April 25, 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 INSPECTION REPORT 50-373/2000003(DRP);
50-374/2000003(DRP)

Dear Mr. Kingsley:

This refers to the inspection conducted from February 12 through April 1, 2000, at the LaSalle Nuclear Power Station. The purpose of the inspection was to determine whether activities authorized by the license were conducted safely and in accordance with NRC requirements. At the conclusion of the inspection, the findings were discussed with those members of your staff identified in the enclosed report.

Overall, station performance during the inspection period was good with no significant issues identified. Feedwater system oscillations and an offgas system leak represented material condition problems which challenged the plant staff. In addition, the inspectors identified material condition deficiencies which had not been identified by your staff despite numerous opportunities during tours by operations personnel, maintenance activities by maintenance personnel, and system walkdowns by engineering personnel.

Also, during the inspection period, your staff identified two welds susceptible to intergranular stress corrosion cracking in a Unit 2 residual heat removal shutdown cooling return line which were not examined as required by Technical Specifications. The NRC granted Notice of Enforcement Discretion 00-6-003 until License Amendment 123 to Facility Operating License NPF-18 was issued on March 22, 2000, which allowed continued operation of Unit 2 until an examination of one of the welds could be performed during the next scheduled refueling outage or December 31, 2000, whichever occurred earlier. Although the missed weld examinations were considered isolated cases with low safety significance, the failure to examine the two welds in accordance with Technical Specification requirements represented a lack of rigor in tracking and performing weld examinations.

Based on the results of this inspection, the NRC has determined that one violation of NRC requirements occurred. This violation is being treated as a Non-Cited Violation (NCV), consistent with Section VII.B.1.a of the Enforcement Policy. The NCV is described in the subject inspection report. If you contest the violation or severity level of the NCV, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the United States Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-001, with copies to the Regional Administrator, Region III, and the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-001.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be placed in the NRC Public Electronic Reading Room (PERR) link at the NRC homepage, http://www.nrc.gov/NRC/ADAMS/index.html.

Sincerely,

/***RA***/ J. Lara for

Kenneth Riemer, Acting Chief
Reactor Projects Branch 2

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

Enclosure: Inspection Report 50-373/2000003(DRP);
50-374/2000003(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:	50-373/2000003(DRP); 50-374/2000003(DRP)
Licensee:	Commonwealth Edison Company
Facility:	LaSalle County Station, Units 1 and 2
Location:	2601 N. 21st Road Marseilles, IL 61341
Dates:	February 12 through April 1, 2000
Inspectors:	E. Duncan, Senior Resident Inspector P. Krohn, Resident Inspector
Approved by:	Kenneth Riemer, Acting Chief Reactor Projects Branch 2 Division of Reactor Projects

EXECUTIVE SUMMARY

LaSalle County Station, Units 1 and 2
NRC Inspection Report 50-373/2000003(DRP); 50-374/2000003(DRP)

This inspection report included aspects of licensee operations, maintenance, engineering and plant support. The report covers a 7-week period of inspection conducted by the resident staff.

Operations

- Material condition deficiencies challenged operators during the inspection period. In addition, the inspectors identified a number of material condition deficiencies during routine plant tours which had not been identified by the plant staff. This represented a weakness in the identification of material condition deficiencies by workers performing activities in the plant. (Section O1.1)

Maintenance

- Overall, the performance of surveillance activities was acceptable and demonstrated that systems were capable of performing their intended safety function. In some cases, however, the inspectors identified examples in which material condition deficiencies associated with safety-related systems had not been identified despite numerous opportunities during tours by operations personnel, maintenance activities by maintenance personnel, and system walkdowns by engineering personnel. (Section M1.1)

Engineering

- Discrepancies between the Updated Final Safety Analysis Report description of installed spray shielding to address Moderate Energy Line Break (MELB) and the installed plant configuration were identified in 1998, but had not been adequately addressed. In addition, the inspectors independently identified material condition deficiencies associated with motor control centers and spray shielding which potentially impacted the ability to mitigate the effects of a MELB. (Section E2.1)

- Due to a clerical error, two welds susceptible to intergranular stress corrosion cracking in a Unit 2 residual heat removal shutdown cooling return line were not examined as required by Technical Specifications. The NRC issued a Notice of Enforcement Discretion and license amendment to allow continued operation of Unit 2 until an examination of one of the welds, RH-2005-29, could be performed during the next scheduled refueling outage or December 31, 2000, whichever occurred earlier. Although the missed weld examinations were considered isolated cases of low safety significance, the failure to examine the welds in accordance with technical specification requirements represented a lack of rigor in tracking and performing weld examinations. (Section E8.1)

Plant Support

- During routine inspections in radiologically controlled areas, the inspectors assessed licensee performance. Overall, the radiation protection staff enforced radiological control standards. No deficiencies in radiation control were identified. (Section R1.1)

Report Details

Summary of Plant Status

Unit 1 and Unit 2 operated at or near full power during the inspection period with the exception of short-duration planned power reductions to support maintenance and surveillance testing activities.

