U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Report No.	89-23 89-31
Docket No.	50-352 50-353
License No.	NPF-39 NPF-85
Licensee:	Philadelphia Electric Company Correspondence Control Desk P.O. Box 195 Wayne, Pa 19087-0195
Facility Name:	Limerick Generating Station, Unit 1 and 2
Inspection Period:	November 21 - December 31, 1989
Inspectors:	T. J. Kenny, Senior Resident Inspector L. L. Scholl, Resident Inspector M. G. Evans, Resident Inspector D. T. Moy, Reactor Engineer

Approved by: Lawrence T. Doerflein 2/9/90
Lawrence T. Doerflein, Chief Date
Reactor Projects Section 2B

Summary: Routine daytime (237 hours) and backshift/holiday (26 hours) inspections by the resident inspectors consisting of (a) plant tours, (b) observations of maintenance and surveillance testing, (c) review of LERs and periodic reports, (d) review of operational events, (e) system walkdowns, and (f) observation and results review of power ascension activities on Unit 2.

During this inspection period:

- Unit 2 completed the power ascension test program with the exception of the warranty run. Major tests performed were the reactor feed pump trip and the main steam isolation valve closure from rated reactor power. The tests were performed well and no significant problems were observed. An unresolved item (50-353/89-31-01) was identified regarding the performance of a turbine trip test prior to the first refueling outage (Section 6.0).

- Charcoal bed leakage was discovered and repaired on the standby gas treatment system charcoal adsorber beds (Section 8.0).

- The PECo fitness for duty training programs were inspected (Section 3.1).

- A maintenance outage was completed on Unit 2 and the plant restarted (Section 2.1).

- The status of the PECo response to Generic Letter 88-20, Individual Plant Examinations for Severe Accident Vulnerabilities, was reviewed (Section 9.0).

DETAILS

1.0 Persons Contacted

Within this report period, interviews and discussions were conducted with members of PECo management and staff as necessary to support inspection activity.

2.0 Operational Safety Verification

The inspectors conducted routine entries into the protected areas of the plant, including the control room, reactor enclosure, fuel floor, and drywell (when access is possible). During the inspection, discussions were held with operators, health physics (HP) and Instrument and Control (I&C) technicians, mechanics, security personnel, supervisors and plant management. The inspections were conducted in accordance with NRC Inspection Procedure 71707 and affirmed PECo's commitments and compliance with 10 CFR, Technical Specifications, License Conditions and Administrative Procedures.

2.1 Inspector Comments and Findings (71707)

At the start of the inspection period on November 21, Unit 1 was operating at 100% power and Unit 2 was at approximately 95% power to support performance of Test Condition 6 power ascension testing.

At 8:40 p.m. on November 21, Unit 2 achieved 100% power operation for the first time.

On November 22 at 12:35 a.m., the Unit 2 reactor building was manually isolated when the reactor enclosure differential pressure indicator was found to be indicating downscale. The cause was found to be a loose electrical connection on a circuit board in the reactor building differential pressure sensing instrumentation. The standby gas treatment system started and operated as designed to maintain the required building differential pressure. The circuit board was reseated to obtain the proper electrical connection and the isolation was reset and the ventilation systems were returned to their normal lineup. The NRC was notified via the emergency notification system (ENS) per 10 CFR 50.72(b)(2)(iii). However, the event was later determined not to be reportable under 10 CFR 50.72 and the notification was withdrawn on November 27. The basis for the withdrawal was that the isolation was initiated to comply with the Technical Specification action statement and not due to a concern for the integrity of the secondary containment.

On November 23 at 4:53 a.m., the Unit 1 reactor water cleanup (RWCU) system outboard isolation valve closed due to a false high differential flow signal caused by instrumentation failure. This instrumentation is part of the Nuclear Steam Supply Shutoff System (NSSSS), an engineered safety feature, thus the event was reported to the NRC via the ENS as required by 10 CFR 50.72(b)(2)(ii). Licensee Event Report 1-89-058 was subsequently submitted as required by 10 CFR 50.73(a)(2)(iv). (See section 8.0)

On November 23, the Unit 2 'B' Reactor Feed Pump tripped due to a faulty circuit board in the vibration monitoring instrumentation. The feed pump trip caused the reactor recirculation pumps to run back so that reactor power was reduced to within the capacity of the remaining two operating feed pumps. All systems operated as designed during the transient. The vibration trip signal was bypassed, the 'B' reactor feed pump was returned to service and reactor power was returned to 100%. The faulty circuit board was subsequently replaced.

