U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Reports No. 50-373/86021(DRP); 50-374/86020(DRP)

Docket Nos. 50-373; 50-374 Licenses No. NPF-11; NPF-18

Licensee: Commonwealth Edison Company
Post Office Box 767
Chicago, IL 60690

Facility Name: LaSalle County Station, Units 1 and 2

Inspection At: LaSalle Site, Marseilles, IL

Inspection Conducted: May 10 through June 11, 1986

Inspectors: M. J. Jordan
J. Bjorgen
R. Kopriva

Approved By: G. C. Wright, Chief
Reactor Projects Section 2C

8/4/86
Date

Inspection Summary

Inspection on May 10 through June 11, 1986 (Reports No. 50-373/86021(DRP); 50-374/86020(DRP))
Areas Inspected: Routine, unannounced inspection conducted by resident inspectors of operational safety; surveillance; maintenance; training; unit trips; refueling/outage; and regional requests.
Results: Of the seven areas inspected, no violations or deviations were identified in six areas; two violations were identified in the remaining area (failure to follow procedures, Paragraph 2). During this inspection period, several times the operator log and shift engineer log did not adequately reflect the shift occurrences. This is a continuing problem at LaSalle. In addition, personnel errors and failure to follow procedures continues to be a problem.

DETAILS

1. Persons Contacted

*G. J. Diederich, Manager, LaSalle Station
*R. D. Bishop, Services Superintendent
*C. E. Sargent, Production Superintendent
D. Berkman, Assistant Superintendent, Technical Services
W. Huntington, Assistant Superintendent, Operations
J. C. Renwick, Assistant Superintendent, Work Planning
R. W. Stobert, Quality Assurance Supervisor
P. Manning, Tech Staff Supervisor
T. Hammerich, Assistant Tech Staff Supervisor
W. Sheldon, Assistant Superintendent, Maintenance
*J. Atchley, Operating Engineer
*D. Winchester, Senior Quality Assurance Inspector

The inspectors also talked with and interviewed members of the operations, maintenance, health physics, and instrument and control sections.

*Denotes personnel attending the exit interview held on June 11, 1986.

2. Operational Safety Verification (71707)

The inspector observed control room operations, reviewed applicable logs and conducted discussions with control room operators during the inspection period. The inspector verified the operability of selected emergency systems, reviewed tagout records, and verified proper return to service of affected components. Tours of Units 1 and 2 reactor buildings and turbine buildings were conducted to observe plant equipment conditions, including potential fire hazards, fluid leaks and excessive vibrations and to verify that maintenance requests had been initiated for equipment in need of maintenance. The inspector by observation and direct interview verified that the physical security plan was being implemented in accordance with the station security plan.

The inspector observed plant housekeeping/cleanliness conditions and verified implementation of radiation protection controls.

The inspectors, while touring the plant, noticed oil leakage from some of the post tension connections for containment on Unit 2. This leakage was brought to the attention of the station management for evaluation based on a similar problem at the Farley Unit 2 plant addressed in IE Information Notice 85-10, Supplement 1. The licensee reported back that no problem existed with the post tension connections because less than one gallon total leakage had occurred and less than one percent of that contained in any one tension was noticed. The next grease coverage check surveillance required by technical specifications is scheduled for late 1986. Since no free standing water was noticed in the tension cap, no further action was needed at this time. The inspector also reviewed the method by which IE

Notice 85-10, Supplement 1, was implemented at the site. LTS 1000-1 had been implemented to require a laboratory analysis of the grease samples and identified the action to be taken if the samples indicate free water.

On May 26, 1986, at approximately 21% power on Unit 2, the unit operator was pulling control rods during a normal power increase in accordance with the Unit Start-up Procedure LGP-1-1. The operator incorrectly pulled Control Rod 10-19 from Position 12 to Position 48 instead of from Position 12 to Position 24. The Rod Worth Minimizer (RWM) was bypassed at the time and a second operator was being used as a verifier to assure that rod movements complied with the rod movement sequence.

