U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-373/82-18(DPRP)

Docket No. 50-373 License No. CPPR-99

Licensee: Commonwealth Edison Company
P. O. Box 767
Chicago, IL 60690

Facility Name: LaSalle County Nuclear Station, Unit 1

Inspection At: LaSalle Site, Marseilles, IL

Inspection Conducted: March 29-31, April 1-3, 1982

Inspectors:	M. A. Ring	4/16/82
	R. D. Walker for L. A. Reyes	4/21/82
	R. D. Walker for N. Chrissotimos	4/21/82
	for F. A. Maura	4/21/82
	for I. N. Jackiw	4/19/82
	R. D. ~~Lanksbury~~	4/19/82
	R. D. Walker for P. T. Gwynn	4/21/82
	R. D. Walker for M. Jordan	4/21/82
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Approved By:	R. D. Walker, Chief Reactor Projects Section 1C	4-21/82

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Inspection Summary

Inspection on March 29-31, April 1-3, 1982 (Report No. 50-373/82-18(DPRP))
Areas Inspected: Routine, resident inspection, preoperational testing and management meeting. The inspection consisted of follow up on previous inspection findings, preoperational test results review, preoperational test results verification, independent inspection efforts (Installation of Startup Sources and Standby Liquid Control Inadvertent System Actuation), review of Preoperational Test Program System Deficiencies, review of Cable Tray Inspection Data, follow up allegations, follow up on NRC identified items and a management meeting held on April 2, 1982. The inspection involved a total of 384 inspector-hours onsite by 10 NRC inspectors including 63 inspector-hours onsite during off-shifts.
Results: No items of noncompliance or deviations were identified.

DETAILS

1. Persons Contacted

CECO

B. B. Stephenson, LaSalle County Station Project Manager b
R. H. Holyoak, Station Superintendent a
R. D. Kyrouac, Quality Assurance Engineer b
G. J. Diederich, Station Operating Assistant Superintendent b
R. D. Bishop, Administrative & Support Services Assistant Superintendent a, b
J. M. Marshall, Operating Engineer
W. Huntington, Assistant Technical Staff Supervisor
H. J. Hentschel, Assistant Technical Staff Supervisor
F. Lawless, Rad-Chem Supervisor
E. E. Spitzner, Startup Coordinator
E. Stevak, Quality Assurance
L. W. Duchek, Project Management Staff
J. Bowers, Onsite Nuclear Safety Engineering Group (ONSEG)
T. Borzym, Security Administrator b
T. E. Quaka, Site Quality Assurance Superintendent b
T. E. Watts, Project Engineer
J. W. Gieseker, Assistant to Site Construction Superintendent
D. L. Shamblin, Staff Assistant Project Manager's Office b
B. R. Shelton, Project Engineering Manager b
L. O. DelGeorge, Director of Nuclear Licensing b
B. Lee Jr., Vice President b
J. Renwick, Technical Staff Supervisor a, b

The inspector also interviewed other licensee employees including members of the technical, operating, and construction staff, as well as certain licensee contractor employees.

a. Denotes persons present at management interview conducted at the close of the inspection period.

b. Denotes persons present at the Management Meeting held at the LaSalle County Nuclear Power Station on April 2, 1982.

2. Follow Up On Previous Inspection Findings

(Open) Unresolved Item (373/79-34-01): Apparent discrepancies of four extra cables in tray 1062B and eight extra cables in tray 250C, which were not listed on the Cable Pan Loading Report (CIS) dated August 24, 1979. (Reference IE Reports 79-34, 79-39, and 81-32).

During this inspection (January 18, 1982), the Region III inspector reviewed Field Change Request No. 4916 dated October 30, 1979 (closed February 19, 1980) and the installation records for cables 1RF055, 1RE036, and 1RH093. The installation records had been correctly marked to include tray node 1062B in the respective routings, as required by FCR No. 4916. The installation record for cable 1FP389 indicated an installation date of August 16, 1979. The licensee correctly identified in IE Report 79-39 that cable 1FP389 had not been installed in time to be included in the August 24, 1979 CIS Report. The CIS Report, dated December 30, 1981, included cable 1FP389 in the loading of tray node 1062B.

On January 18, 1982, the licensee also provided FCR No. 9020, dated December 14, 1981, which indicated a current review of tray 1062B had been performed. FCR No. 9020 identified five additional discrepancies between the CIS Report and the actual routing of the respective cables. Tray node 1062B was involved in four of the discrepancies. All of the discrepancies were properly controlled. The Design Index number identified in CIS Report, dated December 30, 1981 for tray node 1062B was 0.92, which indicated that the thermal and mass loadings were within the design limits.