On November 27, the Unit 2 'C' reactor feed pump was tripped as part of startup test 2STP-23.5. All systems functioned as designed during the test and power was restored to 100% upon completion of the test. (see section 6.3)

On December 1, Unit 2 was shut down from 100% power by closing all main steam isolation valves in accordance with startup test procedure 2STP-25.3, "Full MSIV Isolation." (Refer to section 6.2 and 6.3 of this report for test result details.) The following two ENS notifications were made to the NRC as a result of the test performance:

1) The High Pressure Coolant Injection (HPCI) system auto initiated and injected on a low reactor water level condition. Although a HPCI initiation is expected as a result of the test induced transient, 10 CFR 50.72(b)(1)(iv) requires reporting any emergency core cooling system discharge into the reactor coolant system.

2) A four hour report was made when it appeared that control of the HPCI discharge valve to feedwater was lost, following the initial injection rendering the system inoperable. The followup investigation revealed that only the valve position indication was not fully functional and therefore HPCI would have injected as designed if called upon a second time. Based on this finding the notification was subsequently withdrawn.

Unit 2 was then placed in cold shutdown to support the planned maintenance outage.

On December 4, during the performance of the Unit 1 reactor protection system (RPS) main steam isolation valve closure functional test, the limit switch on the 'B' steam line inboard isolation valve failed to reset when the valve was returned to the full open position. The valve was confirmed to be in the full open position by the operation of limit switches that are independent of the limit switch that provides input to RPS. A temporary circuit alteration was implemented to remove the fuses in the logic circuit for the 'B' steam line to ensure that the RPS signal remains inserted. The RPS logic design is that a three of four steam line isolated condition will generate a full scram signal. The present configuration of the RPS system will still function to initiate a scram if two additional steam line isolation signals are generated. Maintenance activities will be performed when the plant conditions permit access to the valve.

On December 6, at 9:58 a.m., during the performance of Unit 1 surveillance test ST-2-026-618-1, "NSSSS-Reactor Enclosure Ventilation Exhaust Duct Radiation-High Division IA Functional Test," an instrumentation technician inadvertently touched a test jack jumper with a key chain which caused a short circuit to ground. The short circuit caused a fuse to blow which resulted in several system isolation signals. The fuse was replaced and the isolations reset within approximately 30 minutes. There was no effect on plant operation due to the occurrence. The NRC was notified via the ENS. The I&C technicians involved with the event were counseled to stress the importance of a higher level of attention to detail while performing work tasks.

On December 13, PECo discovered a potential bypass release path through the standby gas treatment system (SBGTS) charcoal filters. During routine inspection of the 'A' SBGTS charcoal filter, charcoal was found on the floor. Further investigation determined that the source of the charcoal was an opening in the retention screen due to failure of several spot welds. The level drop of charcoal within the charcoal filter created a bypass leakage path around the filter bed. The last inspection of the filters which involved testing of the bypass flow was conducted in February 1989 and the results were satisfactory. The NRC was informed via the ENS. The filter was repaired, charcoal reloaded and a satisfactory operational test performed. The system was declared operable on December 16. The 'B' charcoal filter was then inspected on December 18 and found to also have a small amount of charcoal loss. However, the extent of leakage was not severe and no bypass leakage path was created thus the 'B' train of SBGT remained operable. The

filter was repaired, charcoal reloaded and a satisfactory operational test performed. See section 5.0 for additional details.

On December 15, several channel checks on Unit 1 required by the plant Technical Specifications were missed. The operator performing the checks was interrupted to respond to a medical emergency and upon his return he recommenced taking readings at the wrong step. The missed checks were satisfactorily performed on the following shift. This occurrence constitutes a violation of the plant Technical Specifications; however, the inspectors have reviewed this event and determined that it satisfies the criteria for licensee identified violations as stated in 10 CFR 2 Appendix C, Section V.G.1 and as such a Notice of Violation will not be issued (NCV 50-352/89-23-01).

On December 21, a meeting was attended by PECo and NRC management to discuss the results of the Systematic Assessment of Licensee Performance (SALP) report which was issued on December 1, 1989.

