U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 84-26 / 84-09

Docket No: 50-352 / 50-353

License No. CPPR-106 / CPPR-107 Priority - Category B / A

Licensee: Philadelphia Electric Company
2301 Market Street
Philadelphia, Pennsylvania 19101

Facility Name: Limerick Generating Station, Unit 1

Inspection at: Limerick, Pa.

Inspection Conducted: June 1 - 30, 1984

Inspectors:

S. K. Chaudhary, Senior Resident Inspector	7/11/84 Date
J. T. Wiggins, Senior Resident Inspector	7/9/84 Date
W. Baunack, Project Engineer	7/13/84 Date
S. Reynolds, Reactor Engineer	7/13/84 Date
D. Vito, Reactor Engineer	7/13/84 Date

Approved by: R. M. Gallo, Chief, Reactor Projects Section 2A — 7/24/84 Date

Inspection Summary: Combined Inspection Report for Inspection Conducted June 1 - 30, 1984. (Report Nos. 50-352/84-26; 50-353/84-09)

Areas Inspected: Routine inspections by the resident inspectors and region-based inspectors of: followup on outstanding inspection items; followup on construction deficiency reports; TMI action plan followup; preoperational test procedure review and test witnessing; calibration of the primary containment vacuum relief valve position indication system; and recirculation valve indication. The inspection involved 135 hours for Unit 1, of which 28 hours were by the regional inspectors, and 5 hours for Unit 2.

Results: Two violations were identified: inadequate test program implementation (para. 6); inadequate calibration procedure (para. 7). In addition, an indication found in the B reactor recirculation pump suction valve was reviewed and found not to be a crack. The test program violation is particularly significant because of the current pace of preoperational activities and because of the importance to safety of the systems involved. Increased licensee attention to this matter is warranted.

DETAILS

1. Persons Contacted

Philadelphia Electric Company

J. M. Corcoran, Field QA Branch Head
R. Scott, Construction Engineer
G. Leitch, Station Superintendent
J. Spencer, Director, Start-up
J. Molito, Field Engineer

Bechtel Power Corporation

W. McCullough, Project Start-up Engineer
R. Bulchis, Resident Project Engineer

General Electric Company

R. Ballou, Start-up Operations

2. Followup on Outstanding Inspection Items

(Closed) Unresolved Item 50-352/81-17-04: Licensee to revise the FSAR to make it agree with Bechtel Specification E-1412 for wire separation in engineered safeguard and reactor protection system equipment. The inspector compared the results of the licensee's design verification test program for PGCC wiring, as discussed in a 1/8/82 letter from the licensee to GE, to the current revision of section 8.1.6.1.14 of the FSAR. The FSAR accurately reflects the separation criteria discussed in the letter.

(Closed) Follow Item 50-352/83-23-02: NRC comments on the rod worth minimizer preoperational test procedure. The inspector reviewed Test Change Notices 2 and 3 which implemented changes in the test procedure P56.1B. These TCN's acceptably addressed NRC's concerns.

(Closed) Unresolved Item 50-352/84-06-01: Generic evaluation of material problems in the scram discharge float switches. This item is discussed further in the report under CDR 84-00-05.

(Closed) Violation 50-352/84-19-04: Failure to properly review three Startup Nonconformance Reports (NCR) for reportability to the NRC. The inspector noted that the licensee reviewed the three NCR's and determined that the deficient condition associated with the ITT electrohydraulic operators for ventilation system dampers was reportable. Consequently the licensee filed a construction deficiency report on May 25, 1984. Further, the inspector verified that Quality Assurance Department Procedure 27.1 had been revised effective 5/84 to address evaluation of identified Startup NCR trends for reportability to the NRC.

(Closed) Violation 50-352/84-19-05: Failure to provide an appropriate procedure to address identification, reporting and correction of nonconforming conditions associated with facility buildings, areas and rooms. The licensee revised Startup Administrative Procedure AD1.2 to require that discrepant facility conditions be entered onto the Facility Punchlist by working through the Bechtel Facility Completion group. Startup Training Bulletins 25 and 28 were issued to inform the Startup staff of the currently implemented process. Further, because PECO Construction is responsible for facility turnover, Construction Division procedure CPL-9 was issued to define responsibilities for facility controls after turnover.

