battery which impacted the RCIC system, and maintenance activities associated with the 1A turbine-driven reactor feedwater pump and the 2A condensate pump motor.

- The inspectors reviewed the maintenance risk assessment for work planned for the week of October 15, 2000. This included planned maintenance and surveillance activities associated with the Unit 2 High Pressure Core Spray (HPCS) system, switchyard equipment, and the Station Air Compressor (SAC) system.

- The inspectors reviewed the maintenance risk assessment for work planned for the week of November 5, 2000. This included planned maintenance and surveillance activities associated with the Unit 1 RCIC system and Unit 2 feedwater system.

b. Findings

No findings of significance were identified.

1R14 Nonroutine Plant Evolutions

.1 Observation of October 8, 2000, Unit 1 Planned Load Drop

a. Inspection Scope

On October 9, 2000, the inspectors observed a planned Unit 1 load drop from full power to 18 percent power to perform planned maintenance. The inspectors verified that operator actions were in accordance with those specified in plant procedures and training. In particular, the inspectors verified that operator actions to address a loss of feedwater heaters which occurred when the recirculation pumps were downshifted were in accordance with LaSalle Abnormal Operating Procedure (LOA) HD-101, "Heater Drain System Trouble," Revision 4. The inspectors also reviewed operator logs, plant computer data, and strip chart recorders and verified that plant parameters were within the limits specified in LOA-HD-101 and TS requirements.

b. Findings

In response to the unexpected loss of a large number of feedwater heaters, licensee personnel properly entered LOA-HD-101, which required that feedwater temperature and core power be monitored to ensure that the plant was operated within the limits of Figure 1 of the procedure. Figure 1, "Limiting FW [Feedwater] Temperature vs. Percent Power," of LOA-HD-101 was based on EMF-96-189, "LaSalle Unit 1 Cycle 9 Principal Transient Analysis Parameters," Revision 0, and required that the unit be manually scrammed in the event that the plant was operated below the scram line which defined the limits of the analysis. During the event, licensee personnel monitored feedwater temperature and core thermal power to ensure that the scram limits of Figure 1 of LOA-HD-101 were not exceeded. Power was subsequently reduced to less than 25 percent power, and the licensee exited LOA-HD-101.

The inspectors discussed this event with the Unit Supervisor and was informed that at the worst point in the transient, feedwater temperature was within three degrees fahrenheit (F) of the scram limit. Following that discussion, the inspectors obtained feedwater and core power history data, independently evaluated the transient, and determined that at the most limiting point of the transient, the plant was operated on the scram line curve. The inspectors discussed this issue with operations and senior licensee management since it appeared that operations personnel were not adequately aware of whether the plant was being operated within the limits defined by EMF-96-189. To address this issue, the licensee revised Condition Report (CR) L200005662, which was originally generated to identify the loss of feedwater heaters, to reflect the inspectors' concerns and assigned an Apparent Cause Evaluation (ACE) to the issue.

That evaluation determined that the loss of numerous feedwater heaters concurrent with the reactor recirculation pump downshift was not a scenario that the operators anticipated. The advance procedure review, just-in-time training, and heightened level of awareness (HLA) briefing did not include the potential loss of most of the heaters. As a result, the licensee concluded the HLA and other activities to prepare the operators for the evolution was inadequate. In addition, because the Average Power Range Monitors (APRMs) were not operable without a gain adjustment, the Powerplex computer was used to provide a more accurate indication of power. The use of the Powerplex computer power data was not referenced in LOA-HD-101, and the use of APRMs was only referenced in the discussion section of the procedure. Since power data could only be obtained about every 60 seconds, information to the operators was limited. This problem was compounded by the fact that feedwater temperature data was updated every 5 seconds. As a result, the operators did not draw temperature and Powerplex power data at the same point in time. The difference in time between when feedwater temperature and power data were documented by the shift crew resulted in the operators observing at least a 3 degree margin to the scram line, when in fact the plant was operated on the scram line.

To address this event, the licensee planned to revise LOA-HD-101 to clearly identify the use of APRMs to determine reactor power during a feedwater transient where the gain of the APRMs are maintained within TS limits. The procedure would also include the possibility of using the Powerplex data for core power determination when the APRMs are inoperable. Actions were also planned to review the need for HLA briefings to consider worst case scenarios as a result of a licensee evaluation of this event.