I. Operations

O1 Conduct of Operations

O1.1 General Power Operation Observations

a. Inspection Scope (71707, 92700, 92901)

The inspectors conducted frequent reviews of plant operations at power. These reviews included observations of control room and in-plant evolutions, shift turnovers, and log keeping.

b. Observations and Findings

Control Room Observations

Offgas System Steam Leak: On March 9, during a routine control board walkdown, the Unit 2 nuclear station operator (NSO) noted slight oscillations in the steam jet air ejector (SJAE) steam flow for the 'A' train of the offgas system. Subsequent investigation by the licensee identified a steam leak from a pressure control valve. Operators placed the 'B' train of the offgas system in service, and secured the 'A' train of the offgas system. During the evolution, about 1½ inch of water was discovered on the Unit 2 SJAE room floor due to a clogged floor drain. This represented a material condition deficiency which unnecessarily challenged operators responding to the event. The inspectors reviewed the control room indications available to the NSO who discovered the problem and observed that the operator had demonstrated good attention-to-detail in noting slight changes in plant parameters. The licensee generated Problem Identification Form (PIF) L2000-01238 and PIF L2000-01240 to identify the material condition deficiencies.

Unit 2 Feedwater System Oscillations: Material condition deficiencies associated with the Unit 2 feedwater system occurred during the inspection period. Although operator response to these problems was appropriate and plant conditions were stabilized following operator action, the degrading material condition of the Unit 2 feedwater system presented a challenge to operators. Examples of feedwater system problems which occurred during the inspection period included the following:

- On February 22, the 2B turbine-driven reactor feedwater pump (TDRFP) flow was unstable with the pump in automatic control. Oscillations in turbine speed from 4270 revolutions per minute (rpm) to 4410 rpm were observed. The licensee generated PIF L2000-00897 to enter this issue into the corrective action program.

- On March 12, during a planned power reduction, 2A TDRFP flow oscillations occurred. Operators placed the 2A TDRFP in manual and restored plant conditions to normal. The licensee generated PIF L2000-01274 to enter this issue into the corrective action program.

- On March 19, following routine strainer cleaning, 2B TDRFP speed oscillated about 800 rpm until eventually dampening out to within normal values. The licensee generated PIF L2000-01411 to enter this issue into the corrective action program.

- On March 20, while attempting to isolate the "B" subloop of the 2B TDRFP, speed unexpectedly decreased from 4390 rpm to 4000 rpm. Operators immediately identified the condition, unisolated the subloop, and plant conditions returned to normal. The licensee generated PIF L2000-01421 to enter this issue into the corrective action program.

At the end of the inspection, troubleshooting to identify the root cause of the feedwater oscillations was in progress.

Observations in the Field

Material Condition Deficiencies: The inspectors conducted routine tours of the plant during the inspection period. During these tours, the inspectors identified a number of material condition deficiencies. Items included loose or missing clamps, bolts, and other fasteners in the Diesel Fire Pump room areas; a steel rod which provided structural support for the turbine building wall was missing; and loose conduit clamps in the Unit 2, Division 1, 125 Volt Direct Current (Vdc) battery room. In addition, during a walkdown of the 2B Emergency Diesel Generator (EDG), the inspectors identified that the coolant expansion tank overflow piping was not present although other similar safety-related EDGs had an overflow pipe to conduct fluid outside the EDG skid. The licensee generated PIFs L2000-00993, L2000-00925, and L2000-01405 to enter these issues into the corrective action program.