(Open) Violation 50-352/84-19-02: Failure to correct the nuclear instrumentation P & ID M42 to show changes made by Design Change Package (DCP) 232. The licensee indicated that an Interim Drawing Change Notice (DCN) to correct P & ID M42 and reviewed other DCP's for similar problems. The inspector reviewed IDCN 002 to M42 dated 5/11/84. This IDCN only partially corrected the P & ID, showing the two new level-indicating switches for the Level 8 HPCI trip LIS IN693 D & H, but not the LIS IN692 D & H for the Level 2 actuation. However, the inspector noted that the responsible system startup engineer had processed a Startup Change Notice (SCN) to fully correct M42. The SCN is currently in Bechtel engineering review. This item will remain open pending the correct revision of M42. The inspector also noted that Bechtel Engineering had reviewed 7 additional General Electric DCP's (Nos. 51, 52, 53, 54, 55, 115 and 176) and determined no problems existed.

(Closed) Inspector Follow Item 50-352/78-11-03 and 50-353/78-07-03: Soil reclamation in the spoil area east and west of Long View Road incomplete. The spoil area east and west of Long View Road was inspected to verify the area has been landscaped and reseeded. Some spaces are being used for equipment storage. Most is being restored with native vegetation.

(Closed) Unresolved Item 50-352/81-08-03: NRC to review resolution of Field Deviation Disposition Request (FDDR) No.HH1-379. Finding Report N-269, FDDR HH1-379-R1 and FDDR HH1-405 were reviewed. These documents evaluated the acceptance criteria and described the repair of the identified indications. The final resolution of this item is acceptable.

(Closed) Unresolved Item 50-352/81-10-01: Verify 10 CFR, Part 21 requirements were included on purchase requisitions for high density spent fuel storage racks and an emergency response facility data system. Also, that procedures have been established to assure implementation of Part 21 requirements. The inspector verified procurement documentation for the high density spent fuel storage racks and an emergency response facility data system contained the Part 21 requirements. The inspector also verified the Part 21 requirements are incorporated in Engineering and Research Department Procedures 4.4, Procedure for the Procurement of Specially Engineered Equipment, Materials, Services or Combination thereof with a Specification and 4.5, Procedure for Procurement of Nuclear Safety Related Items and Services by the Preliminary Requisition Method.

(Closed) Unresolved Item 50-352/84-14-02: Licensee to submit an update report to Significant Deficiency Report (SDR) No. 16. The licensee has on June 1, 1984 submitted an amended report to SDR No. 16 which corrects information previously submitted and indicates the actions which are being taken to assure unacceptable bolting material is not being used.

(Closed) Violation 50-352/83-17-04: Design basis for piping support not translated into specifications, and support incorrectly installed. Licensee records show this and all other supports incorporating seismic anchor movement have been analyzed and that the installed condition is acceptable. In addition, Field Job Rules M-12 and G-5 were verified to have been revised to indicate that information which had been transmitted via Resident Engineering Memorandum will now be included in interim revisions of ISO drawings.

(Closed) Unresolved Item 50-352/80-08-06: Licensee to evaluate nonconformance Reports (NCR) 3540 and 4171 dealing with installation of reactor pressure vessel horizontal stabilizers for reportability. The licensee's evaluation determined the conditions described in the NCR's were not reportable because they are the type of indications both inspectors and welders are on the alert for during the performance of their inspections. The licensee's evaluation is documented in Memorandum Qual 2-10-2 (SDR No. 23P).

(Closed) Unresolved Item 50-352/81-08-02: Copies of controlled drawings which have been "red lined" are not being marked as "reference" or "information only". Job Rules 8031-M-12, Revision 12, and 8031-G-32, Revision 8 were revised to incorporate a description of the control and distribution of xeroxed red-lined drawings.

