September 1, 2000

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

SUBJECT: LASALLE COUNTY STATION - NRC INSPECTION REPORT 50-373-00-12(DRP); 50-374-00-12(DRP)

Dear Mr. Kingsley:

On August 4, 2000, the NRC completed the baseline problem identification and resolution inspection of your LaSalle County Station Nuclear Generating Plant, Units 1 and 2. The results of this inspection were discussed with Mr. C. Pardee and other members of your staff.

The inspection was an examination of activities conducted under your license as they relate to identification and resolution of problems and compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of a selected examination of procedures and representative records, observation of activities, and interviews with personnel.

Based on the results of the inspection, the NRC concluded that, in general, the corrective action program was fully functional and typically identified and corrected problems. We also concluded that the station had effectively established an environment in which personnel freely identified conditions adverse to quality and entered the deficiencies into the station's corrective action program. In some cases, however, weaknesses were noted in both the problem identification and problem resolution areas. The weaknesses identified during this inspection did not result in any risk significant consequences.

Two violations of NRC requirements were identified. The first violation involved incomplete American Society of Mechanical Engineers (ASME) Code required work package quality reviews and failure to prevent recurrence of the same issue during the subsequent refueling outage. This violation is being treated as a Non-Cited Violation (NCV) because it did not affect a cornerstone (No color) and was entered into your corrective action program. The second violation involved the failure to adequately resolve or evaluate the replacement of air intake filters associated with the 2B emergency diesel generator ventilation system. This violation is being treated as an NCV because it was determined to have very low safety significance (Green) and was entered into your corrective action program. The violations are described in the enclosed inspection report. If you contest the NCVs, you should provide a response within

30 days of the date of this inspection report, with the basis for your denial, 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 20555-001; and the NRC Resident Inspector at the LaSalle County Station facility.

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/

Christine A. Lipa, Acting Chief
Reactor Projects Branch 2

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

Enclosure: Inspection Report 50-373-00-12(DRP);
50-374-00-12(DRP)

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-373, 50-374
License Nos:	NPF-11, NPF-18
Report Nos:	50-373-00-12(DRP); 50-374-00-12(DRP)
Licensee:	Commonwealth Edison Company
Facility:	LaSalle County Station, Units 1 and 2
Location:	2601 N. 21st Road Marseilles, IL 61341
Dates:	July 24 through August 4, 2000
Inspectors:	K. Riemer, Team Leader E. Duncan, Senior Resident Inspector K. Green-Bates, Reactor Inspector
Approved by:	Christine A. Lipa, Acting 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

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- 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 corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates 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 05000373-00-12, IR 05000374-00-12, on 7/24-8/4/00; Commonwealth Edison; LaSalle County Station Nuclear Plant; Units 1 and 2; Identification and Resolution of Problems.

The report covers a 2-week inspection by two region-based inspectors and one senior resident inspector. This was an announced inspection to review the effectiveness of the corrective action process which included the methods used for identification, cause investigation and correction of quality related problems. The inspectors used Inspection Procedure IP 71152, "Identification and Resolution of Problems," to conduct the inspection. The inspection identified two green issues which were considered Non-Cited Violations (NCVs). The significance of issues is indicated by their color (green, white, yellow, red) and was determined by the Significance Determination Process.

Problem Identification and Resolution

The corrective action program was fully functional and typically identified and corrected conditions adverse to quality. In general, station personnel effectively identified and entered problems as problem identification forms (PIFs) into the corrective action program. The significance threshold for entering issues into the program appeared appropriate. However, over the past year some weaknesses were identified at LaSalle County Station with both the identification and effective resolution of problems. The inspectors noted examples where station personnel failed to capture specific items into the corrective action program. Additionally, the inspectors noted some cases where repetitive items suggested that the station's initial resolution of issues was not fully effective. Although none of these items was considered safety significant, and thousands of other items were satisfactorily opened and closed in that time frame, these items represented weaknesses in the licensee's program.

Cornerstone: Barrier Integrity

NO COLOR. The inspectors identified several failures to implement the corrective action program when Unit 1 and Unit 2 ASME Code Replacement and Repair Program requirements for Class 1 and 2 maintenance work quality reviews were not met. On several occasions the licensee did not enter into the plant's corrective action program 19 maintenance work packages that did not meet all 10 CFR 50.55a ASME Code or program procedure requirements. In each case, corrective actions were not taken to correct the situation. Over the past year, the licensee had identified technical Code errors in several Class 2 work packages. This reinforced the importance of the quality review process. The inspectors were concerned that since the problem had occurred on multiple occasions during both of the last 2 outages, that if left uncorrected, the issue could become a more safety significant concern. Failure to promptly identify and correct the failure to meet ASME Code quality requirements for Class 1 and Class 2 repair and replacement work was considered a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XVI. The safety significance of this issue was considered very low based on the absence of adverse consequences and the fact that no technical problems were identified at the time of the inspection. Since the issue does not immediately affect a cornerstone, the finding has no color. (4OA2.1)

Cornerstone: Mitigating Systems

GREEN. Engineering personnel failed to adequately evaluate the replacement of air intake filters associated with the 2B emergency diesel generator (EDG) prior to their installation. After high differential pressure alarms were received during surveillance testing, the licensee did not adequately resolve the issue. As a result, a review of the impact of the design change on the ability of the emergency diesel generator ventilation system to fulfill its safety function was not completed until after the inspectors identified the issue. Since the operability of the EDG was not adversely impacted, this issue was screened as having very low risk significance following a Phase 1 Significance Determination Process review. One Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XVI was identified. (4OA2.2)