On March 4, during a walkdown of the Unit 2 reactor core isolation cooling (RCIC) system just prior to planned maintenance on the Unit 2 high pressure core spray (HPCS) system, the inspectors identified numerous support brackets missing from the spray shield associated with piping line 2RI-24A-6, a missing U-bolt nut from an instrument air line, and a loose clamp from an instrument air line. The licensee generated PIF L2000-001133 and Action Requests 990078689, 990078690, 990078691, and 990078692 to identify the deficiencies. The inspectors also determined that although the licensee identified protected equipment pathways prior to maintenance activities, walkdowns of equipment relied upon during planned maintenance on diverse or redundant systems were not routinely practiced or addressed in work control procedures. The inspectors identified a similar issue during a review of licensee plans to utilize a Unit 2 motor-driven reactor feedwater pump isolation valve for throttling flow when the normal feedwater regulating valve was unavailable. During a walkdown of the system, the inspectors identified material condition problems associated with the isolation valve such as oil leaking from the valve and excessive grease on top of the packing gland.

Although none of the issues identified adversely impacted operability, they indicated a weakness in the identification of material condition deficiencies by workers performing activities in the plant.

c. Conclusions

The inspectors concluded that material condition deficiencies challenged operators during the inspection period. In addition, the inspectors identified a number of material condition deficiencies during routine plant tours which had not been identified by the plant staff. This represented a weakness in the identification of material condition deficiencies by workers performing activities in the plant.

O8 Miscellaneous Operations Issues

O8.1 (Closed) Licensee Event Report (LER) 50-374/00001-00: 2B High Pressure Core Spray (HPCS) Emergency Diesel Generator (EDG) Unavailable for Restart Due to Diode Failure.

On February 9, 2000, the 2B EDG start logic control power fuses blew during an unexpected reverse power trip of the EDG. Troubleshooting determined that the failure was the result of a shorted diode. The diode and fuses were replaced and the 2B EDG was returned to service.

Engineering personnel determined that the diode failed prematurely since it had only been installed for 2 years and the design life was 40 years. The cause of the reverse power trip was personnel error since action to open the EDG output breaker following the completion of the surveillance was not timely. The safety significance of the event was minimal since only the emergency power supply to the Unit 2 HPCS EDG was affected and all other accident mitigation systems were operable during the event. The licensee entered this issue into their corrective action program as Action Tracking Matrix (ATM) Item 23595-18.

During a review of the corrective actions for this event, the inspectors determined that a system functional performance review (SFPR) had identified that station procedures which adjusted EDG loading to less than or equal to 200 kilowatts (kW) and 200 kilovolt-amperes-reactive (kVAR) prior to opening the EDG output breaker could lead to a reverse power trip. As a result, procedure change requests to open the EDG output breaker when load was between 200-500 kW and 200-500 kVAR were generated. However, the station procedures were never revised, due to an administrative error.

The failure to revise procedures concerning safety-related EDGs was an example where the requirements of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Actions,” were not met and was a violation. However, this failure constitutes a violation of minor significance and is not subject to formal enforcement action. This LER is closed.

O8.2 (Closed) LER 50-373/99004-01: Average Power Range Monitors Inoperable During Refueling Due to Improper Isolation of Local Power Range Detectors.

This issue was discussed in Inspection Report 50-373/99022; 50-374/99022 and LER 50-374/99004-00. This supplement to the original LER documented that the root

cause of the event was a human performance error by licensed operators who prepared and reviewed the out-of-service to perform the work. The supplement also noted that this problem had occurred on at least one other occasion in 1996. A Non-Cited Violation (50-373/99022-02; 50-374/99022-02) had been previously identified for this issue which was entered into the licensee's corrective action program. This supplement to the LER contained no additional concerns. This LER is closed.

II. Maintenance

M1 Conduct of Maintenance

M1.1 Surveillance Observations

a. Inspection Scope (61726, 62707, 92902)

The inspectors observed all or portions of the following surveillance test activities. Included in the inspection was a review of the surveillance test procedures, as well as the appropriate UFSAR sections regarding the activities. The inspectors verified that the surveillance tests for the activities observed met Technical Specification (TS) requirements.