During this inspection January 18, 1982, the Region III inspector reviewed the installation records for cables 1RE087, 1AN204, 1AN021, 1AN026, 1AN035, 1AN037, and 1AN042 and the CIS Report, dated December 30, 1981, for tray node 250C. The records indicated as stated by the licensee in IE Report 79-39 that all of above cables except 1AN026 were replaced or deleted before installation. A new cable 1AN026 had been installed and the old cable 1AN026 had not yet been removed. On January 20, 1982, the licensee stated that the old cable 1AN026 would be removed immediately. (This one abandoned cable would have essentially no affect on the tray loading). On January 18, 1982, the Region III inspector counted the cables, routed through tray node 250C, at tray riser R137. Exclusively, the same cables were routed through both points. The Region III inspector counted 236 cables. The CIS Report, dated December 30, 1981, identified 235 cables designed for installation in tray node 250C. FCR No. 9020, dated December 24, 1981, indicated that a spare cable (1ZZ087) had been incorrectly identified on the December 30, 1981 CIS Report as being routed through tray node 250C. Therefore, only 234 cables were designed to be in tray node 250C. The licensee provided memo No. DA-331, dated December 24, 1981, which indicated that two extra cables (duplication of 1AN034 and 1AN036) were installed in tray 250C (riser R137) and would have to be removed. Thus, all cables in tray 250C were essentially accounted for.

On February 23, 1982, the licensee provided the Region III inspector a summary of a surveillance program performed by the licensee to determine cable identification and routing discrepancies. The program involved 54 cable tray risers and 2992 cables including (safety related and associated) in the auxiliary (including diesel generator rooms) and reactor buildings. 132 discrepancies were identified. The most significant problem was failure to remove abandoned cables which were not fire retardant. The summary indicated that one of the discrepancies contributed to adverse thermal or mass loading conditions. Based on the surveillance program there appears to be no adverse affect of non-inspected associated cables on raceway loading.

On February 23, 1982, the Architect Engineer (AE), Sargent & Lundy (S&L), provided a summary of calculations and a graph representing the physical loading of all seven safety related control tray nodes having a Design Index (D.I.) greater than 1.0 (i.e., nodes 223C, 224C, 239C, 250C, 371B, 385B, and 386B). No node exceeded a D.I. of 1.4. A separate summary of calculations was provided which represented the physical loading of all eleven safety related power tray nodes having a D.I. greater than 1.0 (i.e., 185A, 365A, 366A, 885A, 1036A, 1037A, 1129R, 1161A, 2514R, 1609A, and 1612A). In no case was the physical loading requirement, as specified in the FSAR, exceeded.

Although a direct relationship does not exist between the design index and physical loading, the above calculations support the use of the design index program to identify when physical loading calculations should be performed.

Included in the summary of calculations for physical loadings, was tray node 250C. The load indicated on the summary was well below the maximum allowable for node 250C.

Though questioned in IE Reports 79-34, 79-39, and 81-32, neither the licensee nor the AE established design control measures to verify or check the adequacy of the load design for cable tray intersections; and control and instrument conduits. This item remains open pending verification of the load design of tray intersections and of conduits (instrument and control).

(Open) Unresolved Item (373/79-39-01): Measures established to control thermal loading of raceways. Reference IE Report 81-32. The LaSalle FSAR, Amendment 59, dated December 1981, relative to Section 8.3.3.1, has been revised to reflect that the cable capacities are based on approximately two-inch fill for four-inch power tray and approximately three-inch fill for six-inch instrumentation and control trays. This FSAR revision does not reflect the actual tray fill design criteria. Per Sargent & Lundy Project Instruction LS-14, Revision 1, the thermal loading of tray is based on a Design Index Program, which only verifies thermal loading of trays with a design index (D.I.) over 1.25 (i.e., 2.5 inches of fill) for a four-inch power tray.

Per P.I. LS-14, Revision 1, the physical (weight) loading of tray is not verified until the D.I. is over 1.4 (i.e., 4.2 inches of fill for a six-inch tray).

On February 23, 1982, the licensee stated that changes to the FSAR, which reflect the actual design bases for tray loading, have been completed in house and would be submitted as Amendment 60 or 61 to the FSAR.

On February 23, 1982, the Architect Engineer provided thermal calculations for the only two safety related power trays (1129A, D.I. = 1.44; and 1524R, D.I. = 1.64) which had a D.I. greater than 1.25. The calculations, dated February 16, 1982 and February 18, 1982 respectively, indicated the thermal loadings were well below allowable limits.

The AE provided a graph of the results of calculations for power trays with a D.I. greater than 1.0 and less than 1.25. The graph indicated the thermal loadings were well below allowable limits.

The AE provided specific information concerning derating of cables which are covered with fire barriers. The consideration given for cables in fire barriers appears to be adequately conservative.

The AE provided the summary of the results of an evaluation of conduit fill dated February 9, 1982 (documented on an S&L inter-office memorandum). The summary indicated that out of 250 safety related power conduits, eight were found to be overloaded, based on the criteria defined in S&L standard EDSB-10. EDSB-10 was derived from the National Electric Code (NEC). The NEC Chapter 9, Table 1, Note 5, indicates that triplexed cable could be considered as one cable. EDSB-10 requires a triplexed cable to be considered as three individual cables, which is more conservative. Two of the eight conduits were overloaded because they violated EDSB-10; they did not violate the NEC. The remaining six overloaded conduits, were essentially insignificant violations (i.e., the contained cables required 0.29 square inches of cross sectional area and the provided area was 0.28 square inches).

On February 23, 1982 the AE provided a summary of the results of an evaluation of co[illegible] sleeve (wall penetrations for cables) fill for sleeves located in the auxiliary building, the auxiliary to reactor building, and the reactor to offgas buildings. 13 out of 81 safety related sleeves exceeded the S&L design criteria (NEC Chapter 9, Table 4).