Report Details

4. OTHER ACTIVITIES (OA)

4OA2 Identification and Resolution of Problems

.1 Effectiveness of Problem Identification

a. Inspection Scope

The inspectors reviewed NRC inspection reports (including the Plant Issues Matrix and Plant Performance Review letters) and licensee corrective action documents to verify that when issues were identified, they were appropriately characterized and entered into the licensee's problem identification and resolution program. The inspectors also reviewed a sample of items in the maintenance work backlog to determine if timeliness was commensurate with safety, and if there were instances where a combination of low significance issues may collectively result in a more significant concern.

b. Issues and Findings

In general, the team found that station personnel effectively identified and entered problems as problem identification forms (PIFs) into the corrective action program. In most cases the significance threshold for entering issues into the program appeared appropriate.

The inspectors identified no risk significant problems in this area. However, the team noted several examples where station personnel did not identify deficient conditions, or identification of the issues was not timely. Several examples are noted below:

Corrosion on Unit 1 Division 2 Battery

The inspectors identified a case where licensee personnel failed to identify in a PIF that a procedure associated with a risk significant system was not adhered to. The licensee identified via PIF L2000-03719, that during Unit 1, Division 2, 125 VDC battery surveillance testing, a small amount of corrosion was found on Cell 7 and the corrosion was removed. The inspectors observed the performance of the surveillance. The PIF stated that electrical maintenance department (EMD) personnel normally check resistance readings and then clean the battery when corrosion is found. The inspectors reviewed this PIF and identified that the fact that the corrosion was identified and removed was clearly documented. However, the inspectors also identified that LOS-DC-Q2, "Battery Readings for Safety-Related 250 VDC and Division 1, 2, and 3 125 VDC Batteries," Revision 17, dated November 18, 1997, was not adhered to. The procedure required that all surveillance test resistance checks be performed prior to removing corrosion. The inspectors observed that this was not the case in that the battery cell was first cleaned, and then the resistance readings taken. The control room supervisor, independent of the inspectors' observations, also noted that the EMD personnel had cleaned the battery cell without first taking the resistance readings. The licensee evaluated and resolved the issue prior to declaring the system operable.

Therefore, a violation of regulatory requirements did not occur. However, the potential procedure non-compliance was not specifically identified or documented in the PIF discussion. The licensee subsequently closed the PIF with the note that actions were taken to resolve the issue. No information regarding the procedural aspects of the issue was identified.

Class 1 and 2 ASME Code Repair and Replacement Work

The inspectors identified repeated failures to initiate PIFs or take corrective actions when ASME Code Replacement and Repair Program requirements for Class 1 and 2 work package quality reviews were not met. The inspectors noted that both engineering and licensing department personnel were aware in July 1999, that 5 required package reviews following the Unit 2 refuel outage in April 1999, did not meet 10 CFR 50.55a "ASME Code" quality technical review requirements; however, no corrective action was taken to ensure proper reviews were completed within the required time frame. As a result, 14 more examples which included components in risk significant systems occurred following the Unit 1 outage in November 1999. The significance of this issue was that the licensee had previously identified ASME Code errors in several Class 2 work packages when their quality reviews were performed. Therefore, this emphasized the importance of the quality review process. Since the failure to perform the required reviews had occurred on multiple occasions during both of the last two outages, if left uncorrected, the issue could become a more safety significant concern.

Federal regulation 10 CFR 50.55a requires licensee compliance with the ASME Code for operation and maintenance. ASME Code XI IWA-6230 1989 Edition states that repairs and replacements on Class 1 and 2 components shall be included in a summary report to the NRC due 90 days following completion of a refuel outage. Owner Repair and Replacement forms (NIS-2) signed by the Authorized Nuclear Inservice Inspector (ANII) indicating ASME Code review with approval are also required to be submitted with the 90 day summary. The ASME NIS-2 forms were not submitted as required because the ANII had not approved the maintenance work packages. Therefore, in the 19 cases reviewed, the Unit 1 and 2 ASME Code Repair & Replacement Program maintenance packages did not meet the licensee's program procedural requirements or Code requirements because final Engineering, Inservice Inspection Program, Quality Control, and/or ANII ASME Code reviews had not been completed.

At the time of the inspection in August 2000, the licensee had not completed any of the 19 delinquent ASME Code Repair & Replacement maintenance packages, nor had ASME Code Relief Requests been considered or submitted to the NRC for the Class 1 and 2 repair/replacements. Inspectors reviewed maintenance and operational problem identification forms and GE analysis reports for the 19 repair/replacements and concluded that although Code quality reviews had not been performed after work completion, there did not appear to be any technical problems with the work at the time of the inspection. The licensee subsequently completed the 19 quality reviews and although some errors were found, they were not of safety significance and therefore the missed reviews did not result in any safety concerns. However, because of the number of packages not reviewed, the length of time the reviews were delinquent, the lack of detailed information in the 90 Day ISI Summary reports, the failure to follow program procedures, and the possibility that errors could have gone undetected, this was considered greater than a minor violation. If left uncorrected this issue could become a more significant safety concern, however the issue does not immediately affect a cornerstone, and therefore this finding has no color.