- LaSalle Operating Surveillance (LOS) RI-Q5, "Reactor Core Isolation Cooling (RCIC) System Pump Operability, Valve Inservice Tests in Conditions 1, 2, 3 and Cold Quick Start - Attachments 1A and 2A," Revision 12
- LOS-RH-Q1, "RHR [Residual Heat Removal] (LPCI) [Low Pressure Coolant Injection] and RHR Service Water Pump and Valve Inservice Test for Operational Conditions 1, 2, 3, 4 and 5," Revision 47
- LOS-DG-M2, "1A Diesel Generator Operability Test - Attachment 1A - Idle," Revision 42
- LOS-FP-R6, "Preaction Sprinkler Systems Functional Test - Unit 2 Cable Spreading Room," Revision 0
- LOS-VG-M1, "Standby Gas Treatment (SBGT) System Operability Test and Inservice Test of 1(2)VG001, 1(2)VG003, and 1(2)VG02Y," Revision 24
- LaSalle Mechanical Maintenance Surveillance (LMS) DG-01, "Main Emergency Diesel Unit Surveillances," Revision 25
- LaSalle Technical Surveillance (LTS) 300-3, "Secondary Containment Leak Rate Test," Revision 13
- LaSalle Instrument Surveillance (LIS) NB-102, "Unit 1 Reactor Vessel Water Level Post Accident Monitoring and Remote Shutdown Indication Calibration," Revision 18

b. Observations and Findings

The inspectors observed the surveillance activities identified above. The results of the surveillance testing were within the acceptance criteria, and the performance of the surveillance activities was adequate, with the exception of the items discussed below.

Unit 1 LOS-RI-Q5 Surveillance Observations: On February 14, 2000, the inspectors observed quarterly surveillance testing of the Unit 1 RCIC pump in accordance with LOS-RI-Q5. During a system walkdown, the inspectors identified water underneath the

RCIC turbine casing. Further investigation identified a RCIC turbine casing steam leak. To address this issue, the licensee generated an action request to remove all turbine casing lagging pads and inspect the turbine for evidence of any additional steam leaks.

Unit 1 LOS-RH-Q1 Surveillance Observations: On February 15, 2000, the inspectors observed quarterly surveillance testing of the 1C RHR pump in accordance with LOS-RH-Q1. During a system walkdown, the inspectors identified two packing leaks and a broken pump room ventilation louver. The licensee generated PIF 2000-00744 to enter these issues into the corrective action program and initiated action requests to correct the problems.

Unit 2 LOS-RI-Q5 Surveillance Observations: On February 29, 2000, the inspectors observed quarterly surveillance testing of the Unit 2 RCIC pump in accordance with LOS-RI-Q5. During the performance of surveillance Step 4.4, the inspectors questioned the local operator and system engineer on how to verify the RCIC turbine exhaust sightglass was not above the indicated normal level, since there was no explicit labeling or indications for normal level. The system engineer responded that the normal level corresponded to a black pencil mark on the side of the sight glass. The inspectors questioned whether the pencil mark provided sufficient operator guidance. The system engineer subsequently submitted a labeling request to better define the normal RCIC turbine exhaust sightglass level for Unit 1 and Unit 2.

Unit 2 LOS-VG-M1 Surveillance Observations: On March 10, 2000, the inspectors observed the performance of LOS-VG-M1 for the Unit 2 SBGT System. During the surveillance test, the inspectors identified that the SBGT supply fan had one very loose hold down bolt, and the remaining three hold down bolts were not shimmed properly. Also, piping on the Unit 2 SBGT train was mislabeled and indicated that the system discharged into the reactor building. The line in question was not a discharge line, but was a source of suction from the reactor building atmosphere. The licensee generated PIFs L2000-01253 and L2000-01268 to enter these issues into the corrective action program.

Secondary Containment Leakage Testing Observations: On March 17, 2000, the inspectors observed surveillance testing of the Unit 1 and Unit 2 secondary containments in accordance with LTS-300-3. The surveillance measured the ability of the SBGT system to establish and maintain a vacuum within the secondary containment. Two previous attempts to perform the surveillance had been unsuccessful due to material condition problems encountered just prior to or during the surveillance. The inspectors attended the pre-job brief for the surveillance and observed the evolution from both the control room, reactor building, and refueling floor. During the first attempt to perform the surveillance, reactor building exhaust ventilation damper 2VR05YB exhibited dual position indication and the test was halted. Following adjustments to a damper limit switch, the test was resumed. The second attempt to perform the surveillance was successful. All equipment performed as expected and all Technical Specification surveillance requirements were satisfied.

c. Conclusions

Overall, the performance of surveillance activities was acceptable and demonstrated that systems were capable of performing their intended safety function. In some cases, however, the inspectors identified examples in which material condition deficiencies

associated with safety-related systems had not been identified despite numerous opportunities during tours by operations personnel, maintenance activities by maintenance personnel, and system walkdowns by engineering personnel.

M8 Miscellaneous Maintenance Issues

M8.1 (Closed) LER 50-373/200003-00: Inadequate Implementation of Technical Specification Surveillance Requirements Results in Missed Surveillance on Reactor Core Isolation Cooling (RCIC) System Time Delay Relays.