(Closed) Unresolved Item 50-352/81-14-04: Copes and cutouts of structural shapes do not appear to meet the AISC ½" radius requirement. Verify that AISC requirements are applicable to pipe support structural elements and are implemented for installation activities. Quality Assurance Finding Report M-283 was prepared to resolve this matter. Resolution consisted of including minimum radius requirements in specification 8031-P-319, including inspection requirements in QCIR's, instructing Q.C.E's on the revised specification, and reinspecting all previously accepted hangers. This reinspection was documented on Field Inspection Report P-319-QCGI-5. Unacceptable results were documented on NCR # 5235 and evaluated. No corrective measures were found to be necessary.

(Closed) Unresolved Item 50-352/81-14-07 and 50-353/81-12-05: Certain items associated with the spray pond not included on licensee's "Q-List". Quality Assurance Finding Report M-279 was prepared to review this matter. Each of the items listed in the inspection report was addressed with regard to its safety significance. To resolve the issue descriptions of Q-listed and non-Q-listed parts of the spray pond were added to FSAR Table 3.2-1.

(Open) Program Weakness 50-352/82-16-01: PECO project management uses Construction Field Office Memorandum (CFOM) to transmit information to Bechtel. Nuclear safety-related information is sometimes transmitted using CFOM's . A weakness was identified in that no mechanism existed to verify that actions had been completed. A log of CFOM's is maintained. The maintenance of this log is directed by a letter dated December 3, 1982 and is intended to act as a followup mechanism for verifying completed actions. A review of this log and discussion with a licensee representative indicates the action completed column is not being maintained current. Other than the December 3, 1982 letter there are no other instructions detailing how the log is to be maintained current. This item remains open until a means is established for verifying completion of actions initiated by CFOM's.

(Closed) Violation 83-05-01: The conduit 1AI073 was found to have total bend of 385 degrees. The governing drawing/specification E-1406, Rev. 35 specified a maximum of 360 degrees of bends in any conduit between the pulling points. The design engineering clarified the specification requirements to show that the maximum bend requirements were for the ease of cable installation, and was more a recommendation than a technical necessity. The drawing/specification E-1406 was revised to incorporate this information. The action by the licensee in this matter is acceptable, therefore, this item is closed.

3. Plant Tour

Periodically during this inspection period, the inspectors toured the Unit 1 containment, reactor enclosure, control room, diesel generator enclosures; Unit 2 reactor enclosure, and the Spray Pond Pumphouse. The inspectors examined completed work and work in progress for indications of defective workmanship, nonconformance to technical requirements, and general adherence to project procedures. The inspectors reviewed drawings, specifications, procedures, and reports to assess the state of completion of the facility. Special emphasis was placed on visual examination of turned-over systems for as-installed conditions. The inspectors also witnessed portions of work in progress on the following items:

- Torqueing of safety relief valve M41-1F013K
- Preparation for preservice examination of pipe-to-valve weld for Valve no. HV-49-1F008

The inspectors examined the above work in progress to verify the adequacy of quality control, conformance to project requirement, requisite cleanliness, and proper measuring and test equipment for the work.

No violations were identified.

4. Followup on Construction Deficiency Reports

(Closed) CDR 81-00-10: Containment box beam design deficiencies. The licensee reported, on May 14, 1982 and June 14, 1982, that design deficiencies existed on 10 containment radial box beams as a result of Bechtel Engineering using higher stressallowables for these beams than those which would be identified by the current methodology for pipe whip load analyses. The box beams in question are used in the containment as supporting elements for pipe whip restraints, snubber loads, pipe supports, equipment and deck grating. There are 77 box beams per unit on four elevations in the drywell.

Based on reanalyses by Bechtel, 10 of the 77 Unit 1 beams required modifications to make them conform to current design criteria. These involve 9 beams on Elev. 296 for which the connections of the beams to the containment wall would have been overstressed and one beam on Elev. 272 for which its pin and supporting plate would have been overstressed. Quality Control Inspection Records (QCIR), nos. QCIR-C-934-C-63-1 (Civil and C-928-W-40 (Welding) were identified to track completion of corrective actions.

(Closed) CDR 83-00-08: Anchor-Darling motor-operated valve stem anti-rotation devices. The licensee initially reported problems with stem collar devices for these valves in a letter dated 11/28/83. As corrective action, the licensee initially determined that the problem would be solved by application of Loc Tite to the stem collar set screws to prevent their falling out of place. The NRC reviewed this proposed action in 1983 and raised questions regarding its adequacy (see Inspection Report 50-352/83-20).