10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” requires, in part, that conditions adverse to quality are promptly identified and corrected. Failure to promptly identify and correct the failure to meet ASME Code requirements for Class 1 and 2 repair and replacement work packages on 19 occasions was a violation of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Actions.” This issue is characterized as a **Non-Cited Violation (NCV 50-373/374-2000012-01)** in accordance with Section VI.A.1 of the NRC Enforcement Policy. The licensee generated PIFs L2000-04197, L2000-04098, L2000-04200 and L2000-04320 to address these issues.

.2 Prioritization and Evaluation of Issues

a. Inspection Scope

The team performed an independent assessment of the appropriateness of the assigned significance level (category) for a selected sample of PIFs. The significance level determined the type and timing of the cause evaluation to be performed. Other attributes reviewed by the team included the adequacy of the root cause analysis or apparent cause evaluations, and the corresponding corrective action plans. The inspectors also assessed the licensee’s evaluation of non-cited violations (NCVs), potential generic issues, selected industry experience, and extent of condition reviews.

The team reviewed a sample of items in the maintenance work backlog to determine if timeliness was commensurate with safety, and if there were instances where a combination of low significance issues may have collectively resulted in a more significant concern.

The team also reviewed the methods used by two separate and independent review committees at LaSalle County Station to verify adequacy, control, and compliance with regulatory requirements. These committees were the Event Screening Committee (ESC) and the Corrective Action Review Board (CARB). The review included the controlling procedures, selected records of activities, and attendance at selected group meetings. In addition, the functions, activities, and findings of the two groups were discussed with cognizant licensee personnel, including selected committee members. The inspectors attended meetings for both committees throughout the 2-week inspection.

A listing of specific documents reviewed is attached to the report.

b. Issues and Findings

There were no significant findings identified during this inspection. With the exception of several minor items identified by the team, station personnel appropriately prioritized and evaluated issues. Licensee internal assessments had identified that the site root cause reports were not always meeting management expectations, and that the CARB process was not always an effective barrier for this problem. In many cases the unsatisfactory root cause reports had also been approved by the plant operations review committee (PORC), or had been modified by PORC without notification to the CARB. The team found that corrective actions had been implemented for the CARB to effectively strengthen this barrier.

The inspectors observed one weaknesses in the implementation of the corrective action program's Root Cause Investigation Effectiveness Reviews. Plant staff reports appeared narrowly focused on evaluating only whether the individual "corrective actions to prevent reoccurrence" were closed out and did not form an evaluation on whether the root cause successfully determined and corrected the original safety significant problem such that it would not reoccur. However during the team inspection interval the CARB chairman identified, and initiated actions to address, this weakness.

The inspectors identified several examples of narrowly focused evaluations. The examples are listed below:

High Pressure Core Spray (HPCS) Emergency Diesel Generator (EDG) Air Filter

Brief Overview

The diesel generator ventilation (VD) system provides year-round ventilation of the diesel generator rooms, day tank rooms, and the diesel generator storage tank rooms. In addition, the system removes equipment heat and provides combustion air when the diesel generators are in operation. Each ventilation system for Unit 1 and Unit 2 is designed to limit the maximum temperature to 122 degrees Fahrenheit to conform with the diesel generator equipment rating. As such, the ventilation system is considered a support system for the emergency diesel generator system.

Discussion

The licensee received a high filter differential pressure (d/p) alarm associated with the 2B EDG ventilation filter on June 1, 2000, during routine EDG surveillance testing. PIF L2000-03023 was initiated to document the alarm and discussed that upon inspection, the filters were found to be clean and in good condition. As a result, the PIF requested that the set points for the high filter d/p alarm be evaluated. The inspectors reviewed this PIF after it was closed and determined that the issue was not thoroughly resolved. New filters had been recently installed in the 2B EDG ventilation system, which changed the flow characteristics. In particular, the previously installed filters had a clean pressure drop of 0.17 inches water gauge and the new filters had a clean pressure drop of about 0.31 inches water gauge which was near the high filter d/p alarm set point of 0.40 inches water gauge. The change in pressure drop was previously documented in engineering request (ER) 9906198, but was not evaluated as a change to the plant that could affect safety-related equipment. Through resolution of the PIF, the licensee failed to identify and evaluate several aspects, including, the new filters had a higher pressure drop than the value described in the UFSAR and the new filters were not made of the same material as described in the UFSAR. After the inspectors discussed the findings with the licensee, engineering department personnel initiated PIF L2000-04349 and reviewed this change to the facility in accordance with 10 CFR 50.59, "Changes, Tests, and Experiments." The review determined that an unreviewed safety question did not exist and that the change did not impact system operability.

Significance Determination Process

The inspectors assessed this issue using the NRC's Significance Determination Process (SDP). Since the filter design change did not adversely impact the operability of the emergency diesel generator ventilation system or the EDG, the issue was determined to be of very low safety significance (Green) using the SDP.

Requirements

10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requires that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. Following filter replacement and the subsequent high d/p alarms, the licensee's investigation failed to identify and correct the differences between the UFSAR and the changes made to the EDG ventilation system. This was a violation, however, this Severity Level IV violation is being treated as a **Non-Cited Violation (50-373/374-2000012-02(DRP))**, consistent with Section VI.A.1 of the NRC Enforcement Policy. This item was entered into the licensee's corrective action program as PIF L2000-04349.