On February 17, 2000, during a review of all safety-related time delay relays listed in LaSalle Technical Procedure (LTP) 100-6, "Time Delay Relay Calibration Program," Revision 5, the licensee discovered that the quarterly channel calibration requirement for the RCIC steam line high flow isolation logic found in Technical Specification Table 4.3.2.1-1, Item 4a did not include time delay relays 1(2)E51A-K47 and 1(2)E51A-K48. The associated pressure switches were calibrated quarterly; however, the time delay relays were calibrated each refueling cycle. Since the time delay relays were part of the instrument channels, the relays were required to be calibrated quarterly.

The licensee determined that the root cause of the event was a personnel error resulting from inadequate knowledge of industry standards when the time delay was added to the instrument channel in 1982, and when the subsequent system reviews were completed in 1991 and 1997. An extent-of-condition review of all safety-related time delay relays identified that only RCIC steam line high flow time delay relays 1(2)E51A-K47 and 1(2)E51A-K48 were not calibrated at the same interval as the associated instrument channel. LaSalle Instrument Surveillance LIS-RI-101/201, "Unit 1 and 2 RCIC Steam Line High Flow Isolation Calibration," was subsequently revised to include quarterly calibration of the time delay relays. This event was entered in to the licensee's corrective action program as ATM Item 23924.

The safety significance of the event was considered low since during the subsequent calibration, the as-found settings for the RCIC time delay relays were found to be within the calibration tolerance. The failure to perform quarterly calibrations of RCIC steam line high flow isolation time delay relays 1(2)E51A-K47 and 1(2)E51A-K48 was considered a violation of Unit 1 and Unit 2 Technical Specification surveillance requirement 4.3.2.1, Table 4.3.2.1-1, Item 4a. However, this failure constitutes a violation of minor significance and is not subject to formal enforcement action. This LER is closed.

III. Engineering

E2 Engineering Support of Facilities and Equipment

E2.1 Resolution of Appendix J Moderate Energy Line Break (MELB) Discrepancies

a. Inspection Scope (37550, 37551, 92903)

As discussed in Section O1.1, during a walkdown of the Unit 2 RCIC system, the inspectors identified support brackets missing from the spray shield associated with piping line 2RI-24A-6. Licensee personnel performed additional walkdowns and

identified examples in which the installed configuration of spray shielding in the plant did not conform with the description in Appendix J, “Analysis of the Effects of Moderate Energy Line Through Wall Leakage Cracks Outside Primary Containment,” of the UFSAR.

The inspectors reviewed the actions to address discrepancies between the UFSAR description of installed spray shielding for MELB and the installed plant configuration. Documents reviewed included the following:

- PIF L1998-02790 dated January 27, 1998
- PIF L1998-00665 dated April 20, 1998
- Sargent & Lundy (S&L) Report SCM-05069, “Engineering Evaluation of LaSalle Unit 1 and 2 Moderate Energy Line Break (MELB) UFSAR Appendix J,” dated May 1, 1998
- Letter from Rich Hall, Mechanical Group Lead, Design Engineering, to Design Engineering Supervisor, “LaSalle Unit 1 and Unit 2 Moderate Energy Line Break,” dated May 1, 1998
- Letter from G. Zwarich, Mechanical Engineer, LaSalle, to A. Benesh, Project Manager, Universal Fabricated Products Division, Transco Incorporated, dated June 12, 1981
- PIF L2000-01182, “Open Holes On Top of MCC [Motor Control Center] 135-2, 235Y-2,” dated March 7, 2000
- NRC Branch Technical Position ASB 3-1, “Plant Design for Protection Against Postulated Failures in Fluid Systems Outside Containment”

b. Observations and Findings

The inspectors determined that PIFs L1998-00665 and L1998-02790 had identified issues concerning the conformance of installed spray shielding to the UFSAR. In particular, these PIFs identified that there were distinct differences between shielding on piping in the RCIC room on Unit 1 and Unit 2. Also, no evidence of spray shielding on the RHR system piping, RHR pumps, or RHR service water pumps existed although depicted by drawings in the UFSAR. The drawings in the UFSAR also depicted shielding around motor control center (MCC) panels although this shielding was not present in the field.