On December 22, outage activities were completed on Unit 2 and the mode switch was placed in "Startup." Reactor criticality was achieved on December 23 at 6:19 a.m. and the main generator was synchronized to the electrical grid on December 24. Power was then gradually increased to approximately 75% where it was held until December 27 when power was reduced to 50% to support leak testing of the main condenser. The purpose of the power decrease was to minimize radiation exposure levels in the condenser area.

During the startup two problems were encountered with the 'B' reactor protection system power supply. On December 23, at 4:48 p.m., the Unit 2 'B' reactor protection system (RPS) power supply circuit breakers tripped due to an undervoltage signal. Reactor power at the time was less than 1%. The loss of the 'B' RPS bus resulted in a half-scram signal, tripping of both reactor recirculation pumps, loss of normal reactor enclosure ventilation, start of the SBGT system and several system isolations. Upon investigation no undervoltage condition was evident and the circuit breakers were reclosed. All systems were returned to their normal lineups. On December 24, at 4:28 p.m., with reactor power at 17%, the 'B' RPS breakers again tripped resulting in similar system responses. Again no sustained undervoltage condition could be detected and the breakers were reclosed and the systems were restored to normal. The 'B' RPS power supply transfer switch was shifted to select

the alternate power source until troubleshooting could determine if the source of the problem was the 'B' RPS inverter. The results of the troubleshooting determined that the problem was likely due to a defective circuit card in the inverter. The inverter performance with a new card is being monitored to ensure proper operation prior to returning the inverter to service.

At the end of the inspection period Unit 1 was at 100% and Unit 2 was increasing power to 100% in preparation for performing the 100 hour warranty run.

3.0 Update/Closeout of Open Items and Temporary Instructions (255104)

3.1 (Closed)

TI 2515/104: Inspection of Fitness for Duty (FFD) Training Program

The resident inspectors attended training on Fitness for Duty for supervisors and nonsupervisory personnel working at the Limerick site. The scope of the training was adequate and the method of training appeared to be effective. PECo does not offer a training session separately for escorts. However, during the nonsupervisory session, a portion of the training has special emphasis on escorting personnel.

The inspectors also reviewed PECo's Drug and Alcohol policy, which was provided to all employees, as well as the implementing procedures, which were provided to all supervisors and managers. The inspectors concluded that the above policy and implementing procedures, as well as the training sessions attended, provide the personnel with a keen awareness of the FFD program and the consequences that can result by nonadherence. The inspectors also concluded that PECo was committed to a sound FFD program with an in-depth set of implementing procedures for administering the program.

4.0 Surveillance/Special Test Observations (61726)

During this inspection period, the inspector reviewed in-progress surveillance testing as well as completed surveillance packages. The inspector verified that surveillances were performed in accordance with licensee approved procedures, plant technical specifications, and NRC Regulatory Requirements. The inspector also verified that instruments used were within calibration tolerances and that qualified technicians performed the surveillances. The following surveillance tests were reviewed:

Unit 1

ST-6-092-313-1	D13 Diesel Generator Operability Test Run
ST-1-076-321-0	A Standby Gas Treatment System Charcoal Adsorber/HEPA Filter Test
ST-2-041-616-1	RPS-MSIV-Closure, Division IA, Channel A1 Functional Test (Partial)
ST-2-041-617-1	RPS-MSIV-Closure, Division IB, Channel B1 Functional Test (Partial)
ST-2-041-618-1	RPS-MSIV-Closure, Division IIA, Channel A2 Functional Test (Partial)
ST-2-041-619-1	RPS-MSIV-Closure, Division IIB, Channel B2 Functional Test (Partial)

No discrepancies were noted.

5.0 Maintenance Observations (62703)

The inspector reviewed the following safety related maintenance activities to verify that repairs were made in accordance with approved procedures, and in compliance with NRC regulations and recognized codes and standards. The inspector also verified that the replacement parts and quality control utilized on the repairs were in compliance with the licensee's QA program.