2B EDG Missing the Cooling Water Expansion Tank Overflow Line

The licensee documented (PIF L2000-01631) on March 29, 2000, that the 2B DG cooling water expansion tank was missing the overflow drain pipe shown on Figure 6-8 of EMD Operating Manual J-0155, Tab 01. The licensee issued the PIF after the inspectors raised questions concerning the missing overflow drain pipe. When the inspectors originally identified the problem, the system engineer stated that he had been aware of the problem, but had not addressed it through the corrective action process. This piping was provided to allow runoff in the event of overfilling or excessive water expansion during operation. The inspectors identified that the subsequent evaluation regarding the operability of the 2B EDG with the cooling water expansion tank overflow line missing was based solely on engineering judgement and a rigorous evaluation to arrive at this operability conclusion was not performed although the operability of the EDG was potentially impacted. The inspectors concluded that this evaluation was potentially insufficient in that the licensee did not specifically address components potentially impacted by water flow from the line. The licensee generated PIF L2000-04285, "Additional Engineering Insight for PIF L2000-01631," to resolve the potential operability concerns.

Unit 1 Reactor Core Isolation Cooling (RCIC) Injection Line Isolation Valve Body to Bonnet Leak

On February 29, 2000, during the performance of quarterly RCIC system testing, operators identified a leak associated with 1E51-F013, the Unit 1 RCIC injection line isolation valve. The licensee planned to increase the torque on the valve bonnet nuts during the next RCIC maintenance window in May 2000.

On April 10, 2000, the inspectors identified that 1E51-F013 was leaking. The inspectors questioned the prioritization of this work activity since piping downstream of the injection valve could potentially be drained which could result in a hydraulic transient when the system was actuated. Following that discussion, the inspectors determined that although system engineering personnel had identified through calculations that up to about 80 feet of RCIC injection piping was potentially voided, a PIF to document the issue had not been generated, the operability of the RCIC system had not been formally determined, and licensee management had not been informed of the issue. On April 11, the licensee generated PIF L2000-01903 to document the issue and on April 13, the licensee approved Operability Evaluation 00-001 and documented that the system was operable.

Operating Experience (OPEX) Reviews

The inspectors reviewed the licensee’s response to selected industry experience. While the inspectors considered the specific reviews listed below to be narrowly focused, no adverse consequences resulted from the station’s review and disposition of industry operating experience.

<u>OPEX Item 00003943 - NRC IN 99-04 Unplanned Radiation Exposures</u>

The licensee determined that IN 99-04 was not applicable for review since the addressee was all radiography licensees. The inspectors identified that since the licensee performed oversight of contractors who performed radiography, the subject IN was applicable for a review for lessons learned.

<u>OPEX Item 00008200 - NRC IN 99-14 Draindown Events at Quad Cities, ANO-2 and Fitzpatrick</u>

The inspectors identified that certain aspects of the events identified in the NRC IN had not been addressed as part of the licensee’s review. In particular, procedure deficiencies, appropriate use of level instrumentation, and insufficient pre-evolution shift briefings, were all factors which were identified as contributing causes to the events detailed in the NRC IN, but were not addressed in the OPEX review.

<u>OPEX Item 00013144 - NRC IN 99-21 Recent Plant Events Caused by Human Performance Errors</u>

The inspectors identified that the licensee closed this NRC IN to a previously written action plan to address human performance errors that had occurred at

LaSalle. The inspectors determined that the licensee failed to evaluate the specific events discussed in the NRC IN for applicability.

.3 Effectiveness of Corrective Actions

a. Inspection Scope

The team assessed the adequacy of the station's plans to ensure that the corrective actions properly addressed the identified cause(s) of the issue or event. The team also verified the implementation of a sample of corrective actions. The samples were selected based on their importance in reducing operational risks. Lastly, the inspectors assessed a sample of corrective action effectiveness reviews performed by the licensee.

b. Issues and Findings

With the exception of several minor items identified by the team, the station's prioritization and evaluation of issues was generally appropriate and there were no significant findings identified during this inspection. However the team had several observations where the corrective action program appeared narrowly focused or where corrective actions were inadequate for the deficiencies noted in the PIF. The inspectors also observed several instances where the corrective action program did not identify when PIFs were closed inappropriately. The examples are listed below:

Emergency Lighting

The inspectors identified that the licensee closure of PIF L1999-01244 failed to address all documented deficiencies. The subject PIF discussed problems encountered during EDG testing. One of the problems discussed was the failure of emergency lighting in the reactor building that occurred during the testing. The inspectors identified that none of the corrective actions or action tracking items assigned to the PIF closure addressed the emergency lighting issue. An unrelated surveillance performed by the licensee a short time later re-identified, and corrected, the problems associated with the reactor building emergency lighting. Therefore, there were no adverse consequences associated with the licensee's initial failure to address and correct the problems associated with the reactor building emergency lighting.

PIF L2000-03629: 2B RHR-WS Header Pressure Low

The inspectors reviewed the subject PIF which identified that actions were taken in accordance with LOR-1H13-P601-B201 to restore 2B RHRSW pressure following the receipt of a low pressure alarm. The inspectors determined that procedure change requests (PCRs) LOR-2000-0135 and LOR-2000-0136 were generated to add an option to flush through and exercise normal keepfill regulator 1(2)E12-F429, as a possible response to a fouled regulator. The inspectors reviewed these PCRs and determined that PCR LOR-2000-0136 was closed on July 4, 2000, although action to revise the LOR had not yet been accomplished.