Subsequently, further valve operation demonstrated to the licensee that the initial corrective action was inadequate. Supplemental corrective actions were determined to include drilling the valve stem to provide a better bearing surface for the set screws in addition to application of Loc Tite. These corrective actions appear to have solved the problem.

Thirty valves were affected by the design problem. The inspector was informed that rework on 23 is complete with the remaining 7 to be completed prior to fuel load.

(Closed) Potential CDR 84-00-05 and Unresolved Item 84-06-01: Failure of float level switches for the scram discharge volume. By telephone on 4/19/84, the licensee reported, as a potential CDR, a deficiency with some of the float level switches associated with the two scram discharge volumes (SDV) for each volume, 4 float-type level switches provide alarms and automatic actions for increasing water level in the SDV's. One float switch for each gives a high water level alarm, one float switch each implements a high water level rod block and two switches each (A and C; B and D) implement a reactor trip through the reactor protection system (RPS). In the case of the reactor trip logic, 2 differential pressure switches per volume provide diverse level monitoring as recommended by NRC and the BWR owner's group.

During preoperational testing, the licensee identified that two of the four float-type level switches (LSH 47-IN013A & B one per volume) failed to implement a reactor trip input to RPS when required. A Bechtel Nonconformance Report (NCR 9086) was prepared which documented that the switches failed as a result of crushing of the floats. Followup by Bechtel and the licensee determined that the material used in the floats that were crushed was 347 stainless steel instead of 17-7 PH as was assumed when the volumes were hydrostatically tested. The 347 SS floats did not withstand the 2100 psig hydrostatic test pressure applied at the site. As a result, Startup issued 3 Startup NCR's (S223M, S224M, S225M) which requested that the acceptability of the remaining six float switches be reviewed.

The inspector reviewed the disposition of the Bechtel NCR, the Startup NCRs and reviewed the Reportability Evaluation Form for Significant Deficiency Report (SDR) 136 prepared by PECO engineering associated with this problem. The Reportability Evaluation concluded the matter to be not reportable based upon document reviews which appeared to substantiate the integrity of the remaining 6 level switches. This evaluation, however, did not appear to consider the Bechtel NCR and its disposition in which General Electric decided to replace all four RPS-associated level switches because of inadequacy of the float material. As a result, the Reportability Evaluation appeared to ignore the failures which had occurred and to determine the matter as being not reportable based on the acceptability of the 6 remaining switches.

The inspector informed the licensee that the Reportability Evaluation for SDR-136 appeared inadequate because it failed to consider the failures which were experienced and the disposition of the Bechtel NCR. The inspector informed the licensee that this matter will remain unresolved pending the licensee's re-review of the matter. (50-352/84-26-01)

(Closed) CDR 81-00-04: The licensee reported that the HVAC subcontractor had fabricated and erected ductwork and duct hangers which were not in compliance with the design documents. The licensee also reported the corrective actions which had been and/or would be taken. Results of both Bechtel and PECO audits which were conducted to verify corrective action had been taken were reviewed. Results of those reviews showed corrective actions specified had been taken.

(Closed) CDR 80-00-12: Improper welding of reactor pressure vessel safe ends. Several instances were identified in which welding was not performed in accordance with procedures. Included were partial weld passes without feeding weld filler material, and in some cases, welding was followed by a spray or mist application of water to accelerate the cooling process and thereby facilitate alignment. This matter was referred to General Electric Company Nuclear Power System Division which determined that the practices described did not deviate from the requirements of the ASME boiler and pressure vessel code or the applicable GE-NEBG specifications. Since there was no deviation from technical requirements, and based on the final acceptance of the required nondestructive examinations, GE proposed accepting the welds. In the final report, dated April 14, 1981, the licensee also concluded there were no detrimental effects from these welding practices.

(Closed) CDR 84-00-07: Damaged shafts on ITT General Controls Actuators. In a letter dated June 22, 1984 the licensee indicated that, as a result of excessive lateral movement of the actuator shaft during actuator operation, the hydraulic system seals on the actuators were damaged. The licensee further indicated that all defective shafts were replaced by the vendor and the actuators for Unit 1 had been reinstalled. The vendor's representatives inspected the installation and identified that design modifications were required to prevent recurrence. The inspector verified that the modifications had been completed for Unit 1.