The S&L design criteria was more conservative than allowed by the NEC Chapter 9, Note 5. None of the power sleeve loadings exceeded the NEC Chapter 9, Note 5. The summary indicated that analyses were performed on all 13 sleeves that exceeded the design criteria. The thermal loading of all 13 sleeves was acceptable (per the summary).

This item remains open pending revision to the FSAR, Amendment 59 to reflect the actual tray fill criteria.

(Closed) Open Item (373/80-41-04): Establishment of an Independent Safety Engineering Group on site that meets the training and experience requirements defined in the SER. The inspector reviewed the qualifications of the four individuals assigned to this group and found that they meet the training and experience requirements defined in the SER. The inspector also determined that the Working Instructions for this group are not complete at this time. The licensee has committed to complete the development of these Working Instructions by January 1, 1983. The inspector will follow up on the completion of this commitment under Open Item (373/81-18-01).

(Closed) Unresolved Item (373/82-11-16): The volume of jumpers, lifted leads, out-of-service outages, caution tags, hold cards, etc. associated with running a preoperational and construction test program make it a safety concern that Unit #2's logs governing those items are combined with Unit #1's. Provisions must be made to separate control of these items by unit. The inspector reviewed the current equipment out-of-service, jumper and lifted lead logs and determined that these logs are separated by unit. In addition to the above logs used to control the status of safety related equipment, the licensee has implemented a degraded equipment log which keeps track of degraded equiqment required by Technical Specifications. The inspector considers the above actions to be adequate to close this unresolved item even though the equipment hold tags are not separated by unit.

(Open) Unresolved Item (373/82-11-14): The jumper and lifted lead controls used by the Preoperational Test/System Demonstration System Test Engineer are virtually unauditable by the people using these devices, the personnel responsible for control of the systems, or personnel responsible for auditing compliance with these controls. The inspector stated in Inspection Report 50-373/82-11 that the licensee would be required to complete action on this item by Unit #1 fuel load. The inspector followed up on this item and determined that the licensee verbally committed to include a list of procedure steps that remove and install jumpers or lifted leads in the Shift Engineer's copy of the Preoperational Test/System Demonstration procedure. They agreed to complete this list prior to commencement of each Unit #2 Preoperational Test/System Demonstration. For Unit #1 Preoperational Tests/System Demonstrations which remain incomplete at the time of Unit #1 fuel load, the licensee will complete this list for the portions of the test which are incomplete prior to continuation of Unit #1 Preoperational/System Demonstration testing after receipt of a Unit #1 license. In addition, the licensee committed to closing all action on this item prior to October 1, 1982. The inspector cautioned the licensee that because the current jumper and lifted lead logs are not separated by systems as well as by unit, he has some residual concerns that the Operating Department Shift Engineers will not be able to readily determine if an individual systems operability is effected by reviewing the outstanding jumper and lifted lead log. The licensee stated that it was their feeling that the volume of jumpers and lifted leads logged in a given jumper lifted lead log will be reduced by the actions taken to close Open Item 373/81-11-16 and the actions committed to in this open item. The licensee concludes that this reduction in volume of jumpers and lifted leads in the logs will adequately address these concerns. The inspector stated that he would address the final closure of this item as a Category 2 dated open item because no equipment necessary for fuel loading or operation of Unit #1 should be affected by the commitments addressed in this open item. The inspector further stated that if additional concerns with jumpers and lifted leads develop during the Unit #1 startup or operating phase, NRC RIII will pursue corrective action that would address separating the jumper and lifted lead logs by system.

(Open) Open Item (373/81-00-91): SER, Chapter 22, TMI Action Item II.D.3. The inspector verified that interim emergency procedures are available and provide symptomatic and detailed methods to aid the operator in identifying backup position indication on main steam relief valves. This completes the interim actions required by the TMI Task Action Plan. The licensee must install qualified position indication switches on the main steam relief valves prior to startup from the first refueling outage to close the remaining portions of this open item.

(Open) Open Item (373/81-00-97): TMI Action Item II.K.3.22. The inspector verified that procedures for the manual switch-over of the RCIC suction are presented in a way that brings out pertinent and fundamental points to accomplish objectives. This item remains open pending completion of TMI Action Plan Requirements II.K.3 Item 13, 15, 18, and 21 per schedule commitments defined in the Safety Evaluation Report.

Test Program Section Review

(Closed) Open Item (373/81-20-16): Diesel generator air start motor replacement and 10 CFR 50.55(e) Report No. 82-01, NRC Docket No.'s 50-373 and 50-374, diesel generator air start motor defects.

The inspector reviewed Commonwealth Edison Company's report to NRC Region III No. 82-01, dated January 27, 1982; Work Requests No. L12832 (for diesel generator 0), No. L12832 (D/G 2A), No. L12833 (D/G 1B), and No. L12866 (D/G 1A); all applicable receiving inspection reports; and interviewed the QC inspector who performed the receipt inspection and the cognizant test engineer. As a result, the inspector has determined that:

a. 20 new air start motors were received and inspected to ensure the housing port spacing dimensions were correct,

b. 16 were installed on diesel generators 0, 1A, 2A, and 1B and four remain as spares, and

c. the diesels were satisfactorily tested after the installation of the new motors.