PIF L2000-01182: Open Holes on Top of MCC 135Y-2, 255Y-2

The PIF stated that UFSAR Appendix J, Section J.3 stated that motor control centers (MCCs) that are watertight were excluded from the medium energy line break (MELB) analysis. The inspectors identified that this statement is not present in the UFSAR.

The inspectors did not identify any adverse consequences from the noted examples. Therefore, the inspectors considered the deficiencies minor and administrative in nature.

.4 Effectiveness of LaSalle County Station Audits and Assessments

a. Inspection Scope

The inspectors reviewed a sample of self-assessments and Nuclear Oversight audits to evaluate the effectiveness of these activities in assessing performance and identifying problems. The samples included various functional areas within the plant and departmental self-assessments.

b. Issues and Findings

There were no risk significant problems identified in this area. In general, the team observed that the Nuclear Oversight assessments were thorough and contained good findings and recommendations. The scope, depth and quality of departmental self-assessments varied significantly and the team noted that LaSalle was taking actions to improve their self-assessment process.

The team reviewed the station's assessments of the corrective action program and response to the issues identified. The inspectors noted two examples where the station's response to the findings was narrowly focused.

- Internal assessments had found that the site root causes were not always meeting management expectations, and that the CARB process was not always an effective barrier for this problem. In many cases the unsatisfactory root cause reports had also been approved by PORC. Although the plant had experienced some self-revealed indications that past root cause evaluations were not effective, the inspectors identified that the corrective action program did not address going back to fix past root cause evaluations; it only strengthened barriers for the future root cause preparation and review.

- The nuclear safety review board (NSRB) identified that NO was not utilizing the site corrective action program to document and trend deviations found during site assessments. The inspectors identified that the corrective actions for this issue did not address going back to enter trend data points for the last two Quarters of identified plant deviations; it only entered future deviation trend points into the correction program.

Although the inspectors identified no adverse consequences as a result of the licensee's actions in response to internal assessments of the corrective action program, the inspectors considered the licensee's response to the internal assessments to be narrowly focused.

Consistent with the results of the station’s audits and assessments, during the inspectors review of root cause reports, the team noted two examples where root cause reports were not fully effective.

- Improperly Controlled High Radiation Area (NRC NCV)

  On December 6, 1999, operations personnel entered the Unit 1 Reactor Water Cleanup valve aisle and received a high rate electronic dosimeter alarm. Licensee followup (PIF L1999-05963) determined that dose rates on a previously identified hot spot had increased from about 800 millirem per hour (mrem/hr) to about 2200 mrem/hr at 30 centimeters. As a result, the posting requirements had changed to require that access to the area be controlled through a locking device which had not been implemented. Following a review of this event during an Apparent Cause Evaluation (ACE), the licensee was unable to determine the cause for the dose rate increase. As discussed in NRC Inspection Report 50-373/2000-08;50-374/2000-08, a region-based NRC inspector conducting a review of radiation exposure-related performance indicator results, identified that the root cause for the dose increase had been the implementation of noble metals injection which had caused dose rate to increase throughout the plant. The licensee’s effort to identify the root cause for the increase in dose rates was narrowly focused since a region-based NRC inspector identified the root cause during a routine inspection after the licensee’s effort had failed.

- Evaluation of Feedwater Density Correction Did Not Adequately Identify Operational Issues at Low Power (PIF L2000-03491)

  During the Unit 2 startup from a forced outage, the licensee determined that the plant computer heat balance was indicating about 3 percent to 4 percent higher than other plant indications at about 23 percent core thermal power. This was due to a feedwater density correction that was installed in the plant computer after the identification of an overpower issue in December 1999. The PIF also identified that a corrective action to address the operational impact at low power of the added density correction was overlooked by the Root Cause Investigation for the overpower event.

  The corrective actions performed by the licensee fixed the initial problem, but created a second problem that was not identified until the subsequent plant startup occurred. The inspectors concluded that the licensee’s root cause investigation was weak since corrective actions to address the operational impact at low power were not identified.

.5 Assessment of Safety Conscious Work Environment

a. Inspection Scope

During the conduct of interviews, document reviews and observations of LaSalle County Station activities, the inspectors looked for evidence that suggested plant employees may be reluctant to raise safety concerns. The inspectors utilized the type of questions included in Appendix 1 to NRC Inspection Procedure 71152, “Suggested Questions For

Use In Discussions With Licensee Individuals Concerning PI&R Issues," during interviews with licensee personnel. The inspectors also discussed with licensee staff the evaluation and resolution of issues that were addressed by the LaSalle County Station employee concerns program in the past year.

c. Issues and Findings

There were no significant findings during this inspection. The inspectors concluded, based on information collected from interviews licensee personnel, that licensee management fostered an environment in which station personnel felt free to identify and enter safety issues into the corrective action program. However, the inspectors noted the licensee did not always provide feedback to originators of PIFs or procedure change requests concerning how the issue was resolved. The inspectors noted that the licensee had just recently identified a similar issue during a review of the corrective action program prior to the NRC inspection. The inspectors considered this to be a potential discouragement to fostering an environment to raise issues in that plant.