5. TMI Action Plan Followup

The inspector reviewed the licensee's actions taken in response to the following item from NUREG 0737:

(Closed) Item I.C.5 Procedures for Feedback of Operating Experience tc Plant Staff

This item was reviewed by NRR. The results of this review are documented in Supplement 1 to the Limerick SER which discusses the contents of a draft of procedure NGD-A-5, "Procedure for Review and Utilization of Operating Information". The procedure acceptably implements this TMI item.

The inspector verified that a procedure of the same title, but numbered NSS-I-4, had been formally issued by the licensee's corporate Nuclear Safety Section and had implemented the program described in NGD-A-5.

6. Preoperational Test Procedure Review and Test Witnessing

The inspector reviewed the following preoperational test procedures to verify their technical adequacy their conformance to administrative requirements, and to assure their implementation of the testing commitments documented in Section 14.2 of the FSAR:

1P30.1	Control Enclosure HVAC system
1P73.1	Containment Atmospheric Control System
1P59.1	Containment Isolation and Nuclear Steam Supply Shutoff System
1P100.4	Standby Diesel Generator Loading

Additionally, the inspector witnessed performance of portions of the following tests to verify that the Test Director was knowledgeable of the methods and purposes of each test and of the administrative requirements associated with preoperational testing (e.g. Test Exception control and Test Change control):

1P24.1	Standby Diesel Generators
1P32.1	Control Room HVAC system
1P32.2	Control Room Isolation and Purge system
1P100.4	Standby Diesel Generator Loading

Except for those findings discussed below, no violations were identified.

Findings

1P32.2 Control Room Isolation and Purge System

On 6/12/84, the inspector witnessed performance of parts of preoperational test 1P32.2 for the Control Room Isolation and Purge system. The inspector observed the Startup Engineers performing section 6.4.4.2 of the procedure, which dealt with the manual initiation of a chlorine isolation from Channel C.

Originally, section 6.4.4.2 was to be performed by first jumpering a relay in the control for valve HV-78-020C, the control room emergency fresh air supply, to simulate a high radiation condition in the control room HVAC inlet. The jumper should have caused the valve to open. Subsequent test steps would have implemented a manual control room chlorine isolation from Channel C. Acceptable operation would be achieved provided 1) control room HVAC valves successfully closed upon manual isolation, and 2) the chlorine isolation signal overrode the radiation isolation signal and closed HV-78-020C. Step 6.4.4.2(17) would then test system response as the chlorine isolation was reset.

The inspector observed that step 6.4.4.2(17)e indicated HV-78-020C should have opened upon reset of the chlorine isolation. However, when this step was performed, the valve remained closed. The inspector asked the Startup Engineers (SSE) to explain why HV-78-020C remained closed and a test exception was not being considered. HV-78-020C is normally closed, but opens on a radiation isolation signal to provide filtered fresh air to the control room. However, on a chlorine isolation, it remains closed because the emergency fresh air system cannot remove chlorine from the incoming air. The SSEs informed the inspector that a test change notice (TCN-27) had altered the method by which HV-78-020C was to be opened on a radiation isolation. In step 6.4.4.1(6), instead of using a jumper to simulate a radiation isolation signal to the valve, TCN-27 required that a full radiation isolation would be implemented manually. Further, due to the design of the control room isolation reset switch, resetting of the chlorine isolation also resulted in reset of the radiation isolation. Therefore, upon isolation reset, HV-78-020C should remain closed as was observed.

The inspector reviewed earlier similar testing on HV-78-020A. The inspector noted that similar results were achieved. As discussed, upon restoration of the A channel chlorine isolation in section 6.4.4.1, HV-78-020A remained closed because all isolation conditions were reset using the reset switch. However, the inspector noted that in 6.4.4.2(2)(e) the SSEs verified HV-78-020A closed on a high chlorine A channel manual isolation. Based on a review of the procedure and on discussions with the SSEs , HV-78-020A apparently had never been reopened prior to performance of step 6.4.4.2(2)(e), therefore the verification of its closure was invalid.