The unit operator selected the next rod in the sequence, Rod 10-43, and pulled it one notch, from Position 12 to Position 14 before the movement error for Rod 10-19 was identified. He then reinserted Rod 10-19 to the correct position of 24. The total elapsed time during this sequence was approximately one minute based on a review of the process computer alarm typer. The unit operator then contacted the nuclear engineer on-call to verify that the actions taken were correct. The nuclear engineer approved the operator's actions based on his knowledge of the rod sequence controls and preferred operator actions and authorized continued rod movements. The nuclear engineer subsequently reviewed the event on his home computer console to confirm nothing abnormal had occurred due to the mispositioned rod. Reactor power was confirmed to be above the 20% technical specification limit thereby eliminating the need for the RWM or for a second person to act as a verifier.

Later the nuclear engineer stated to the inspector that the RWM would allow movement of the noted rod to position 48 before an "out-of-sequence" light would be displayed. Furthermore, the Rod Sequence Control System (RSCS), which also does not enforce rod blocks above 20% power, would not prevent rod movement above notch position 12.

A thorough review of this event was performed by the inspector. This review included the applicable operator and shift engineer logs, the applicable alarm printouts, licensee procedures, discussions with the personnel involved, and the licensee's corrective actions.

Technical Specification 6.2.A.1 requires detailed written procedures to be prepared, approved, and adhered to including the applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978. Section 4 of Appendix A to this Regulatory Guide requires procedures for plant start-up. Licensee Procedure LGP-1-1 for normal plant start-up, Paragraph F.2.h., requires controls rods to be withdrawn in accordance with the approved rod sequence provided by the nuclear engineer.

Contrary to the above, Unit 2 Control Rod 10-19 was withdrawn from Position 12 to Position 48 on May 26, 1986 in lieu of the required position of 24. This is considered to be a violation (374/86020-01(DRP)).

In addition to the mispositioned rod, the inspectors are concerned about three other aspects of this event: the immediate operator actions; the timeliness of the identification of the error by the independent verifier; and the failure of the unit operator to make a log entry describing the error. The licensee's Procedure LOA-RD-03 provides the applicable instructions for mispositioned control rods. For a control rod withdrawn beyond its in-sequence position, Section D.2 of this procedure requires the operator to demand process computer printouts OD-3 (Core Thermal Power), OD-7 Option 2 (Control Rod Position), and OD-8 (Local Power Range Monitor readings) and then to consult the Nuclear Engineer for the method of returning the mispositioned rod to its correct in-sequence position. Contrary to the procedure, on May 26, 1986, the operator failed to demand the required process computer printouts and failed to consult the Nuclear Engineer prior to returning the mispositioned control rod to its correct in-sequence position. This is considered to be a violation (374/86020-02(A)(DRP)). In this case, fortuitously, the operator action was what the nuclear engineer would have recommended. The inspector is concerned, however, that the operator took the action prior to consulting with the nuclear engineer. Depending on circumstances, this practice could complicate a problem rather than help it. In addition, during an interview with the unit operator subsequent to the event it became apparent that the operator was not aware that he had selected and moved another rod prior to discovery of the mispositioned rod, a result of not obtaining and reviewing the required process computer printouts, OD-3, OD-7 Option 2, and OD-8.

The timeliness of the independent verifier noting the error was the second concern. The inspector noted that the duties of the verifier, i.e., what was expected of him, were not clearly defined. However, the action statement for Technical Specification 3.1.4.1 (Rod Worth Minimizer) states that with the RWM inoperable control rod movement and compliance with the prescribed control rod pattern shall be verified by a second licensed operator or other technically qualified member of the technical staff who is present at the reactor control console. Therefore, this individual should have been aware of his duties even without the benefit of a prescriptive procedure. (The inspectors must note again, however, that the referenced technical specification does not apply above a reactor power of 20%. Therefore, there was no technical specification requirement for the second operator to verify rod movement.) The licensee plans to revise the appropriate procedures to provide this guidance. Completion of this action will be tracked as an open item (374/86020-02(B)(DRP)).