The inspectors reviewed all of the information presented and concluded that due to inadequate preparation for the evolution, operators were challenged with an unanticipated condition for which they had not been specifically trained. In addition, since LOA-HD-101 did not specifically direct actions to take in the event that the APRMs were inoperable due to gain out-of-tolerances, operators were forced to use an alternative source of information, in this case, the Powerplex computer. Operators failed to appreciate the ramifications of the fact that because power data was only available every 60 seconds and feedwater temperature every 5 seconds, their efforts to take data that would ensure that the plant was operated within analyzed boundaries were inadequate. However, the inspectors review of available plant data did not indicate that the plant was operated below the required scram line and was therefore within the limits of EMF-96-189. Based on this review, this event was determined to be of low safety significance.

10 CFR Part 50, Appendix B. Criterion V, “Instructions, Procedures, and Drawings,” requires that activities affecting quality shall be prescribed by procedures of a type appropriate to the circumstances and shall include appropriate qualitative or quantitative acceptance criteria to ensure that important activities have been satisfactorily accomplished. The failure to have an adequate procedure to direct operator actions to address the loss of feedwater heaters as discussed above is an example where the requirements of 10 CFR Part 50, Appendix B, Criterion V, were not met and is considered a violation. 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.

.2 (Closed) Licensee Event Report (LER) 50-374/00-004: Unit 2 RCIC Isolation

On September 1, 2000, while preparing to return the Unit 2 RCIC system to service following maintenance, an unexpected RCIC steam line differential pressure alarm was received and the associated RCIC containment isolation valves closed. The licensee conducted a root cause investigation and determined that residual water in the system due to the system being isolated during maintenance flashed to steam when the warming line bypass valve was opened and the system was placed in standby in accordance with LOP-RI-05, "Preparation for Standby Operation of the Reactor Core Isolation Cooling System," Revision 21, dated November 9, 1999. This resulted in a sensed high steam flow condition and subsequent containment isolation signal to the RCIC containment isolation valves. Although LOP-RI-05 had been implemented numerous times in the past, the particular maintenance activity required repeated cycling of the RCIC steam line isolation valve and resulted in an excessive amount of moisture buildup in the RCIC piping. To address this issue, the licensee planned to revise LOP-RI-05 to not return steam flow instrumentation to service until after all flashing-induced steam flow transients had occurred. Since the emergency core cooling system was not challenged, the containment isolation was not due to an actual steam line break, and control room personnel took appropriate actions to address the isolation, the safety significance of the event was minimal. The inspectors reviewed the LER, noted that the issue is captured in the licensee's corrective action program and determined that no findings of significance were identified. This LER is closed.

.3 <u>(Closed) LER 50-373/00-004</u>: Inadequate Reactor Protection System (RPS) Testing

On July 16, 2000, during the performance of channel functional testing in accordance with LaSalle Operating Surveillance (LOS) RP-Q3, "Main Steam Isolation Valve [MSIV] Scram Functional Test," one of two valve position limit switches for the 2B MSIV failed to reset after the valve was opened. During a review of this condition licensee personnel identified that only one of two limit switches was required to actuate the alarm and the testing method required that only a single alarm be received to satisfy the test. The test was inadequate since both limit switches were not verified to actuate within the specified acceptance criteria. To address this issue, LOS-RP-Q3 was revised to provide instructions which verified that both MSIV limit switches send closed signals to the RPS logic. This surveillance was subsequently re-performed utilizing the revised procedure. All data fell within required acceptance criteria. As a result, the safety significance of this issue was minimal. The inspectors reviewed LOS-RP-Q3, Revision 8, dated July 17, 2000, and verified that the procedure had been revised appropriately to verify that both limit switches associated with an MSIV actuated to satisfy the surveillance requirements. No findings of significance were identified. This LER is closed.