To address these concerns, S&L completed SCM-05069, “Engineering Evaluation on LaSalle Unit 1 and Unit 2 Moderate Energy Line Break UFSAR Appendix J,” dated May 1, 1998. The evaluation documented that the licensing basis for MELB was NRC Branch Technical Position ASB 3-1 along with the additional information provided in Appendix J of the UFSAR. The evaluation concluded that although not all the spray shielding identified in Appendix J of the UFSAR was currently installed in the plant, the existing spray shielding was acceptable. The report also concluded that Appendix J of the UFSAR should be updated to reflect the current plant configuration. Design Engineering reviewed and concurred with the documented evaluation by letter dated May 1, 1998.

The inspectors reviewed documentation, including S&L report SCM-05069, conducted walkdowns of installed plant equipment, and identified the following concerns:

Justification for Acceptance of Shielding Discrepancies: The S&L report concluded in many cases that although MELB shielding was not present in locations indicated in the UFSAR, the configuration was still acceptable since the shielding was not required per Section B.1.c of NRC Branch Technical Position ASB 3-1. This section stated that plant arrangement or system features that did not meet MELB physical protection requirements should be limited to those for which protection is impractical because of the stage of construction of the plant, because the plant design is based on an earlier plant accepted by the NRC, or for other substantive reasons. The inspectors were unable to determine following a review of the report, and the licensee had not determined at the end of the inspection period, which exception had been credited.

Shielding Assumption Deficiencies: The inspectors reviewed a letter from G. Zwarich, Mechanical Engineer, LaSalle, to A. Benesh, Project Manager, Universal Fabricated Products Division, Transco Incorporated, which confirmed decisions regarding the placement of spray shields to address MELB events. In that letter, piping that was identified to require spray shielding assumed that all MCCs were drip-proof from above. The inspectors conducted a walkdown of MCCs protected against MELB shielding and identified that these MCCs were not drip-proof since holes due to missing bolts and unused cable penetrations existed. The inspectors' observations were entered in the licensee's corrective action program as PIF L2000-01182.

To address the issues, the licensee planned to re-review the S&L MELB report, clearly define the LaSalle design criteria for MELB and document the review in an engineering report, and revise the UFSAR as appropriate. This issue is considered an Unresolved Item (URI 50-373/200003-01(DRP); 50-374/200003-01(DRP)) pending NRC review of the results of these actions.

c. Conclusions

Discrepancies between the UFSAR description of installed spray shielding to address MELB and the installed plant configuration were identified in 1998, but had not been adequately addressed. In addition, the inspectors independently identified material condition deficiencies associated with MCCs and spray shielding which potentially impacted the ability to mitigate the effects of a MELB.

E8 Miscellaneous Engineering Issues

E8.1 (Closed) LER 50-374/200003-00: Clerical Errors During Construction Results in Missed Augmented Weld Examinations for Residual Heat Removal System Welds RH-2005-28 and RH-2005-29.

During a review of historical intergranular stress corrosion cracking (IGSCC) weld examination data, including Inductive Heat Stress Improvement (IHSI) data and Mechanical Stress Improvement (MSIP) records, the licensee discovered a clerical error involving two IGSCC susceptible welds in the 2A RHR shutdown cooling return piping. The two welds were RH-2005-28 and RH-2005-29 which connect the upstream piping and a downstream elbow to valve 2E12-F090A, "'An' RHR Shutdown Cooling Return Header Manual Stop Valve." Due to the clerical error, the IHSI stress relief that was believed to have been applied to these two welds in 1983 was actually applied to two other welds. The error went undetected, and in 1987 when the remaining balance of the IGSCC weld population was subjected to the MSIP process, it was believed that

welds RH-2005-28 and RH-2005-29 had already been stress relieved in 1983. Consequently, the two welds were never subjected to any stress improvement process.

Based on having had no stress improvements, these two welds should have been categorized as IGSCC Category D welds in accordance with NRC Generic Letter 88-01. The Category D examination schedule required the two welds to be examined during the 3rd, 5th, and 7th refueling outages. Examinations of weld RH-2005-28 were not completed during the 3rd or 5th refueling outages, but was completed during the 7th refueling outage with no indication of cracking noted. Therefore, weld RH-2005-28 was not required to be examined again until the 9th refueling outage (L2R09) scheduled for the Fall of 2002. Weld number RH-2005-29 was examined in the 3rd refueling outage with no indication of cracking noted, but was not re-examined during the 5th and 7th refueling outages. Examination of either of these welds required the plant to be in a shutdown condition. The inspectors reviewed RH-2005-28 and RH-2005-29 construction radiographs, dye penetrant examinations, and baseline and inservice ultrasonic examination data. In all cases, no indications of inservice cracking were noted in either RH-2005-28 or RH-2005-29. The licensee also performed a risk assessment and determined that the change in core damage frequency and large early release frequency associated with deferring this examination were lower than the risks associated with shutting down the reactor for a forced outage.