4OA6 Meetings

Exit Meeting Summary

The inspectors presented the inspection results to Mr. C. Pardee and other members of licensee management at the conclusion of the inspection on August 4, 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

C. Pardee, Site Vice President
K. Bartes, Nuclear Oversight Manager
R. Book, Corrective Action Program Manager
D. Bost, Site Engineering Manager
B. Brady, Nuclear Generation Group Regulatory Services
T. Conner, Assistant Design Engineering Supervisor
D. Czufin, Assistant Engineering Director
T. Gierich, Work Control Manager
J. Henry, Shift Operations Superintendent
J. Pollock, System Engineering Manager
M. Schiavini, Maintenance Manager
F. Spangenberg, Regulatory Assurance Manager

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-373/2000012-01; 50-374/2000012-01	NCV	Failure to Identify, Correct, and Prevent Recurrence of Delinquent ASME Code Requirements
50-373/2000012-02; 50-374/2000012-02	NCV	Failure to Identify and Correct Discrepancies Regarding Replacement Air Intake Filters Associated with the 2B EDG

Closed

50-373/2000012-01; 50-374/2000012-01	NCV	Failure to Identify, Correct, and Prevent Recurrence of Delinquent ASME Code Requirements
50-373/2000012-02; 50-374/2000012-02	NCV	Failure to Identify and Correct Discrepancies Regarding Replacement Air Intake Filters Associated with the 2B EDG

Discussed

None

LIST OF DOCUMENTS REVIEWED

The following is a list of licensee documents reviewed during the inspection, including documents prepared by others for the licensee. Inclusion of a document on this list does not imply that NRC inspectors reviewed the entire documents, but, rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort. In addition, inclusion of a document on this list does not imply NRC acceptance of the document, unless specifically stated in the body of the inspection report.

Corrective Action Program Description

CAP-1	"Problem Identification Form Threshold Information Handbook," Revision 3
CAP-2	"Significant Apparent Cause Evaluation (SACE) Handbook," Revision 1
CAP-3	"Root Cause Investigation and Report Handbook," Revision 2
CAP-4	"Trend Investigation and Report Handbook," Revision 0
CAP-6	"Coding and Trending Handbook," Revision 1
CAP-8	"Apparent Cause Evaluation (ACE) Handbook," Revision 1
NSP-AP-1004	"Corrective Action Program Process," Revision 3
NSP-AP-2004	"Corrective Action Program Process Roles & Responsibilities," Revision 3
NSP-CC-3001	"Operability Determination Process," Revision 0
NSP-AP-3004	"Corrective Action Program Handbook," Revision 4
NSP-AP-4004	"Corrective Action Program Procedure," Revision 4

Procedures

EC-AA-101	"Employee Concerns Program," Revision 0
RP-AA-460	"Controls For High and Very High Radiation Areas," Revision 1
LOS-SC-M1	"SBLC [Standby Liquid Control] Pump Operability Test and Explosive Valve Continuity Check," Revision 25
NEP 10-03	Disposition of Design Basis Discrepancies," Revision 0
LOS-DG-M3	"2B DG Idle Start - Attachment 2B," performed 5/3/00
RS-AA-115	"Operating Experience (OPEX)," Revision 1
NSP-WC-3007	"Rework Reduction," Revision 1
RPJS-6.4	"LSCS Radiation Protection Job Standard; Providing Radiological Condition Briefings," Revision 0
RPJS-6.8	"LSCS Radiation Protection Job Standard; RP Coverage For Entry Into High, Locked High, and Very High Radiation Areas," Revision 0
LOS-DC-Q2	"Battery Readings for Safety-Related 250 VDC and Division 1, 2, and 3 125 VDC Batteries," Revision 17
LAP-1300-13	"ASME Section XI Program," 11/19/97
MA-AA-AD-6-03009	"Work Package Close Out Routing," Revision
AD-AA-104-101	"Plant Procedure Process," Revision 0
NEP-17-04	"Nuclear Engineering PSA Model Update Procedure," Revision 0

Problem Identification Forms (PIFs)