The inspector determined the problems encountered with testing HV-78-020A resulted from the SSEs not fully identifying all the effects caused by TCN's to the original test procedure. Following discussions with the SSEs and the Project Startup Engineer, the licensee stopped the test to reevaluate the situation.

The inspector informed the Project Startup Engineer that failure to fully control the performance of 1P32.2 constituted a violation of 10 CFR 50 Appendix B, Criteria XI. (50-352/84-26-02)

1P59.1 Containment Isolation and Nuclear Steam Supply Shutoff (NSSS) System

On June 18, 1984, the inspector reviewed a version of 1P59.1 which had been approved by the Test Review Board. The inspector compared the test method and the acceptance criteria to the method and criteria defined in the FSAR Test Abstract.

As a result of this comparison, the inspector identified one discrepancy. Acceptance criterion 2 in the FSAR Test Abstract states that the closure times for automatically actuated valves controlled by the containment isolation and NSSS systems would be demonstrated during this test. When the inspector compared the listing of containment isolation valves in FSAR Table 6.2-17 to the list of valves to be timed under 1P59.1 (and other procedures referenced in 1P59.1), IP5.9.1) 34 valves were not being timed. These included 10 valves for which closure times were listed in both the FSAR and the Limerick Draft Technical Specifications and 24 valves for which closure times were listed in the FSAR but not directly in the Draft Technical Specifications.

The inspector discussed the above discrepancy with the licensee's Startup representatives and with the Test Review Board Chairman. The licensee indicated that out of the 10 valves for which times are defined in both the FSAR and Technical Specifications, 8 are marked in the FSAR with either a single or double asterisk. The asterisk means that the time shown is critical to radiological release calculations.

The inspector reviewed the final approved version of 1P59.1, dated June 19, 1984 and noted that the same discrepant condition existed. Therefore, the inspector informed the licensee that, because the preoperational test program did not include all testing necessary to demonstrate the suitability of the containment isolation and NSSS system for service, the test program was in violation of 10 CFR 50, Appendix B, Criterion XI requirements. (50-352/84-26-03)

1P100.4 Standby Diesel Generator Loading

This procedure conducts the 24 hour full load test and the design basis accident loading and load shedding sequences for each diesel generator. The inspector reviewed the final approved version of 1P100.4 to assure it fulfilled the testing commitments described in the FSAR and the SER. Further, the inspector compared the diesel generator testing which had been performed in 1P24.1 and what was to be performed in 1P100.4 to the testing recommendations contained in section C.2.a of Regulatory Guide 1.108.

The inspector determined that 1P100.4 was a technically adequate test and its acceptance criteria agreed with those provided in section 14.2 of the FSAR. However, the inspector had some minor questions regarding the test, most of which were satisfactorily answered during a 6/18/84 meeting with the Startup Director, the responsible Startup Group Leaders, Startup Engineers and a Test Review Board co-chairman. Those questions which were answered and those remaining open are discussed below.

The inspector inquired as to which voltage, frequency and real and reactive power indicators would be designated as the official indicators during the test. The Startup representatives stated that the control room indicators would be used except that the startup and transient response of the diesel generators would be evaluated using a strip chart.

During the 24 hour full load test, each diesel generator would be loaded to its 100% design load capacity (2850 kw) for at least 22 hours and its 110% capacity (3135kw) for at least 2 hours. For both aspects of the test, reactive load would be kept at about 75% of real load; thus maintaining a 0.8 power factor. Note 4.2(9) of 1P100.4 provided the load tolerances as -0, +100kw and -600, +100 kvar. Further, this note permitted brief excursions beyond these ranges. However, the procedure failed to adequately define "brief excursions" in terms of measurable parameters such as kw and kvar versus time. The inspector expressed concern regarding the imprecise definition of "brief excursions", especially in light of the fact that data during the 100%/110% runs would only be taken every 2 hours. Startup representatives indicated that "brief excursions" implied short-time transient conditions which may occur during testing, but did not further amplify the definition. Finally, the Startup Director agreed to increase the data acquisition rate for kw, kvar, voltage and frequency to every 15 minutes during the 100%/110% tests. For any 15 minute reading below the ranges specified in Note 4.2(9), the load run would be extended a commensurate 15 minutes. The inspector accepted this commitment and Test Change Notice 5 was approved to implement it.