(Closed) Noncompliance (373/82-10-04B): Administrative procedures are deficient in that the removal of fuses are not controlled. The inspector verified that LAP 240-3, has been revised (Revision 6) to control the removal of fuses except when such removal is controlled by the "out-of-service" procedure.

(Closed) Open Item (373/82-10-03): Reinspection of all MCC breakers supplying systems covered by 10 CFR 50, Appendix A to ensure thermal and magnetic overloads are set in accordance with engineering specifications. The inspector reviewed the results of the reinspection performed by the licensee of all MCC cubicles. Approximately 1300 cubicles involving over 2500 settings were inspected, and approximately 10% of the settings were found in error. While some of the differences can be attributed to the accuracy of the readings in many cases the differences were substantial and cannot be explained in the majority of the cases. The licensee plans to leave the settings where found and is requesting approval of changes to the Electric Service Order (ESO) from Sargent and Lundy. Where changes to settings must be lowered, the equipment will be retested to verify operability unless the thermal overloads are bypassed when the equipment is required to operate.

(Closed) Open Item (373/81-07-03): Test Procedure PT-IN-101 satisfying Section C.4 of Regulatory Guide 1.80 with respect to flow and temperature. The inspector reviewed Sections 10.5.G., 10.5.H, 10.6.N, and 10.6.0 of PT-IN-101 and verified that these sections verify by test that the instrument air systems will meet specifications, relating to flow and temperature as required by C.4 of Regulatory Guide 1.80.

(Closed) Unresolved Item (373/81-28-07): Adequacy of torque wrench method of verifying proper opening set points for vacuum breaker valves in PT-VP-101. The licensee changed the method of verifying proper opening set points for vacuum breaker valves from torque wrench to use of air pressure differential. The inspector reviewed this method as performed in LST 82-1 and as proposed for periodic surveillance in LTS 500-2, Drywell Suppression Pool Vacuum Breaker Valve Force Check and it is considered acceptable.

(Closed) Open Item (373/81-20-01): Diesel Generator "0" tests involving Unit #2 hardware deleted because equipment not available at the time PT-DG-101A was performed. The inspector reviewed those sections of PT-DG-201A which incorporated the deleted portions of PT-DG-101A for Diesel Generator "0" tests involving Unit #2 hardware and determined them to be completed satisfactorily.

(Closed) Noncompliance (373/81-28-01A): Failure to incorporate applicable design requirements into a preoperational test. The inspector verified that the licensee performed the required testing of the air sparging subsystem as required by the FSAR. It was noted during a review of the data that the licensee failed to meet the acceptance criteria of the test. The inspector discussed the test results and what effect it would have on Technical Specifications with NRR (A. Bournia and J. Wing) and was informed that the 0.42-0.44% spread in the samples has no effect on the licensee's Technical Specifications.

(Closed) Open Item (373/81-00-35): Verify adequacy of the design of safety related bus voltage levels and Open Item (373/82-10-01) ensuring that voltages are within design specifications. The inspector reviewed the results of PT-AP-103, Special Tests A.1 and A.2 and verified that the field measured Class 1E bus voltages down to the 120/208 level with the SAT loaded to at least 30% of its rating, at steady state condition and during the starting of a large Class 1E load (LPCS motor) and a non-Class 1E load (Service Water Pump 1A) were within 2% of the analytical results. The largest percent difference between the computed minimum voltage and the measured minimum voltage was 1.7%. When the results are compared to FSAR, Figure 8.2-4 (Amendment 48) it shows that at all times the running voltages will be within the rated range for the Class 1E equipment.

(Open) Open Item (373/81-20-08): Licensee does not conform to diesel generators reliability requirements of Regulatory Guide 1.108. The inspector reviewed the licensee's summary history of all diesel generators (0, 1A, 1B, and 2A) valid tests and failures for the period since the conclusion of their respective 23 starts reliability tests. The inspector is in agreement with the data presented by Commonwealth Edison Company to NRR by letter (C. W. Schroeder to A. Schwencer) dated March 25, 1982 which notes a total of 6 failures out of 79 valid tests. In addition, the inspector noted 4 additional cases which based on the available documentation would have to be classified as "failures", but which the licensee feels were caused by either operator error or other causes not defined as a failure.

3. Preoperational Test Results Review

Test Programs Section

a. PT-DG-101B, HPCS Diesel Generator

The inspector reviewed the results of PT-DG-101B against the FSAR, the SER and Regulatory Guides 1.68 and 1.9. The following problems were noted:

(1) The Division III, 125V DC battery bank consists of 60 cells which normally float at 135 volts. An equalizing charge on the bank requires a voltage of 140-141 volts, which exceeds the maximum voltage rating (137.5V) of the equipment energized by it. The licensee is trying to obtain concurrence from the General Electric Company to reduce the number of cells from 60 to 58. In the meantime, the plans were to transfer the Unit 1, Division III, 125V DC loads to the Unit 2, Division III, 125V bus whenever the Unit 1 battery requires an equalizing charge. The inspector stated that since the Unit 2, Division III, 125V DC system has not been preoperationally tested, the planned load transfers are not acceptable unless the components transferred are declared inoperable and the Technical Specification LCO action statements applied. During PT-DG-101B the licensee demonstrated a battery capacity of at least 102%. According to the licensee, the removal of two cells reduces the capacity by approximately 7%, therefore the demonstrated battery capacity for the 58 cells would be at least 95%. The proposed Technical Specifications allow plant operation with a demonstrated capacity of 80%. Verification of the licensee's position by the Engineering Analysis and confirmation of General Electric Company's concurrence as stated above are an open item (373/82-18-02) pending resolution by the licensee.