The failure of the unit operator to log the error is another item of concern. In this case, the operator was instructed not to make an entry until the shift engineer obtained clarification on the seriousness of the error. The shift engineer then neglected to provide the unit operator with the appropriate clarification. The inspector has previously expressed concern with the adequacy of log entries. Furthermore, the inspector is concerned that the operator did not make an entry because of instructions from the shift engineer. The log book in question is the unit operator

log and is required to contain all pertinent information related to operation of the facility. The seriousness of the error has no bearing on whether it should be recorded. The log book is used to record facts; analyses of these facts can be done in another forum.

Secondly, the inspectors are concerned that the shift engineer instructed the unit operator not to make the initial entry. As the senior management representative onsite he should be aware of the procedural requirements. If he so desires, his log book can describe clarifications of operational events but he should not instruct operators to not record information.

Corrective actions for this example of inadequate logs will be monitored as an open item (374/86020-02(C)(DRPP).

In summary, a control rod inadvertently was mispositioned to Position 48 instead of to Position 24. Although not required by technical specifications for the operational conditions at the time (21% power) a second operator had been assigned to the control room as a verifier but did not notice the mispositioned rod until the next rod in the sequence had been moved.

During the month of May, the inspector walked down the accessible portions of the following systems to verify operability:

Units 1 & 2 Emergency Diesel Generators
Units 1 & 2 Standby Gas Treatment Systems

3. Monthly Surveillance Observation (61726)

The inspector observed technical specifications required surveillance testing and verified for actual activities observed that testing was performed in accordance with adequate procedures, that test instrumentation was calibrated, that limiting conditions for operation were met, that removal and restoration of the affected components were accomplished, that test results conformed with technical specifications and procedure requirements and were reviewed by personnel other than the individual directing the test, and that any deficiencies identified during the testing were properly reviewed and resolved by appropriate management personnel.

The inspector witnessed portions of the following test activities:

LIS-NB-214	Calibration of Reactor Vessel Pressure Switch 2B21N039N
LST 86-096	Unit 2 Special Test to Check Trip Point of Low Level Scram Switches
LIS-NR-402	Intermediate Range Monitor Rod Block and Reactor Scram Functional Test
LIS-NB-204	Unit 2 Reactor Vessel Low-Low Water Level RCIC Initiation and Low-Low-Low LPCS/RHR Initiation Calibration

On June 1, 1986, the licensee was performing a surveillance test on the "A" turbine driven feedwater pump when problems developed and water level dropped below the scram setpoint but the reactor failed to scram. A preliminary notification (PNO-III-86-52) was issued by Region III and an Augmented Investigation Team was dispatched to the site to investigate the matter. This entire issue will be addressed in Special Inspection Report No. 50-374/86023.

On May 25, 1986 at approximately 35% power on Unit 2, the licensee attempted to shift the reactor recirculation pumps from slow speed to fast speed. The "2B" pump tripped on neutral overcurrent. This left the "2B" recirculation pump off and the "2A" pump in slow speed. A discussion was held among the plant management concerning meeting the technical specifications for operation with one recirculation pump and it was determined that resetting of the Average Power Range Monitors (APRM) and the Rod Block Monitor (RBM) could not be accomplished within the four hours allowed by technical specifications so a reactor shutdown was commenced. The reactor power was then reduced to less than 20% power by inserting rods. Technical Specifications 4.1.4.1 and 4.1.4.2 require that surveillance testing of the Rod Worth Minimizer (RWM) and the Rod Sequence Control System (RSCS) be performed prior to reducing power to less than 20%. However, this was not recognized by the licensee and therefore these surveillances were not performed during this shutdown (the surveillance had been performed on May 23, 1986 prior to commencing the startup). Subsequent to the event a discussion between the inspector and shift engineer revealed the RSCS and RWM were enforcing during the power reduction from 20% to 15%. At approximately 15% power the "2B" recirculation pump was restarted (successfully) at slow speed and reactor shutdown was stopped. The RSCS and RWM then were calibrated prior to power ascension.