TCA 1898	Temporary Circuit Alteration for Unit 1 MSIV HV-41-1F022B Failed Limit Switch
MRF 8910679	Unit 2 'F' Safety Relief Valve Acoustic Monitor Reinstallation and Retest
MRF 8909102	'A' Train Standby Gas Treatment System Charcoal Bed Replacement
MRF 8910687	'B' Train Standby Gas Treatment System Repair and Charcoal Replacement
MRF 8910610	Unit 2 Post LOCA Radiation Monitor Cable Repair

On December 11, PECo maintenance personnel began replacing the 'A' Standby Gas Treatment (SBGT) system charcoal adsorber bed. The bed was being replaced because the four sample canisters had all been removed for a laboratory analysis during the life of the bed. Without an available sample canister the adequacy of the charcoal bed could not be ascertained should the bed be exposed to any detrimental chemicals or paint fumes. Thus, the replacement was being performed on a scheduled basis to ensure it could be accomplished within the seven day technical specification requirement. During the work, charcoal was found to be leaking from the charcoal bed. The leakage was caused by failed tack welds where the retention screen is welded to a lower support band. The failure had apparently occurred sometime after February 1989 which is when it was successfully tested for bypass leakage. A weld repair was performed and the bed was returned to operation on December 16, 1989.

When the charcoal leakage was found on the 'A' train bed, the accessible portions of the 'B' train were inspected using a borescope and no leakage was found. After the 'A' bed was returned, a more detailed visual inspection was performed on the 'B' bed and a minor amount of charcoal leakage was found. However, the limited extent of the leakage was such that the charcoal bed was operable and able to perform as designed if called upon. The leakage in the 'B' train bed was also due to a failed weld on the retention screen. Following the inspection PECo decided to take the 'B' train out of service, remove the charcoal, repair the weld and replace the charcoal. The work and retesting was completed on December 21 and the system declared operable.

PECo is continuing the evaluation into the weld failures to determine the cause of their failure. The results of this evaluation will be provided in a supplemental report to the LER. In the interim, PECo management is currently looking into ways of inspecting the charcoal beds during operation to identify if this is a recurring problem. The resident inspectors will continue to follow PECo's progress.

6.0 Power Ascension Test Program (PATP) Unit 2 (72301, 72302)

6.1 Overall Power Ascension Test Program

At the beginning of this report period, Test Condition (TC) 6 (95-100% power, 100% control rod line) testing was in progress. Major testing evolutions conducted included the trip of one reactor feed pump and the full closure of the main steam isolation valves (MSIV). A scheduled Unit 2 maintenance outage was begun on December 1, 1989, following the MSIV full closure test and completed on December 22, when the mode switch was placed in start-up. At the end of the inspection period, all TC-6 testing and test result reviews were completed. Reactor power was being increased in preparation for the warranty run (100 hours at 100% power).

6.2 Power Ascension Testing Activities

The inspectors witnessed portions of the power ascension testing activity discussed below. The performance of this test was witnessed to verify the attributes previously identified in Inspection Report No. 50-353/89-24, section 4.3.

2STP-25.3, Full MSIV Isolation - On December 1, 1989, the inspectors witnessed the Full MSIV Isolation test from 100% reactor power. The pre-test briefing was witnessed and verified to be adequate. The test director reviewed the test acceptance criteria and cautions and coordinated how the test would be conducted. The full MSIV isolation was initiated by tripping two Rosemount trip units to simulate a low steam line pressure signal. The MSIV closure resulted in a reactor scram. The plant performed as expected and all systems functioned normally with the exception of the HPCI discharge valve to feedwater position indication as discussed in section 2.1.

6.3 Power Ascension Test Results Evaluation

All startup tests for Test Condition 6 were reviewed to verify that all acceptance criteria had been satisfied, the Test Exception Reports were adequately resolved and that the test results were appropriately reviewed and approved. In addition, the startup tests discussed below were reviewed for the attributes identified in Inspection Report 50-353/89-24, Section 4.4. Except as noted below, all startup test results were found to meet the attributes referenced above.

6.3.1 Test Condition 6

2STP-5.8, Scram Timing of Selected Rods During Planned Scrams of the Startup Test Program, results approved November 22 and December 12, 1989

Scram timing was conducted for control rods 22-11, 26-23, 34-19 and 58-43 during the full Main Steam Isolation Valve (MSIV) Isolation test on December 1, 1989. All scram times were acceptable.

In addition, scram times for the above four control rods had been previously analyzed to verify the requirements for conduct of 2STP-5.8 during the inadvertent scram due to a turbine load reject on November 10, 1989. The inspector reviewed Test Exception Report (TER) 186 which justified use of scram times from the inadvertent scram and found the TER to be acceptable. The control rods scrammed to position 05 in less than or equal to 7.0 seconds.