Technical Specification Section 3.4.8 required the structural integrity of American Society of Mechanical Engineers (ASME) Class 1 components to be maintained in accordance with the surveillance requirements of TS 4.4.8. Technical Specification 4.4.8 invoked the surveillance requirements of TS 4.0.5. Technical Specification 4.0.5 required that piping susceptible to IGSCC be examined in accordance with the NRC staff positions on schedule, methods, personnel and sample expansion included in Generic Letter 88-01. The failure to inspect weld RH-2005-29 during the 7th refueling outage constituted a missed surveillance requirement and necessitated entry into TS 3.4.8, Action A, for Class 1 piping. Technical Specification 3.4.8, Action A, required restoration of the structural integrity of the weld to within its limit or isolation of the affected component prior to increasing reactor coolant system temperature more than 50 degrees fahrenheit (°F) above the minimum nil ductility temperature. Since Unit 2 was operating at full power at the time the missed weld examination was discovered, the action statement could not be complied with and entry into limiting condition for operation (LCO) action statement 3.0.3 was required.

Limiting condition for operation action statement 3.0.3 required that within 1 hour action be taken to place the unit in an operational condition in which TS 3.4.8 did not apply by placing the unit in startup within the next 6 hours, hot shutdown within the following 6 hours and cold shutdown within the subsequent 24 hours. Since TS 3.0.3 required actions to place Unit 2 in a shutdown condition in a time less than 24 hours, surveillance requirement 4.0.3 was applied due to the missed surveillance requirement of TS 4.4.8. Technical Specification surveillance requirement 4.0.3 allowed 24 hours to resolve the issue. During that period, LaSalle County Station requested and received a Notice of Enforcement Discretion to allow the deferral of the RH-2005-29 weld examination until the upcoming refueling outage scheduled for November 2000 or December 31, 2000, whichever was earlier. License amendment 123 was subsequently issued on March 22, 2000, to formally amend Unit 2 Facility Operating License NPF-18 to defer this inspection.

The safety significance of having missed the examination of weld RH-2005-29 and the likelihood of the weld failing prior to the next inspection was considered low due to the following:

- There were 432 IGSCC Category D welds that were examined 1325 times at 33 nuclear power plants in the county. Of this population, there was only one known Category D weld to have cracked. The crack occurred in a dissimilar metal weld that had experienced multiple repairs. Weld RH-2005-29 was not a dissimilar metal weld and had not been repaired.

- Industry experience suggested that small-bore, thin-wall piping would develop flaws early in life due to higher weld residual stresses. Consequently, if weld RH-2005-29 was going to crack, it would probably have done so early in life and prior to the inservice ultrasonic examination performed in 1990.

- Chemistry parameters had been maintained within Electric Power Research Institute guidelines 93.7 percent of the time since the last inspection of weld RH-2005-29 in 1990 and when a non-standard condition developed, the guidelines were followed to restore chemistry to within normal operating parameters.

- The weld and base metal material of RH-2005-29 were austenitic stainless steel which was a ductile material. Therefore, if a crack were to develop, the leakage detection system would detect the leak before a pipe break occurred.

- The weld on the opposite side of the valve, RH-2005-28, was also a Category D weld. The operating environment for weld RH-2005-28 was similar to that for weld RH-2005-29 since both were exposed to the same pressure, temperature, and fluid. Weld RH-2005-28 was inspected during the 7th refueling outage with no recordable indications. Since the operating environments and materials of construction for each weld were similar, the licensee concluded that weld RH-2005-29 would have behaved similarly as RH-2005-28 and therefore, was not expected to be flawed.

The failure to examine weld RH-2005-29 during the 5th and 7th refueling outages was considered a violation (50-374/2000003-02(DRP)) of Technical Specification requirement 3.4.8 which directed that the structural integrity of ASME Code Class 1 components be maintained in accordance with TS 4.4.8. However, this Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy. The safety significance of having missed the examinations of weld RH-2005-29 and the likelihood of the weld failing before the next inspection were considered low. This item was entered into the licensee's corrective action program as PIF L2000-00790.