The missed surveillances during reactor shutdown were identified by the next oncoming shift control room engineer during shift turnover. Further evaluation by the licensee and by the inspectors revealed that Procedure LGP 3-1, "Power Changes," which the shift was using to change power did not have any precautions or warnings that these surveillances had to be performed before reducing power below 20%. A change to this procedure was initiated. No violation will be issued at this time concerning the missed surveillance because the licensee notified the NRC, took prompt corrective action, and this issue meets the requirements for not issuing a violation specified in 10 CFR Part 2, Appendix C.

The inspectors reviewed the log books for this period of time and determined that neither the unit operator log nor the shift engineer log contained any information relative to any problem occurring with missed surveillances. There was no reference to any technical specification violation covering missed surveillance. The procedure for both of these logs (LAP 220-1 and LAP 220-2) requires the logging of any technical specification violation, corrective actions, and time of return to technical specifications. Therefore, this is a violation of the regulations and a citation normally would be issued by the NRC. Although the licensee did not identify this particular instance of failure to log the required information, the licensee has acknowledged the generic problem of failure to record

significant information in log books at LaSalle and has identified corrective actions to improve the quality of the log books. At the time of this latest violation, the corrective action programs had been started but not yet fully implemented. Therefore, the inspectors believe that this issue meets the intent of the requirements for not issuing a Notice of Violation as set forth in 10 CFR Part 2, Appendix C and that issuing a violation at this time would serve no purpose. However the licensee has been informed that appropriate enforcement action will be taken for future violations.

4. Monthly Maintenance Observation (62703)

Station maintenance activities of safety related systems and components listed below were observed/reviewed to ascertain that they were conducted in accordance with approved procedures, regulatory guides and industry codes or standards and in conformance with technical specifications.

The following items were considered during this review: the limiting conditions for operation were met while components or systems were removed from service; approvals were obtained prior to initiating the work, activities were accomplished using approved procedures and were inspected as applicable; functional testing and/or calibrations were performed prior to returning components or systems to service; quality control records were maintained; activities were accomplished by qualified personnel; parts and materials used were properly certified; radiological controls were implemented; and fire prevention controls were implemented.

Work requests were reviewed to determine status of outstanding jobs and to assure that priority is assigned to safety related equipment maintenance which may affect system performance.

The following maintenance activities were observed/reviewed:

The inspector observed replacement of the Unit 1 Control Rod 34-23 Hydraulic Control Unit 111 valve (Work Request L 57655), using Procedure LMP-RD-07.

The inspector also observed the partial disassembly and inspection of the Unit 2 Residual Heat Removal (RHR) heat exchanger discharge valve, 2E12F003A, to investigate the reason for its failure to operate on June 3, 1986. The valve is a normally open valve which is required to be closed during warming of the shutdown cooling piping in preparations for entering cold shutdown. After warming is completed, the valve must then be reopened to place the RHR heat exchanger in service for shutdown cooling. The valve had failed to reopen either remotely or locally when trying to enter cold shutdown. The "A" RHR shutdown cooling loop was declared inoperable and the limitorque operator was removed for repairs. Upon disassembly, the valve operator was found to have a broken drive sleeve and damaged drive gear teeth.

Discussions with licensee personnel indicate that this valve has a history of hydraulic lock, a condition that allows leakage past the valve seating surface into the bonnet area. This high pressure water then provides a hydraulic lock between the valve bonnet and the wedge, preventing valve movement. To release the hydraulic lock, the licensee routinely loosens the valve stem packing to vent the bonnet area. The licensee suspects that attempts to open the valve against the hydraulic lock may have resulted in overtorquing the operator and causing the failure. The drive sleeve failure also suggests the possibility of a fatigue failure due to the presence of rust and signs of aging in portions of the break area.