.4 <u>(Closed) LER 50-374/00-003</u>: P-Bypass Setpoint Set Non-Conservatively

On June 24, 2000, during the Unit 2 restart from forced outage L2F29, licensee personnel identified that the P-bypass protective trip functions required to meet TS 3.3.1, "Reactor Protection System Instrumentation," and TS 3.3.4.2, "End-of-Cycle [EOC] Recirculation Pump Trip [RPT] System Instrumentation," had not enabled at 23 percent power as expected. The licensee conducted a root cause investigation and determined that the corrective actions identified in December 1999 to address the inappropriate removal of a feedwater density compensation computer subroutine for core power which occurred in July 1988 (Unit 1) and February 1989 (Unit 2) did not consider the impact on P-bypass. As a result, the licensee could not confirm that the P-bypass interlock would enable the reactor scram and EOC-RPT from turbine stop valve and turbine control valve closure at 25 percent power. As part of their immediate corrective actions, licensee personnel removed fuses associated with the trip bypass logic on Unit 1 to enable the reactor scram and EOC-RPT to function at all power levels. Since Unit 2 was below 25 percent power, no immediate action was required. Subsequently, design changes were implemented on Unit 1 and Unit 2 to lower P-bypass setpoints to ensure that protective functions were enabled above 25 percent power. The inspectors reviewed this event and determined that since the margin to thermal limits was relatively large at power levels less than 30 percent, and the impact of the feedwater density compensation on the P-bypass reset value was not large, that the safety significance of the issue was minimal. No findings of significance were identified. This LER is closed.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed selected operability evaluations (OEs) of degraded and non-conforming conditions affecting mitigating systems and barrier integrity to ensure that operability was properly justified. The inspectors also reviewed selected calculations associated with emergency diesel generator fuel oil consumption rates to determine if emergency alternating current (AC) power availability was properly supported and no unrecognized increase in risk had occurred. The following operability evaluations and calculations were reviewed:

- OE 94002 Failure of Main Steam Isolation Valve (MSIV) Limit Switches to Reset

  This operability evaluation reviewed the impact of a number of MSIV limit switch reset failures which occurred over a relatively short period of time. The inspectors verified that the problems observed during this time frame were not related to a recent Unit 1 MSIV limit switch reset failure.

- OE 91001 Loss of RCIC System Barometric Condenser Vacuum Pump

  This operability evaluation reviewed the impact of the loss of the RCIC barometric condenser vacuum pump on system operability. The inspectors verified that the function of the pump was to limit internal plant contamination which did not impact the ability to inject water into the reactor vessel.

- Emergency Diesel Generator [EDG] Diesel Fuel Storage and Day Tank Capacity Calculations D0-7, “Diesel Oil Storage Capacity;” DO-11, “Diesel Oil Storage and Day Tank Available Capacity - HPCS Diesel;” and DO-14, “HPCS Diesel Fuel Storage Capacity Margin.”

  These calculations determined the required size of the diesel fuel oil storage and day tanks based on 7-day mission times, design basis load profiles, and assumed diesel fuel oil consumption rates. The inspectors reviewed these calculations and compared them against design basis fuel oil consumption rates and assumptions provided in the UFSAR. The inspectors also obtained fuel oil consumption rate values from the EDG vendor and verified that consumption rates assumed by the licensee were appropriate.

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds

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 306: Offgas Ventilation Dampers Require Manual Assistance To Operate During System Startup and Shutdown

  The ‘A’ and ‘B’ offgas ventilation train dampers do not open and close properly and need manual assistance. Supply and exhaust fans have start permissives that require the dampers to be full open to start. When swapping trains, the damper problems contribute to discharge pressure permissive problems that also prevent the fans from starting.

  The inspectors verified that problem identification forms (PIFs) had been written to document the operational problems associated with the dampers and that appropriate action requests (ARs), engineering requests (ERs), and work requests (WRs) had been generated to address the problems. The inspectors confirmed that actions to repair the dampers were planned. The inspectors verified that no impact existed relative to the ability to remove noncondensible gases from the condenser, condenser vacuum, or the treatment of radionuclides. The inspectors also verified that slight pressure fluctuations in the offgas building caused by damper problems while swapping trains would not result in any automatic offgas train actions or protective features.

- OC 255: Reactor Recirculation System Suction and Discharge Valve Galling

  The Unit 2 reactor recirculation system suction and discharge isolation valves, 2B33-F023A/B and 2B33-F067A/B, were identified to have experienced galling of the disc to disc guide, which could potentially impact their ability to be operated electrically. These valves were subsequently modified to resolve the galling concern and allow remote electrical operation. However, 2B33-F067A, the “A” recirculation loop discharge valve, required manual backseating after being electrically opened to move the valve disc completely out of the flow path. The inspectors verified that the generation of a LaSalle Abnormal Operating Procedure (LOAs) or a LaSalle Emergency Operating Procedure (LGPs) was not required for operation of this valve. This valve was scheduled to be repaired during LaSalle Refueling Outage L2R09.

- OC 122: SBLC [Standby Liquid Control] Heat Trace Alarms at 85 Degrees

  Local reactor building area ambient temperatures have caused the high heat trace temperature alarm to actuate for the SBLC pump suction piping. Operations personnel have treated the alarm as a heat trace failure with the heater in the "ON" position. The inspectors reviewed the SBLC pump suction high temperature alarm frequency, required operator actions and procedural guidance, and the impact of high SBLC pump suction temperatures on the associated design bases.