2STP-19.2, Process Computer Calculation, results approved November 17, 1989

At a reactor thermal power of approximately 97% of rated and core flow of approximately 95.5% of rated, the reactor was verified to be operating within its Technical Specification thermal limits.

2STP-23.4, Loss of Feedwater Heating, results approved December 1, 1989

All acceptance criteria were satisfied.

2STP-23.5, Feedwater Pump Trip, results approved December 7, 1989

One feedwater pump was tripped from a reactor power of approximately 97% rated. All acceptance criteria were satisfied. The reactor avoided the low water level scram by a margin of 14.8 inches (requirement of $\geq$3 inches).

2STP-25.3, Full MSIV Isolation, results approved December 19, 1989

Overall reactor response to the full MSIV isolation was satisfactory. However, six TERs were written to document operational concerns regarding the Reactor Coolant Isolation Cooling System (RCIC), the High Pressure Coolant Injection (HPCI) system and the Safety Relief Valves (SRVs). The inspector reviewed the TERs and the resolutions for each and found them to be adequate. The TERs included an acceptance criteria violation for RCIC and HPCI systems not reaching rated flow following automatic initiation due to the operator taking manual control of the systems. This TER was evaluated and accepted as is based upon the observed system operation prior to taking manual control and the previous successful testing of HPCI and RCIC earlier in the power ascension test program. Of the TERs written to document SRV concerns, one included evaluation of two relief valves having setpoints of 1150 psi lifting at a maximum reactor pressure of approximately 1132 psi. This condition was evaluated and accepted as is based upon the premature opening of the SRVs resulting from the vibration caused by the opening of the other SRVs on the same main steam line. The inspectors discussed this condition with PECo test engineers and reviewed the safety evaluation. The evaluation was determined to be adequate and the inspectors had no further questions.

2STP-27.4, Turbine Trip at TC-6, results approved November 22, 1989

On November 22, 1989, the inspector attended a Plant Operations Review Committee (PORC) meeting at which TER 188 was reviewed and approved. This TER was used to document the technical justification for using the inadvertent generator load rejection scram of November 10, 1989 to verify the requirements of 2STP-27.4, "Turbine Trip at TC-6." In the TER, each acceptance criteria of 2STP-27.4 was addressed individually as to whether the results of the inadvertent scram could be used directly to verify the criteria or whether additional analysis and review was required to accept the results. All acceptance criteria were verified with the exception of reactor water level staying below +54 inches (level 8 trip). It was not possible to verify this criteria because the operator tripped the feed pumps before reactor level had peaked during the transient.

An evaluation conducted by GE San Jose showed that insufficient data was available to demonstrate that the level 8 trip would have been avoided. However, if the level 8 trip had occurred, it does not involve a safety issue because the feedwater system is not a safety related system. During a turbine trip, the normal operator response, as demonstrated by this event, is to trip the feed pumps in order to control reactor level, thereby preventing the high water level trip of the feed pumps. Based upon this evaluation, it was determined that conduct of a turbine trip test prior to completion of the power ascension program was not required. However, GE did recommend that the turbine trip test be performed prior to the first refueling outage for the following reasons: (1) to demonstrate that the system performs as designed in accordance with U.S. NRC Regulatory Guide 1.68.1, Revision 1, "Preoperational and Initial Startup Testing of Feedwater and Condensate Systems for Boiling Water Reactor Power Plants," which requires tests to verify that the acceptance criteria for maximum and minimum water levels in the reactor vessel are not exceeded as a result of plant transients, such as a turbine trip, with the control system in the automatic mode of control; and (2) system performance per design reduces post trip operator burdens.

TER 188 was closed and TER 198 was written to track completion of the required testing. The inspector reviewed the TERs and found the analysis to be acceptable. PECo currently plans to perform the turbine trip prior to the first Unit 2 refueling outage. This issue is considered an unresolved item (50-353/89-31-01) pending completion of the required testing and evaluation and approval of the results.

<u>2STP-30.1, Recirculation System One Pump Trip, results approved November 22, 1989</u>

One recirculation pump was tripped from approximately 96% rated thermal power. All acceptance criteria were satisfied. The reactor water level margin for avoiding a high level turbine trip during the single recirculation pump trip was 10.3 inches, well within the acceptance criterion of $\geq$3 inches. The average power range monitor (APRM) margin to avoid a scram was verified to be 29% which was greater than the minimum required margin of 7.5%.