The inspector continues to be concerned about the possible generic problem of this failure for other limitorque valve operators as to the root cause of equipment failure and corrective actions as well as how to resolve hydraulic lock problems of valves in this system and other systems. Until licensee evaluations and corrective action plans are completed, this will remain as an open item (374/86020-02(DRP)).

5. Training (41400)

The inspector, through discussions with personnel and a review of training records, evaluated the licensee's training program for operations and maintenance personnel to determine whether the general knowledge of the individuals was sufficient for their assigned tasks.

Specific areas reviewed are identified in Paragraphs 2, 3, and 4. The adequacy of training to prevent personnel from over torquing limitorque valve operators was identified as a concern. Personnel have not been provided with specific instructions that would limit the amount of force applied to valve operators. This contributed to the failure of a valve operator as noted in Paragraph 4. This concern was identified to the licensee for evaluation.

6. Unit Trips (93702)

On May 9, 1985 at 9:10 a.m. CDT, LaSalle Unit 2 experienced a reactor scram from 85% reactor power. The unit scrammed on low reactor water level due to the loss of power to the feedwater control system. With a loss of the feedwater control system, the "B" Turbine Driven Reactor Feedwater (TDRFP) Pump "locked up" and maintained a constant speed. The "2A" TDRFP did not lock up and coasted down. The motor driven feedwater pump was started but its feedwater control valve locked up at 20% open. With a decrease in feedwater flow, vessel level decreased to the low reactor water level scram setpoint and the unit scrammed. The loss of power to the feedwater control system was caused by a worker accidentally bumping and tripping a 120V power supply disconnect breaker to the feedwater control system. The licensee determined the lock up of the flow control valve and the coast down of the ATDFWP was due to the loss of power to the feedwater controller.

The inspector attended a meeting with licensee personnel on the scram and a discussion was held on possible delay in receiving a low level scram when it should have occurred. Discussions with the shift engineer indicated that he was watching the wide range recorder and saw the recorder reading eight inches. However, based on previous experience with the indication on the recorder, a six inch difference between indicated recorder level and actual level was known to exist. The reason for the difference is that these recorders are calibrated with no flow in the recirculation system. Flow in the recirculation system causes the recorder to read low.

The licensee then checked the calibration on the level switches and the control room narrow range indicator in the control room. No problems were identified. Due to a similar problem with these level switches during a feedwater transient on June 1, 1986, this event will be reevaluated and documented in a special report (374/86023).

On May 11, 1986 at 4:42 a.m. while returning Unit 2 to power from the May 9 scram, the unit received a Group I isolation on indicated high main steam line flow and a reactor scram. The Group I isolation was received while synchronizing the main turbine generator to the grid and increasing load. All systems functioned as expected. The licensee investigated the cause of the high main steam line flow signal and was unable to determine the cause. A walkdown of all steam lines indicated no problem. Testing of the EHC system indicated no problem. A review was made of the electrical systems and no cause could be determined. The unit was returned to power on May 24, 1986 after a short maintenance outage. The licensee replaced the seals on the "B" recirculation pump and worked on leaking valve packing seals. The licensee connected recorders to the high steam flow sensors and switches during the startup to see if they could determine the cause of the May 11 scram. The inspector was present in the control room during the turbine roll and synchronizing of the generator to the grid. No problems were identified at that time. See Paragraph 3 for events occurring later on in the startup.

7. Refueling/Outage (61701)

During a May 21, 1986, Unit 1 cold hydrostatic test of the primary piping, two small leaks were found in a weld on a plugged hole which had been welded up during construction. The leak rate was less than a drop a minute. The leaks were on the B and C low pressure injection line between the reactor vessel and the inboard manual isolation valve, and as such, were unisolatable. The licensee has determined the corrective action for repair is to weld a one inch pipe coupling over the existing welded plug. This will be accomplished prior to starting the unit up.