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:

- Work Request (WR) 990162439-01, "Disassemble, Clean, and Inspect 2C RHR [Residual Heat Removal] Seal Cooler"

- WR 990117040, "1E22-S001 Reset Mechanical High and Low Speed Stops on 1B DG [Emergency Diesel Generator] Governor"

- WR 990065306, "RHRSW [Residual Heat Removal Service Water] Effluent Monitor Loop 'A' Sample Pump, Replace Centrifugal Pump," and LaSalle Special Test LST-2000-018, "Modification Test for the 2A RHRSW PRM [Process Radiation Monitor], DCP [Design Change Package] 9900332," Revision 0

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.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors observed surveillance testing on risk-significant equipment and verified that the systems and components selected were capable of performing their intended safety function and that the surveillance tests satisfied the requirements contained in TSs, the 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 (LTS) 700-5, "Unit 1(2) 250V [Volt] Battery Surveillance Test Discharge," Revision 15
- LOS-RH-Q1, "RHR (LPCI) [Low Pressure Coolant Injection] and RHR Service Water Pump and Valve Inservice Test For Operational Conditions 1,2,3,4 and 5, Attachment 2C," Revision 47
- LOS-DG-M1, "0 Diesel Generator Operability Test - Idle Start," Revision 42

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed Temporary Modification 9900397, "Temporary Substitution of LS-1 Limit Switch for Unit 1 'B' Inboard MSIV [Main Steam Isolation Valve] RPS [Reactor Protection System] B2 Scram Channel." The purpose of this temporary modification was to utilize Unit 1 'B' inboard MSIV limit switch LS-1 in place of limit switch LS-2 which had failed to reset following surveillance testing. The inspectors verified that limit switch LS-1 had characteristics, such as actuation setpoints, that met the requirements of TSs. The inspectors also verified that post-installation testing was adequate to confirm that there was no unintended impact on the plant.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

Cornerstone: Mitigating Systems, Occupational Exposure Control Effectiveness

4OA1 Performance Indicator Verification

.1 Emergency Alternating Current (AC) Power System Unavailability

a. Inspection Scope

The inspectors reviewed reported 2nd quarter 2000 data for the Unit 1 and Unit 2 Emergency AC Power System Unavailability performance indicator. The inspectors utilized the performance indicator definitions and guidance contained in Nuclear Energy Institute (NEI) 99-02, "Regulatory Assessment Indicator Guideline," Revision 0.

The inspectors reviewed Licensee Event Reports (LERs), operator log entries, maintenance rule functional failures, and out-of-service logs for periods of emergency AC power system unavailability. The inspectors verified that planned and unplanned unavailability hours were characterized correctly in determining performance indicator results. The inspectors also verified performance indicator data through independent calculations.

b. Findings

No findings of significance were identified.

.2 Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspectors reviewed TS high radiation occurrences, very high radiation area occurrences, and unintended exposure occurrences for LaSalle Unit 1 and Unit 2 over the last five quarters. The inspectors utilized the performance indicator definitions and guidance contained in NEI 99-02. During plant status walkdowns, the inspectors checked the condition of 41 locked high radiation area boundary doors which represented approximately 50 percent of the total locked high radiation area doors at LaSalle. The inspectors also reviewed corrective action program records, exposure control database records for individuals receiving greater than 100 millirem (mrem) of exposure in a single day, radiation work permits, and selected accelerated investigation reports to verify that no unplanned exposures occurred. The inspectors also reviewed and verified the classification of two occupational exposure control performance indicator occurrences that were reported for the fourth quarter of 1999.

b. Findings

No findings of significance were identified.

4OA6 Meetings

Exit Meeting Summary

The inspectors presented the inspection results to Mr. C. Pardee and other members of licensee management on November 16, 2000. 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

ComEd

K. Bartes, Nuclear Oversight Manager
D. Bost, Site Engineering Manager
R. Gilbert, Operations Manager
J. Henry, Shift Operations Superintendent
G. Kaegi, Site Training Manager
J. Meister, Station Manager
C. Pardee, Site Vice President
W. Riffer, Regulatory Assurance Manager
S. Taylor, Radiation Protection Manager

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-373/00-018-01	NCV	Operator Response to Loss of Feedwater Heaters

Closed

50-373/00-018-01	NCV	Operator Response to Loss of Feedwater Heaters
50-374/00-004	LER	Unit 2 RCIC Isolation
50-373/00-004	LER	Inadequate RPS Testing
50-374/00-003	LER	P-Bypass Setpoint Set Non-Conservatively

Discussed

None