(2) Two discrepancies from the FSAR were identified by the licensee. One requires that the "low fuel oil level" local alarm be deleted from FSAR, Section 7.3.1.1.6, and the second requires a change to the operating engine water temperature range specified in Section 9.5.5.2. The licensee stated both changes will be submitted with Amendment 61.

b. PT-AP-103, Emergency Power Redundancy

The inspector reviewed the results of PT-AP-103 against the FSAR, the SER and Regulatory Guides 1.68 and 1.9. The following problems were noted:

(1) The Division 1 diesel generator failed to meet the minimum voltage and frequency requirements during the simulated LOCA with loss of offsite power. During the LPCS pump start, the voltage dropped to 2650 volts or 63.7% of nominal versus the 75% minimum stated in Regulatory Guide 1.9. The frequency dropped to 56.8HZ or 94.7% of nominal versus the 95% minimum.

During the second pump start (RHR A), the voltage and frequency dropped to 2990 volts (71.9%) and 57HZ (96.3%) respectively. Both voltage and frequency recovered in both cases well within the required time frame. It was also noted that the duration of the voltage dip below the minimum voltage requirement of 75% of nominal was approximately 0.5 seconds during the starts of the LPCS pump and the RHR pump.

(2) For the Division 2 diesel generator only the frequency dropped below 95% (56.6HZ or 94.3%) during the start of the first RHR pump and to (56.3HZ or 93.8%) during the start of the second RHR pump. In both cases, the frequency recovered well within the allowable time period. Open Item (373/81-20-17) remains open until either NRR waives the licensee's commitment to Regulatory Guide 1.9 on voltage and frequency, or the licensee can improve the performance of the diesel generator.

The inspector reviewed the results of PT-AP-103 and LST-82-107 regarding ECCS response time and found them to be within the FSAR and Technical Specification limits.

c. PT-VG-101, Standby Gas Treatment System

The inspector reviewed the results of PT-VG-101 against the FSAR, the SER and Regulatory Guide 1.68. No problems were identified.

Projects Section 1C

d. SD-TS-101, Technical Support Center Diesel Generator

The inspector reviewed the Test Summary and Evaluation for the Technical Support Center Diesel Generator SD-TS-101.

The inspector verified that appropriate engineering functions evaluated the test results and signified that the testing demonstrated system design requirements. Test results were compared to manufacturer and design criteria.

Personnel responsible for review and acceptance of test results, including off-site review, have documented their review.

Comments and corrective action were included in this review. The inspector determined that the licensee follow up on corrective actions has been performed. The inspector determined that adequate verification and evaluation of the test was accomplished by the licensee.

e. PT-MS-101C, Main Steam Safety Relief Valves and ADS

The inspector reviewed the test results of PT-MS-101C, the Automatic Depressurization System. The review included test changes, deficiencies, summary and evaluation, and verification that test results have been approved.

The inspector determined that test changes, deficiencies and the summary and evaluation portions of the test conducted by the station were in conformance with applicable regulations and commitments.

No items of noncompliance or deviations were identified.

4. Preoperational Test Results Verification

The inspectors verified that the following Preoperational Test results appear to meet the requirements of the FSAR and the SER and were reviewed and approved by licensee management in accordance with the licensee's QA Manual.

a. PT-VC-101, Control Room HVAC
b. PT-RP-101, Reactor Protection
c. PT-VX-101, Switchgear Heat Removal, Unit 1
d. PT-VX-201, Switchgear Heat Removal, Unit 2
e. PT-VE-101, Auxiliary Electric Room HVAC
f. PT-VG-101, Standby Gas Treatment System

The inspector also reviewed LST 81-103 and LST 82-39 which documents the re-tests associated with the replacement of the Unit 1, Division 1 battery charger. The original charger had been tested by PT-AP-102 which was previously reviewed. The re-test results described in LST 81-103 amd LST 82-39 were satisfactory and no items of noncompliance were noted.

5. Independent Inspection Efforts

a. Installation of Start-up Sources

The inspector witnessed the installation of start-up sources and reviewed the procedures associated with the installation.

The inspector determined that the implementation of the procedures were adequate and that the personnel involved in the installation were cognizant of their duties and responsibilities.

b. Standby Liquid Control System Inadvertent Actuation

While conducting a review of the control room logs, the inspector determined that during the conduct of portions of the Preoperational Test PT-SI-102, "Pipe Vibration Monitoring", on the Standby Liquid Control System, the squib valves were fired and the pump tripped when operated with both discharge paths isolated. This item is an unresolved item pending; 1) review by the inspector of the sequence of events that led to the actuation of the system and 2) the corrective actions taken by the licensee. (373/81-18-03)

No items of noncompliance or deviations were identified.