The inspector then questioned the acceptance criterion to be applied to the LOCA-sequence start of the emergency service water (ESW) pumps from each diesel generator bus. Under loss of offsite power (LOOP) conditions, Table 8.3-1 of the FSAR indicates a ESW pump would start 42 seconds after the diesel generator reenergized its associated 4160V bus. Step 6.4.2(9) of 1P100.4 indicated a 45 second criterion. The inspector reviewed Licensing Document Change Notice 534 which changed the loading sequence during LOCA-LOOP conditions. The inspector reviewed LDCN 534 which provided a new Table 8.3-1. The revised table indicated that the ESW pump would start 45 seconds after the diesel started. Further, it added the automatic start times for the control room chillers (at 177 seconds) and the reactor enclosure recirculation fans (at 193 seconds). The inspector noted that the times listed in the preoperational test for the start of these components differed from the times in the revised table.

Finally, with regard to conformance to the Regulatory Guide 1.108 positions on preoperational testing, the inspector had two questions.

First, with respect to the C.2.a(4) position on testing the diesel generator units during load shedding, the inspector noted that 1P100.4 tested the diesel generator response to loss of the largest load (the RHR pump) but did not test the diesel generator response to a complete loss of load. The responsible Startup Engineer indicated the complete loss of load was tested during 1P24.1. The inspector requested the test data from 1P24.1 for review. Subsequently, he was informed that only 2 of the 4 diesel generators had undergone this test, therefore, TCN 12 to 1P100.4 was added to perform the complete loss of load test for the 4 units.

Second, position C.2.a(8) requires the licensee to demonstrate that the capability of the diesel generator units to supply emergency power within the required time is not impaired during periodic surveillance testing. The inspector asked in which test procedure this demonstration would be performed. The responsible Startup Engineer indicated that IP 24.1 tested the diesel generator responses to a LOCA signal with offsite power available, occurring while the diesel generator units were supplying power to their respective 4160 v buses. However, the responses to a LOCA-LOOP, while the diesel generators were supplying power, were not tested in either 1P24.1 or 1P100.4 or scheduled to be tested in 1P100.1, Loss of Offsite Power.

The inspector informed the licensee's Startup representatives that because the test program associated with the diesel generators did not include all tests necessary to demonstrate their reliability, it was not in compliance with 10 CFR 50 Appendix B, Criterion XI requirements. (50-353/84-26-04)

7. Calibration of the Primary Containment Vacuum Relief Valve Position Indication System

The inspector reviewed the calibration records for the primary containment vacuum relief (PCVR) position indicating switches. This review was to assure these switches were calibrated to the sensitivity required to demonstrate that the potential steam bypass of the suppression pool through a partially open vacuum relief would be adequately indicated to plant operators.

In its response to FSAR question 480.7, the licensee stated that valve opening is detectable at a disc lift of 0.06 inches or greater above the valve seat. If all eight vacuum relief valves (2 in series on each of 4 downcomers) were open 0.06 inches, the corresponding bypass leakage area would be less than the 0.05 ft^2 assumed in the containment analyses.

FSAR section 9.4.5 described the valve position indicators as sets of redundant, plunger-type switches with a differential travel of 0.004 inches. This differential travel, when multiplied through the mechanical linkage to the valve disc, would be attained if the valve disc travelled 0.06 inches off its seat.

Based on the above, the inspector sought to verify that the calibration procedure for the position switches was such that the 0.06 inch travel distance at time of switch actuation was verified. A review of calibration data on the switches, ZS-57-137A-1/A-2 through D-1/D-2 showed that the required sentitivity was not attained. The records indicated only the open/closed indications were tested and the exact actuation points for the open/closed switches were not recorded or adjusted.