On May 30 during the performance of LTS 300-4 (Integrated Leak Rate Test) for Unit 1, a Primary Containment Isolation System (PCIS) Group 2 and Group 4 isolation for Unit 1 and Group 4 isolation for Unit 2 occurred. The cause was a high drywell pressure isolation signal due to a momentary open circuit during the installation of a jumper. Upon receiving the PCIS initiation, the reactor building ventilation isolated and both trains of standby gas treatment system started.

The loss of continuity occurred while a station electrician was installing a jumper per procedure LTS 300-4. Per the technical staff engineer's direction, a spade lug jumper was used. As the jumper was being installed, the wire lugs already on the terminal block apparently lost continuity, despite care being taken to maintain continuity. Since the isolation logic is normally energized, the isolation occurred upon loss of the continuity.

Upon receiving the isolation, the spade jumper was removed and all systems reset and returned to normal. The root cause of the isolation stems from an inadequate procedure and a personnel error. A meeting was held to determine and implement a feasible method for installing jumpers without disrupting the existing circuitry. The station electrician attended the meeting and was given instructions on how to eliminate future problems. Also, procedure LTS 300-4 has been changed to incorporate a note as to the consequences of the loss of continuity of the circuit. Due to the fact that there have been several other occurrences of jumpers falling off and/or causing isolations/actuations, the inspector will monitor the actions of the licensee with respect to installation of jumpers and lifted leads.

8. Regional Requests (92705)

The inspector completed a request for assistance on review of the low level radioactive waste storage facility at LaSalle. The request was from C. E. Norelius dated May 16, 1986. The results are as follows:

a. Is the licensee building or planning to build an onsite low-level waste storage facility? When?

   Yes, the facility is scheduled to be completed by August 1986.

b. What is the general method of construction e.g. reinforced concrete, concrete block, butler building, etc.?

   The facility is constructed of reinforced concrete.

c. What is the proposed capacity of the facility in cubic feet of waste, square feet of floor space, and/or maximum curie content?

   The facility has a floor space of 5,700 ft^2 and can store 153,900 ft^3 of waste. The facility is designed with the assumption that each barrel has a reading of 5 R/hr which is more conservative than the average activity distribution using curie analysis.

d. Is the structure to be free standing or will it be attached to some existing plant structure? If it is to be attached to an existing plant structure, what is that structure?

   The facility is free standing and not attached to any existing facility.

e. Has the licensee performed a 50.59 evaluation of the proposed structure? If such an evaluation has been performed, what were the conclusions of the evaluation?

Yes, the licensee performed a 50.59 review by the Station Nuclear Engineering Department (SNED). The review is part of the modification package. No new unresolved safety issues were generated nor were Technical Specifications required to be changed.

f. Has the licensee estimated the contribution to off-site dose rate from the facility? If so, what is the estimated dose rate and what regulatory limits were used to judge the acceptability of this estimated does rate?

The licensee has determined that the offsite dose rate from addition of this facility has not increased above the original design of the entire facility. The regulation used was 40 CFR 190 and EPA regulations. This information was obtained by telecon between the inspector and a SNED representative.

g. Will the facility house low-level waste processing equipment or will it simply be a repository for waste?

The facility is to house low-level waste only.

Inspection of this facility was accomplished by regional based inspectors in September 1985 and is documented in Inspection Reports No. 373/85030; 374/85031.

9. Open Items

Open items are matters which have been discussed with the licensee, which will be reviewed further by the inspector, and which involve some action on the part of the NRC or licensee or both. Open items disclosed during the inspection are discussed in Paragraphs 2 and 4.

10. Exit Interview (30703)

The inspectors met with licensee representatives (denoted in Paragraph 1) throughout the month and at the conclusion of the inspection period and summarized the scope and findings of the inspection activities. The licensee acknowledged these findings. The inspector also discussed the likely informational content of the inspection report with regard to documents or processes reviewed by the inspector during the inspection. The licensee did not identify any such documents or processes as proprietary.