6. Review of Preoperational Test Program System Test Deficiencies

The inspectors reviewed the System Test Deficiencies for the following Preoperational Tests/System Demonstrations:

PT-AP-101, "Unit 1 AC Distribution"
PT-AP-201, "Unit 2 AC Distribution"
PT-AP-202, "Unit 2 DC Distribution"
PT-DO-101, "Unit 1 Diesel Oil System"
PT-MS-101A, "Main Steam Isolation Valve Leakage Control System"
PT-PC-103, "Primary Containment Isolation System"
PT-RD-101B, "Rod Sequence Control System"
PT-RI-101, "Reactor Core Isolation Cooling System"
PT-RP-102, "Remote Shutdown"
PT-VD-101, "Unit 1 Diesel Ventilation"
PT-VP-101, "Primary Containment Vent and Purge System"
PT-VP-202, "Unit 2 Post LOCA Hydrogen Control"
PT-VP-104, "Primary Containment Chilled Water"
PT-VW-101, "Rad Waste Area HVAC"
PT-VY-102, "Unit 1 Core Standby Cooling Systems Equipment Ventilation"
SD-CD-101, "Condensate and Condensate Boosters"
SD-CD-102, "Condenser and Auxiliaries"
SD-CW-101, "Circulating Water and Auxiliaries"
SD-CY-101, "Cycled Condensate"
SD-FW-102, "Feed Water Control"
SD-HD-101A, "Feed Water Heater Drains"
SD-HD-101B, "Main Turbine Moisture Separator and Reheater"
S-PS-101, "Process Sampling System"
SD-RT-101, "Reactor Water Cleanup System"
SD-SA-101, "Service Air System"
SD-WE-101A, "Waste Collector System"
SD-WE-101B, "Floor Drain Reprocessing and Disposal System"
SD-WE-101C, "Laundry Equipment and Floor Drain Reprocessing System"
SD-WE-101D, "Chemical Waste System"
SD-WE-101E, "Station Equipment and Floor Drain System"
SD-WR-101, "Reactor Building Closed Cooling Water System"
SD-WS-101, "Service Water and Auxiliaries"
PT-MS-101C, "Main Steam Relief Valve and Automatic Depressurization System"
PT-VL-101, "Laboratory HVAC"

SD-WX-101, "Solid Radwaste System"
PT-CM-102, "Unit 1 Post LOCA Containment Monitoring"
PT-RD-101A, "Reactor Manual Control System"
PT-VY-101, "Unit 1 Core Standby Cooling Water System Equipment Cooling Water"
PT-AR-101, "Area Radiation Monitors"
PT-AP-102, "Unit 1 DC Distribution"
PT-RP-101, "Reactor Protection System"
PT-FR-101, "Fuel Handling Equipment"
PT-HP-101, "High Pressure Core Spray System"
PT-LD-101, "Leak Detection System"
PT-LP-101, "Low Pressure Core Spray"
PT-NB-101, "Nuclear Boiler System"
PT-RD-101A, "Rod Drive Control and Rod Position Indication System"
PT-RD-102, "Control Rod Drive Hydraulics"
PT-RH-101, "Residual Heat Removal System"
PT-SC-101, "Standby Liquid Control System"
PT-VD-201, "Unit 2 Diesel Ventilation"
PT-VP-103, "Primary Containment HVAC"
PT-VR-101, "Reactor Building HVAC"
SD-EH-101A, "Turbine Electro Hydraulic Control System"
PT-VG-101, "Standby Gas Treatment System"
SD-TS-101, "Technical Support Center Diesel Generator"
SD-CQ-101, "Communications System"
SD-CX-101, "Process Computer"
SD-FP-101A, "Unit 1 Fire Protection System-Water"
SD-FP-101B, "Unit 1 Fire Protection System-CO2"
PT-PR-101, "Process Radiation Monitoring"
PT-RR-101, "Reactor Recirculation System"
PT-SI-101, "Seismic Instrumentation"
PT-VE-101, "Auxiliary Electric Equipment Room Ventilation"
PT-VP-102, "Unit 1 Post LOCA Hydrogen Control"
PT-VC-101, "Control Room HVAC"
PT-VX-101, "Unit 1 Switchgear Heat Removal System"
SD-CM-101, "High Range Containment Radiation Monitors"
SD-VS-102, "Technical Support Center Ventilation"
SD-FP-201A, "Unit 2 Fire Protection System-Water"
SD-FP-201B, "Unit 2 Fire Protection System-CO2"
PT-VX-201, "Unit 2 Switchgear Heat Removal System"
PT-VY-201, "Unit 2 Core Standby Cooling Water System Equipment Cooling Water"
PT-MS-101B, "Main Steam Isolation Valve and Main Steam Instrumentation"

The inspectors reviewed the open deficiencies in these system tests to assure that they are properly categorized. The inspectors also reviewed a sample of the closed deficiencies to assure that they have been properly closed.

The inspectors determined that the open deficiencies in these system tests are being properly classified and that the licensee is correctly closing deficiencies found during the Preoperational Test Program.