L1999-05352 HPCS Full Flow Test Valve Failure
L1999-06034 CY [Cycled Condensate] Tank Hi/Lo Setpoint Tolerance Incorrect
L1999-06174 Concerns During Performance of LOS-SC-M1
L1999-04821 Improper Control of High Radiation Area
L1999-05963 Improperly Controlled High Radiation Area
L1999-05515 1B33-F019 Actuator Damage During LLRT
L1999-03402 DCP Closed Without Proper Setpoint Changes Made to EWCS
L1999-03749 Maintenance Human Performance Common Cause Analysis
L1999-04489 1B Diesel Generator Overspeed Trip
L1999-05978 Process Computer Temperature Correct FW and U2 Millivolt Correction Factors
L1999-06052 Weaknesses in the NEI Performance Indicator Process
L1999-03971 NO Identified 80+ PCRs Not Being Generated As Required During Procedure Revisions
L1999-05828 Powerplex Feedwater Flow Input Problems
L1999-04753 Inadequate Corrective Action Regarding Undersized Fillet Welds
L1999-03402 2 DCPs Closed Without Proper Changes Made to EWCS
L1999-05804 LOS-TG-SR1 Procedure Problems
L1999-02773 Effectiveness Reviews Not Generated Per Root Cause Requests
L1999-01244 Unexpected Loss of Station Lighting During EDG Testing
L1999-05293 Inadequate Design DCP #9800350 - No Access to Allow Test/Calibration of Level Switches
L1999-04821 Improper Control of High Radiation Area
L1999-05963 Improperly Controlled High Radiation Area
L1999-01244 Unexpected Loss of Station Lighting During Emergency Diesel Testing
L1999-02902 ASME Section XI Code Repair/Replacement Activity did not Meet Code
L1999-02906 ASME Section XI Code Repair/Replacement Activity did not Meet Code
L1999-02930 ASME Section XI Code Repair/Replacement Activity did not Meet Code
L1999-03036 ASME Section XI Repair/Replacement Activity w/o LAP Attachment Requirement Completed
L1999-03225 Two RPCU Air Operated Valves Returned to Service With Solenoid Coils Not Installed
L1999-03372 10CFR50.46 LOCA Analysis One Year Reporting Requirement
L1999-03846 NO Identified: No Documented Evidence of PRA Reviews for DCP's
L1999-03971 NO Identified: Procedure Change Requests not being Generated as Required During Procedure Revisions
L1999-04407 LIS-NR-107 As Left Flow Voltages Were Left Out of Tolerance (in the Conservative Direction)
L1999-05515 1B33-F019 Actuator Damage During LLRT
L1999-05352 HPCS Full Flow Test Valve Failure During Div III Response Time Test
L1999-05846 Jet Pump 9 & 10 D/P Out of Specification
L1999-06168 Inadequate Verification and Validation of Regulatory Related Correspondence
L2000-03094 Loss of Division Two Annunciators
L2000-00355 Level Indication Change When Selecting C Level Channel on RWLC
L2000-00400 Transient Due to Feedwater Level Select Switch Deck Assemblies Loose,
L2000-00635 Loose Parts Monitor Failed LIS-LM-301
L2000-00869 Missing Check Valve Parts

L2000-01200	Unit 1 Loose Parts Monitor Channel #10 Inop as of 3/6/00
L2000-01646	NO Identified: Weakness in the Maintenance Verification Process Implementation
L2000-02227	Improvement Needed in Station Self-Assessment Program
L2000-02316	Downers Grove NO Identified: 5 LaSalle Unsatisfactory Root Cause Reports
L2000-02475	B RR HPU Problems
L2000-02508	Shuttle Valve Found Missing Spring and Balls
L2000-02712	NO Identified: 1E51-F013 Pressure Check Not Proceduralized,
L2000-03609	NRC Identified: Concern with RCIC Piping Mod Meeting ASME Code Requirements
L2000-03777	Unit 2 'A' OG Hydrogen Analyzer Indicates Downscale
L2000-03863	ASME Section VIII/ IDNS Relief Valve Design does Not Meet Code
L2000-04077	Valve Repair Not Appropriate
L2000-04105	Crew 4 Post Shift Critique for 7/24/00 Nights
L2000-04107	RP Week in Review, PIF for Week of 7/17/00
L2000-04189	CARB Rejected Effectiveness Review for ATM 16680
L2000-01919	Unable to Locate Vendor Drawing in Central File
L2000-02378	ORAM/Sentinel Program Will Not Meet Operations Needs
L2000-02253	Concerns With Operations Training
L2000-03043	Missing Mounting Hardware on Piping on Three PRM [Process Radiation Monitor] Skids
L2000-03477	Reactor Thermal Power (Thermal Heat Balance) is Indicating Higher Than Expected
L2000-03420	Unit 2 SAC [Station Air Compressor] 2SA01C Surged Following Scram
L2000-03123	1E51-F013 Drilled Disc Modification
L2000-03060	Station Air Dryer Solenoid Leaks
L2000-03723	Potential Discrepancy Between HPCS [High Pressure Core Spray] Performance and the UFSAR [Updated Final Safety Analysis Report]
L2000-03719	Corrosion on Unit 1 Division 2 125 Volt Direct Current Battery
L2000-03629	2B RHR-WS [Residual Heat Removal Service Water] Header Pressure Low
L2000-03258	Untimely Resolution of Operator Workarounds
L2000-03023	HPCS [High Pressure Core Spray] DG [Emergency Diesel Generator] Filter
L2000-03491	Evaluation of Feedwater Density Correction Did Not Adequately Identify Operational Issues at Low Power
L2000-03430	RE/RF [Equipment Drain/Floor Drain] Sump Found in Degraded Condition
L2000-03240	NO [Nuclear Oversight] Identified: CAP [Corrective Action Program] Not in Compliance with NSP-AP-4004
L2000-03501	Excessive Leakage From 2A TDRFP [Turbine-Driven Reactor Feedwater Pump] Inboard Seal While Shutdown
L2000-04285	Additional Engineering Insight for PIF L2000-01631
L2000-04030	Frame Mounted on Top of RBCCW [Reactor Building Closed Cooling Water] Sample Chamber is not Per Design
L2000-04349	Concerns Regarding Configuration Control of Replacement HVAC [Heating, Ventilation, and Air Conditioning] Filters
L2000-04141	NO Identified Inadequate Review of Training Observation Forms
L2000-04040	Out-Of-Date Information in Procedure LMP-HO-G-3 for Piping Strut Installation
L2000-04109	Lack of Authorization Documentation for FAC Program Software
L2000-01778	Missing Support Hardware in 2H13-P603
L2000-02257	Through Bolts on 2A DG Engine Control Switch Loose