The inspector informed the Startup Director and the Lead Results and Test Engineer that the calibration procedure used for these position switches was inadequate. Failure to provide an adequate calibration procedure for the PCVR position indicating switches violated 10 CFR 50 Appendix B, Criterion V requirements. (50-352/84-26-05)

8. Visual Indication on the Internal Surface of Reactor Recirculation Valve

The licensee identified internal surface indications in the reactor coolant recirculation system valve B32-1F023B.

In inspection report 50-352/84-24, the inspector documented the results of his review of the radiograph reader sheets and the accompanying vendor and receipt documents associated with the B reactor recirculation pump suction valve B32-1F023B. There were no problems identified in the documents reviewed.

In response to a 5/29/84 letter from NRC Region I, the licensee conducted a visual inspection of the valve internals. Access to the valve was gained by entry into the suction line 28" pump via the reactor vessel. As a result, the licensee identified a circumferential indication at the weld joint between the valve seat ring and the valve body casting.

A region-based inspector also reviewed the document package for the valve. The valve body is cast stainless steel SA351, grade DF8M and the seat ring is centrifugally cast SA351, grade 3A (with high ferrite). The seat ring was welded to the valve body with E308L filler metal and the SMAW process. Discussions with GE NEBO (San Jose) indicated the seat was hardfaced with the GTAW process. Available data showed that the filler metal was Stellite 6 meeting MIL-R-17131A, Type R Co-Cr-A. The hardfacing is approximately 3/32" thick with a minimum thickness determined by (dilution) hardness requirements. GE NEBO stated that the ring to body weld penetrant test was done with a water-washable technique. The location of the indication is consistent with the layout of the weld area and the junction of the joint level on the valve body side of the seat ring to valve body weld.

Representatives of GE, Bechtel, PECO and the NRC reviewed photographs of the indication. The conclusion of the review was that the cause of the indication was a lack of weld metal sufficient to "clean-up" the weld area during post-weld machining. The indication was not a crack and was of a configuration such that no stress concentration was to be expected. The stress applied in service for the valve body to seat ring weld was determined to be negligible and the weld is not part of the valve's pressure boundary. Further, the materials involved are notch insensitive and the indication (surface irregularities) would not have an adverse effect on the valve's performance.

The inspector visually examined another valve, B32-2F031B, which was identical to the valve with the indication. The inside weld face (reported by GE to be a 45^{o} bevel on the valve body side and 20^{o} bevel on the ring side) was observed with minor round visual indications that would pass a water-washable penetrant test.

The NRC inspector concurred with the technical findings of the licensee and had no further questions regarding this matter.

9. Unresolved Items

Unresolved items are matters about which more information is necessary to ascertain whether they are violations, deviations, or acceptable items. Unresolved items are discussed in paragraph 4 of this inspection report.

10. Meeting on Preoperational Test Program Implementation

On June 20, 1984, during a tour of the Limerick facility, Mr. R. W. Starostecki, Director, Division of Project and Resident Programs, Region I, met with Mr. G. M. Leitch to discuss NRC-perceived weaknesses in the licensee's implementation of the Unit 1 preoperational test program. These weaknesses were considered to have resulted in the violations identified during this and previous reporting periods. Special emphasis was placed on the extent of involvement in program activities by the permanent PECO station staff.

On June 26, 1984, Mr. J. S. Kemper, Vice President Engineering and Research and Mr. M. J. Cooney, Manager Nuclear Production met onsite with Mr. Starostecki and Mr. H. B. Kister, Chief, Projects Branch 2, to describe those actions taken to strengthen the program. These actions will be evaluated during future inspections.

11. Acknowledgement of Licensee Response Letters

During this inspection period, licensee letters, dated June 18 and June 20, 1984 were received in response to NRC Region I correspondence. The first letter documented the results of the licensee's visual inspection of a recirculation system valve (see section 8.0 of this report for details). The second letter reported the corrective actions taken regarding the Notice of Violation issued with Inspection Report 50-352/84-19.

In both instances, the inspector determined the licensee's actions to have been acceptable. Routine inspection followup of the corrective actions in the June 20, 1984 letter will be documented in subsequent inspection reports.

12. Exit Meeting

The NRC resident inspectors discussed the issues and findings in this report throughout the inspection period and at an exit meeting held with Messrs. J. Corcoran and G. Leitch on June 29, 1984.