This review closes the following NRC Region III Inspection Report Open Items:

(373/82-10-05), (373/82-10-06), (373/82-10-07), (373/82-10-10), (373/82-10-11), (373/82-10-12), (373/82-10-13), (373/82-10-14), (373/81-28-02), (373/81-28-03), (373/81-28-04), (373/81-28-06), (373/81-28-08), (373/81-28-09), (373/81-28-11), (373/81-28-12), (373/81-28-13), (373/81-28-14), (373/81-28-15), (373/81-28-16), (373/81-28-18), (373/81-28-19), (373/81-40-02), (373/81-40-03), (373/81-43-04), (373/81-43-07), (373/82-45-01), (373/81-18-06), (373/81-18-07), (373/81-43-08), (373/81-18-09), (373/81-18-10), (373/81-18-11), (373/82-04-01), (373/82-04-02), and (373/82-04-03).

During the review of these deficiencies, the inspector noted that Deficiency No. 980 on the Residual Heat Removal System Preoperational Test (PT-RH-101) documented vibration problems on RHR pump 1B and Deficiency No. 331 on the Low Pressure Core Spray System Preoperational Test (PT-LP-101) documented vibration problems on LPCS pump 1A. The inspector determined that these pumps had both been removed for repair. The inspector reviewed the problems with these pumps and the work being done on them to correct these problems and stated that he would be satisfied with the licensee's corrective action only if each of the pumps received a test that meets the requirements of Section XI of the ASME Boiler and Pressure Vessel Code and that the test is at least 200 hours in duration. The test for RHR pump 1B will be followed under Open Item (373/82-10-15) and the test for LPCS pump 1A will be followed under Open Item (373/82-18-04).

No items of noncompliance or deviations were identified.

7. Follow Up on NRC Identified Items

There were three items to follow up this inspection period.

a. General Electric Corporation report to the Tennessee Valley Authority (TVA) regarding an error in the fabrication drawing of an orifice in the RHR line to the suppression pool for the Hartsville site.

   The inspector contacted the Region II Section Chief cognizant of the Hartsville site to clarify the information available. The fabrication of the orifice as per the drawing error would have resulted in inadequate flow in the minimum flow line to the suppression pool. The inspector's review of the preoperational test results concluded that the minimum flow lines met the acceptance criteria.

b. International Instruments Division, Sigma Instruments, Inc. report to Southern California Edison Company regarding erroneous readings in Sigma Lumigraph Indicators (Model 9270) in use at LaSalle. The inspector forwarded the above mentioned report to the licensee for review. The licensee's review indicated there are no Sigma Lumigraph Indicators (Model 9270) in use at LaSalle.

c. Arkansas Nuclear One, Unit 1 LER 79-036 regarding the seismic qualification of the emergency diesel generator high speed differential relays in the standby mode. The unqualified relays are General Electric Model 12 CFD12B1A.

The inspector forwarded the above mentioned report to the licensee for review. The licensee's review indicated that the relays used at LaSalle are not of the above mentioned model and that seismic qualification documentation indicates the relays in use at LaSalle were tested in both the operating and standby mode.

No items of noncompliance or deviations were identified.

8. Review of Cable Tray Inspection Data

During the summer months of 1981, several fires occurred in cable trays at LaSalle County Nuclear Station. These fires were reported to the NRC Region III office via daily reports and the investigation of the nature and cause of these fires was undertaken by the Federal Bureau of Investigation at the request of NRC Region III. The inspector received a verbal commitment from licensee on-site management at the time of the fires to perform walkdowns of safety related cable trays to assure that no insulation damage, as a result of an undetected fire, existed. This walkdown of the safety related cable trays was to be completed prior to fuel load. During this inspection period, the inspector reviewed cable tray cleaning and inspection reports and determined that:

Cable tray cleaning and inspection was started on a full-time basis on July 15, 1981, under the supervision of a full-time cable tray inspection coordinator. Since starting these inspections, the tray systems in Units 1 and 2 have been cleaned and inspected approximately every 2 weeks. The number of men involved ranged from about ten (10) to the present crew size of four (4).

During the cleaning and inspection process, some cables were found to be marked by welding or burning operations. These cables were identified and repaired, if necessary. The repairs were made by either taping the damaged area or installing new cable. Each repair scheme used was decided upon a case by case basis in accordance with the contractor's NCR procedure. There are no cases where cable damage resulted from man-made cable tray fires. The inspector believes this action to be adequate corrective action and has no further concerns in this area.

No items of noncompliance or deviations were identified.

9. Follow Up on Allegations

Allegations

On February 3, 1982, Region III received allegations from an individual who was previously employed at the LaSalle Site by CECO in a non-welding capacity. The individual observed what he believed to be deficient welds in Unit 1 on the CRD housing. The man hole cover was welded in place December 11, 1981 preventing access to the area. The following allegations were made:

Allegation 1.

. Undercut and possible below minimum wall thickness of CRD housing reduced by grinding approximately 1/32" deep in two or three areas approximately 1" long each.

Allegation 2.

. Arc strikes on or near draw bead welds on CRD housing.

Allegation 3.

. Starts and stops on CRD housing not ground on draw bead welds.

Allegation 4.

. No identification numbers stamped near the above items of concern. The results of the Region III inspection performed as a reponse to these allegations is as stated below:

Allegation 1 (Undercut and Reduced Wall Thickness on CRD Housing)

ASME Section III, 1974 Edition, Summer 1975 Addenda, Paragraph NB-4424(c) states that undercuts shall not exceed 1/32" deep.