2STP-36.1, Main Steam Piping Vibration during Main Turbine Stop Valve and Control Valve Closures, results approved November 22, 1989

TER 187 was written to document the justification for using the inadvertent generator load rejection scram of November 10, 1989 to verify the requirements of 2STP-36.1. In the TER, each acceptance criteria of 2STP-36.1 was addressed individually as to whether the criteria was met during the inadvertent scram. It was determined that all criteria were met with the exception of the verification of main steam line vibration data for the balance of plant piping outside the drywell. The sample plan scan rate of the Plant Monitoring System (PMS) computer was too low to evaluate the data. Bechtel Engineering conducted a post-transient walkdown of the piping and an evaluation of the available data and judged that the integrity of the piping or supports will not be compromised for an additional similar trip (95-100% power) at which time data will be collected and evaluated and reviewed for system impact.

The inspector reviewed TER 187 and found the analysis to be acceptable. The requirement to perform a planned turbine trip to acquire the needed vibration data is also documented in TER 198 (see discussion of 2STP-27.4) and is considered part of the unresolved item. In addition the PMS computer sample plan is currently set with the correct data points and sample rate to obtain the required data if an inadvertent trip should occur.

7.0 Review of Periodic and Special Reports (90713)

Upon receipt, the inspector reviewed periodic and special reports. The review included the following: inclusion of information required by the NRC; test results and/or supporting information consistent with design predictions and performance specifications; planned corrective action for resolution of problems; and reportability and validity of report information. The following periodic report was reviewed:

Monthly Operating Report - November 1989

The inspector had no questions regarding this report.

8.0 Licensee Event Report Followup (90712, 92700)

The inspector reviewed the following LERs to determine that reportability requirements were fulfilled, that immediate corrective action was taken, and that corrective action to prevent recurrence was accomplished. In accordance with the above inspection modules the inspector considers the following reports closed. The inspector had no further comments or questions except as noted.

LER Number	Subject/Comments
1-89-055	The 'A' Reactor Protection System (RPS) shunt trip breaker tripped resulting in various Nuclear Steam Supply Shutoff System (NS4) isolations. The loss of power also caused reactor enclosure and refuel floor isolations, reactor enclosure recirculation system and standby gas treatment system initiations and a half scram signal. The breaker trip occurred as the RPS power supply was being transferred from its alternate to its normal supply following maintenance. The operators quickly recognized the problem and promptly restored power and returned the systems to a normal lineup. There was no significant effect on plant operations during this occurrence. The exact cause for the breaker trip could not be immediately determined; however, the undervoltage relay has been removed and sent to the vendor for testing. If an exact cause is found the LER will be supplemented to document the findings. Several additional problems associated with the Reactor Water Cleanup System were encountered due to the various isolations. The inspector reviewed the license actions to resolve these equipment problems and determined that they were appropriate.
1-89-056	The Reactor Water Cleanup (RWCU) system isolated due to a defective switch in the nonregenerative heat exchanger room temperature instrumentation. The RWCU system was restored to service in approximately 19 minutes, thus there was no significant effect on reactor water chemistry. The defective switch was replaced. This appears to be a random failure and not a design deficiency.
1-89-057	The refuel floor secondary containment isolated during the passage of a severe storm front. At the time of the event, winds were gusting up to 70 mph and outside air temperature dropped 15 degrees Fahrenheit in a short period of time. The combined effects of the wind gusts on the normal ventilation system and the colder air expanding, as it entered the warm building, resulted in a decrease in the building differential pressure until the isolation setpoint was reached. The standby gas treatment system started and restored the required differential pressure, thus the loss of the normal ventilation system had no safety impact. When the front passed normal ventilation was restored and operated as designed.