L2000-01851 Additional Discrepancies Identified During An Extent of Condition Review for Instrument Fasteners
L2000-01646 NO Identified Weakness in the Maintenance Verification Process Implementation
L2000-04136 Control Room Observation
L2000-01635 Adverse Trend Identified in Operating Manpower
L2000-03846 No Documented Evidence of PRA Reviews for DCPs
L2000-02712 NO Identified 1E51-F013 Pressure Check Not Proceduralized
L2000-02715 Inconsistency Between plant and UFSAR
L2000-00241 Failure of 1A DG to Start During LOS-DG-M2 Idle
L2000-00355 Level Indication Change When Selecting C Level Channel on RWLC
L2000-02134 Op Eval 00-001 Was Inadequate
L2000-03031 U2 Circ Water Biocide Injection Line Plugged

Operating Experience Reports (OPEX)

OPEX Item 00003943 - NRC IN 99-04 Unplanned Radiation Exposures
OPEX Item 00005999 - NRC IN 99-07 Failed Fire Protection
OPEX Item 00007749 - NRC IN 99-13 Insights From Breaker Inspections
OPEX Item 00008200 - NRC IN 99-14 Draindown Events at Quad Cities, ANO [Arkansas Nuclear One]-2 and Fitzpatrick
OPEX Item 00013144 - NRC IN 99-21 Recent Plant Events Caused by Human Performance Errors
OPEX Item 00017294 - NRC IN 99-28 Recall of Star Fire Protection Sprinkler Heads
OPEX Item 00023896 - NRC IN 00-01 Issues Identified in BWR Trips and Transients
OPEX Item 00028495 - GE Service Information Letter (SIL) 625 Rod Block Monitor Selection Failure

Action Tracking Items

14413-17 Effectiveness Review of Corrective Actions for PIF L1999-03749 (7/5/00)
16680-39 Effectiveness Review of Corrective Actions for PIF L1999-04489 (7/20/00)

Root Cause Investigations

Root Cause Unit 2 Reactor Water Level Transients When Selecting ‘C’ Narrow Range Indication, Revision1
Root Cause Technical Specification 4.0.3 entry on Units 1 & 2 Due to Inadequate Surveillance Testing of Relays,” ATM 32083 Revision 0
Effectiveness Review for Root Cause L1999-03749 Maintenance Human Performance Common Cause Analysis,” Revision 0
Effectiveness Review for Root Cause L1999-04489 1B Diesel Generator Overspeed Trip,” Revision 0

Self-Assessments

Site	Engineering Department "ISI and CISI Programs," 7/9/00
LaSalle	Nuclear Oversight Self-Assessment First Quarter 2000, 4/12/00
LaSalle	Nuclear Oversight Self-Assessment 4th Quarter 1999, 1/27/00
LaSalle	Nuclear Oversight Self-Assessment 2nd Quarter 2000, 7/16/00

ATs

AT 2672 "Update NIS-2 Forms for L2R07 design Change Packages," 2/10/99

Misc Documents

ComEd	Calculation No. L-002508, "Components in Top 90 percent of CDF Sorted by Raw and FV," Rev 2
GE	Nuclear Company GE-NE-B13-02047-21-02P DRF B13-02047-21 "Vibration Evaluations of Increased Jet Pump Flow due to Inlet Mixer Replacement," February 2000
NGG	Standard Charter Corrective Review Board (CARB), March 10, 2000
CARB	Meeting Minutes for July 25, 2000
CARB	Meeting Minutes for August 2, 2000
LaSalle	Letter No. 99-127
LaSalle	Letter No. 00-031
LaSalle	Memorandum January 7, 2000 "LaSalle Nuclear Safety Review Board Meeting December 15 and 16, 1999,"
Maintenance	Work Request 980021156 "LPCI Min Flow Line Check Valve (Class 2)," 12/9/98
Maintenance	Work Request 970106677 "HPCS Pump Discharge Check Valve (Class 2)," 91//98
Maintenance	Work Request 990054626 "Check Valve Internals," 1/17/00

LIST OF ACRONYMS USED

ACE	Apparent Cause Evaluation
ANII	Authorized Nuclear Inservice Inspector
ASME	American Society of Mechanical Engineers
CARB	Corrective Action Review Board
DCP	Design Change Package
d/p	differential pressure
DRP	Division of Reactor Projects
EDG	Emergency Diesel Generator
EMD	Electrical Maintenance Department
ER	Engineering Request
ESC	Event Screening Committee
HPCS	High Pressure Core Spray
IN	Information Notice
LER	Licensee Event Report
LLP	LaSalle Special Procedure
LOA	LaSalle Abnormal Operating Procedure
LOS	LaSalle Operating Surveillance
LPCI	Low Pressure Coolant Injection
MCC	Motor Control Center
MELB	Medium Energy Line Break
NCV	Non Cited Violation
NO	Nuclear Oversight
NSRB	Nuclear Safety Review Board
OPEX	Operating Experience
P&ID	Piping and Instrumentation Drawing
PCR	Procedure Change Request
PERR	Public Electronic Reading Room
PIF	Problem Identification Form
PORC	Plant Operations Review Committee
psi	pounds per square inch
QC	Quality Control
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
SDP	Significance Determination Process
UFSAR	Updated Final Safety Analysis Report
VDC	Volts Direct Current
WR	Work Request