The inspector reviewed GE Drawing #197R616, Reactor Assembly and GE Drawing #922D124, CRD Housing. The CRD housing above the stub tube is not part of the pressure boundry. The outside diameter (OD) of the housing is 5.990" and the inside diameter (ID) is 4.845" which is 0.087" thicker than the area under the stub tube. The housing under the stub tube has a pressure up to 1250 psi.

The stub tube weld to the CRD housing is a J Groove, partial penetration weld. ASME Section III, 1974 Edition, Summer 1975 Addenda requires a liquid penetrant examination (PT). General Electric (GE) also requires an ultrasonic examination (UT) for defects and weld length information in accordance with GE specifications. All the welds are blended smooth. The inspector reviewed the following UT, PT and visual examinations (VT) procedures that were used. Also, NOE certifications were reviewed of two individuals in accordance with SNT-TC-1A, 1975 Edition.

. RCI, Ultrasonic Examination of Control Rod Drive Housing to Stub Tube Weldments, RCI-UE-1, Revision 3.

. RCI, Visual Examination Procedure VE-1, Revision 2.

- RCI, Liquid Penetrant Examination PE-1, Revision 3.

The inspector also reviewed RCI, Welding Procedure of CRD Housing to Stub Tube, WP-5 and 2 welder certifications.

The inspector reviewed several Production Weld QC Data Sheets of the CRD Housing to Stub Tube welds, that included final PT, UT, & VT, these were also accepted by the authorized nuclear inspector (ANI).

The following CECO QA Surveillance Reports were reviewed:

- Several inprocess of welding surveillances

- A CECO inspector noted undercut on the CRD housing to stub tube. This was evaluated and found to be acceptable in Audit Report #1-79-30

- Cleaning of stub tube area

- Weld plates removed by grinding from CRD housing

- Welding CRD housing to stub tube

Allegations 2 and 3 (Arc Strikes and Starts and Stops Not Ground)

Final inspection was made by CECO, GE, and RCI just before December 11, 1981, when the man hole cover was welded in place.

The inspector reviewed the RCI Procedure for Correcting Alignment by Draw Welding Reactor Internals Installation, #DB W-1, Revision 2, and Draw Bead Welding QC Data Sheets, which included VT and PT performed by Level II NDE personnel certified in accordance with SNT-TC-1A. VT and PT cannot accept arc strikes or unacceptable starts and stops.

The draw beads were welded with ER 308L, welding rod, the draw beads are approximatley 3/8" wide and 1/16" high and were left in the as welded condition.

The following are several more CECO QA Surveillance Reports reviewed:

- Draw bead welding on CRD housing (GTAW)

- PT of draw bead welds on surface of CRC housing

- Final PT of CRD housing

Allegation 4 (No Identification Numbers)

ASME Section III, 1974 Edition, Summer 1975 Addenda, Paragraph NB-4322.1 states that a record of permanent welded joints in a component and of the welders used in making each of the joints shall be recorded or stamped.

The inspectors observed that the records states that the stamp numbers of all welders involved in the welding.

No items of noncompliance or deviations were identified.

10. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, items of noncompliance, or deviations. An unresolved item disclosed during this inspection is discussed in Paragraph 5.

11. Management Exit Interview

The inspector met with the licensee representatives (denoted in Paragraph 1) at the conclusion of the inspection and summarized the scope and findings of the inspection.

12. Management Meeting Held At The LaSalle County Nuclear Power Station On April 2, 1982

a. Attendance

CECO

Those denoted in Paragraph 1.

NRC

A. Davis, Deputy Regional Administrator
R. L. Spessard, Director, Division of Project and Resident Programs
R. C. Knop, Chief, Reactor Projects Branch 1
A. Bournia, Licensing Project Manager
R. A. Purple, Deputy Director, Division of Licensing
J. Creed, Chief, Safeguards Section
I. N. Jackiw, Chief, Test Program Section
R. D. Walker, Chief, Projects Section 1C
N. Chrissotimos, Senior Resident Inspector
W. Little, Chief, Engineering Inspection Branch
L. A. Reyes, Senior Resident Inspector
Members of the Public

b. Meeting Summary

This meeting was held between Commonwealth Edison Company (CECO) and NRC Region III representatives to review the substantive issues that are germane to the NRC Region III review with respect to license issuance. The issues identified were as follows:

(1) NRC Region III took the position that the LaSalle County Nuclear Power Station Unit 1 appears to be licenseable within the next one to four weeks.

(2) NRC Region III will maintain the license related document in a continuous update status.

(3) NRC Region III requested a pump vibration monitoring program of approximately 200 hours in duration for the recently modified RHR pump 1B and LPCS pump 1A.

(4) The current status of deficiencies was addressed by both CECO and Region III.

(5) Region III recommended a memo be issued to all site personnel addressing the change from a construction status to operational status and the importance of procedure adherence during the conduct of all activities.

(6) Region III informed CECO that a final evaluation of the operational readiness of the security program could not be performed until all security related activities are conducted as they will be implemented upon issuance of the license.

During the meeting, it was agreed that a formal meeting may not be necessary to issue the license and that a weekly assessment of Operational Readiness will be made by Region III.