The failure to examine weld RH-2005-28 during the 3rd and 5th refueling outage was also considered a violation of TS 3.4.8. However, since weld RH-2005-28 was inspected during the 7th refueling outage and found to have no reportable indications, the failure to examine the weld during the 3rd and 5th refueling outages was considered a violation of minor significance and was not subject to formal enforcement action.

E8.2 (Closed) LER 50-374/97001-01: Discovery of a Broken Fuel Rod in the LaSalle Unit 2 Spent Fuel Pool.

This issue was discussed in Inspection Report 50-373/97011(DRP); 50-374/97011(DRP) and LER 50-374/97001-00. This supplement to the original LER documented that the broken fuel section had been recovered, and all components had been accounted for. A Non-Cited Violation (50-373/97011-07; 50-374/97011-07) had been previously identified for this issue which was entered into the licensee's corrective action program. This LER is closed.

IV. Plant Support

R1 Radiological Protection and Chemistry (RP&C) Controls

R1.1 General Comments (71750)

During routine inspections in radiologically controlled areas, the inspectors assessed licensee performance. Overall, the radiation protection staff enforced radiological control standards. No deficiencies in radiation control were identified.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the results of these inspections to licensee management at an exit meeting on March 31, 2000. The licensee acknowledged the findings presented. The inspectors asked the licensee if any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Commonwealth Edison

C. Pardee, Site Vice President
J. Meister, Station Manager
D. Bost, Site Engineering Manager
W. Riffer, Nuclear Oversight Manager
G. Kaegi, Site Training Manager
R. Gilbert, Operations Manager
F. Spangenberg, Regulatory Assurance Manager
J. Pollock, System Engineering Manager
F. Gogliotti, Design Engineering Supervisor
J. Burns, Chemistry Supervisor
S. Taylor, Radiation Protection Manager

INSPECTION PROCEDURES USED

IP 37551	Onsite Engineering
IP 61726	Surveillance Observation
IP 62707	Maintenance Observation
IP 71707	Plant Operations
IP 71750	Plant Support Activities
IP 92700	Onsite Follow-up of Written Reports of Nonroutine Events
IP 92901	Followup - Plant Operations
IP 92902	Followup - Maintenance
IP 92903	Followup - Engineering

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-373/374/200003-01	URI	MELB Analysis Discrepancies
50-374/200003-02	NCV	Failure to Examine Weld During Refueling Outage

Closed

50-374/200003-02	NCV	Failure to Examine Weld During Refueling Outage
50-374/00001-00	LER	High Pressure Core Spray Emergency Diesel Generator Unavailable Due to Diode Failure
50-373/99004-01	LER	APRMs Inoperable During Refueling
50-373/200003-00	LER	Missed Surveillance on RCIC Time Delay Relays
50-374/97001-01	LER	Broken Fuel Rod in the Unit 2 Spent Fuel Pool
50-374/200003-00	LER	Missed Examinations for RHR System Welds

Discussed

None

LIST OF ACRONYMS USED

APRM	Average Power Range Monitor
ASME	American Society of Mechanical Engineers
ATM	Action Tracking Matrix
DRP	Division of Reactor Projects
EDG	Emergency Diesel Generator
EMD	Electrical Maintenance Department
°F	Degrees Fahrenheit
HPCS	High Pressure Core Spray
IGSCC	Intergranular Stress Corrosion Cracking
IHSI	Inductive Heat Stress Improvement
kVAR	Kilovolt-Amperes-Reactive
kW	Kilowatts
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LES	LaSalle Electrical Maintenance Surveillance
LIS	LaSalle Instrument Surveillance
LMS	LaSalle Mechanical Maintenance Surveillance
LOP	LaSalle Operating Procedure
LOS	LaSalle Operating Surveillance
LPCI	Low Pressure Coolant Injection
LTP	LaSalle Technical Procedure
LTS	LaSalle Technical Surveillance
MCC	Motor Control Center
MELB	Moderate Energy Line Break
MSIP	Mechanical Stress Improvement
NSO	Nuclear Station Operator
PERR	Public Electronic Reading Room
PIF	Problem Identification Form
RBCCW	Reactor Building Closed Cooling Water
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
RP&C	Radiation Protection and Chemistry
rpm	Revolutions Per Minute
SBGT	Standby Gas Treatment
S&L	Sargent & Lundy
SBLC	Standby Liquid Control
SFPR	System Functional Performance Review
SJAE	Steam Jet Air Ejector
TDRFP	Turbine-Driven Reactor Feedwater Pump
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
Vdc	Volts direct current