1-89-058	The Reactor Water Cleanup (RWCU) system isolated due to a circuit card failure in the differential flow detection instrumentation. The card was replaced and the RWCU system returned to service after approximately 1 1/2 hours. During this time the reactor water system chemistry remained within the plant Technical Specification limits. As this occurrence was the first failure of the flow summer card, the cause appears to be due to a random component failure and not a design deficiency.
2-89-011	The NSSS outboard containment isolation logic was deenergized when a fuse blew during surveillance testing. The fuse blew when a test lead became disconnected from a voltmeter and caused a short to ground. Although several system outboard valves closed as designed, the plant operators responded quickly to the event and promptly restored the systems to normal lineup, thereby avoiding any adverse effects on plant operation. New test leads have been procured to prevent recurrence.
2-89-012	The RWCU System isolated due to a steam leak detection system signal generated when the regenerative heat exchanger relief valve developed a seat leak which resulted in a high ambient temperature in the room. The system was isolated for approximately 55 hours, however reactor water chemistry remained within plant Technical Specification limits. This type of failure has been experienced in the past and a different model relief valve has been installed in an attempt to improve the system reliability. The inspectors will monitor the effectiveness of the new valve during future operation.
2-89-013	A reactor scram occurred during power ascension due to actuation of a main generator differential current relay. The relay actuation resulted in a turbine trip and subsequent reactor scram due to a turbine control valve fast closure signal. The cause of the relay actuation was an improper setting that resulted from a calculation error during the Unit 2 design. The reactor protection system operated as designed; however, some momentary spiking of reactor vessel level instrumentation was observed and the high pressure coolant injection system operated erraticly after receiving a momentary actuation signal. The scram and associated problems are discussed in detail in NRC Inspection Report 50-352/89-21, section 2.2.

The licensee actions during and following the event were found to be appropriate and in accordance with established procedures.

2-89-014 An actuation of the Primary Containment and Reactor Vessel Isolation Control System occurred when plant operators were breaking the main condenser vacuum during a planned reactor shutdown. The plant shutdown procedure, GP-3, did not direct the operators to bypass the condenser low vacuum isolation prior to breaking vacuum. The main steam isolation valves were already closed when the isolation signal was actuated thus no valve movement occurred as a result. GP-3 has been revised to include additional direction on the operation of the condenser low vacuum bypass switches.

9.0 NRC Generic Letter 88-20, Individual Plant Examinations for Severe Accident Vulnerabilities

On November 23, 1988, the NRC issued Generic Letter (GL) 88-20 to request an Individual Plant Examination (IPE) for Severe Accident Vulnerabilities from all licensees. The general purpose of this examination is to (1) develop severe accident behavior, (2) understand the most likely severe accident sequences that could occur, (3) gain quantitative understanding of the overall probabilities of core damage and fission product releases and (4) reduce the core damage and fission product releases by modifying hardware and operating procedures that are intended to prevent or mitigate severe accidents.

The NRC issued NUREG 1335 in August 1989 to provide specific guidance for developing an IPE. Supplement No. 1 to GL 88-20 was issued on August 29, 1989 to announce the issuance of NUREG 1335. GL 88-20 and its supplement requested the licensee to submit their proposed program for completing the IPE to the NRC within 60 days of the publication of NUREG 1335 and address the following:

1. Identify the method and approach selected for performing the IPE.
2. Describe the method to be used for the examination.
3. Identify the milestones and schedules for performing the IPE and submitting the final results to the NRC.

Additionally, the GL requested the licensee to complete the IPE and submit the final report within three years of the issuance of NUREG 1335.

The PECo Reliability Assessment Program was established to provide the program objectives and planned activities for the Probabilistic Risk Assessment (PRA) in support of Limerick Station. One of the objectives of the program was to revise and submit the Limerick PRA as outlined in the IPE GL.

PECo's proposed program developed in response to GL 88-20 was submitted to the NRC on October 31, 1989. The inspector reviewed the licensee's actions in response to GL 88-20 and verified the following:

1. NUREG 2300/PRA Procedure Guide was used to develop the Limerick Unit 1 and 2 IPE. (NUREG 2300 is one of the acceptable methods listed in NUREG 1335.)
2. The final submittal for the Limerick IPE is presently schedule for July 1992. This meets the three year schedule outlined in GL 88-20.
3. PECo updates the Limerick PRA once every year and maintains it as a "living" document.
4. PECo uses the Limerick Level 1 PRA to plan system outages and to prioritize maintenance work.
5. General concepts of core damage conditions have been included in the licensee's operator requalification training program.

Conclusion

PECo's actions in response to GL 88-20 were timely and adequate. PECo has implemented a Reliability Assessment Program. The general scope of this Reliability Program is to maintain an updated risk model based on current as-built configuration of both Limerick units and to increase the awareness of PRA concepts within station operations. This program has achieved the following:

1. Developed a well documented Limerick PRA model for predicting the core damage frequency and the ability of containment to mitigate accident sequences.
2. The Limerick computer risk model is maintained in a "living" format with a one year update interval.
3. The program also assures that the Limerick PRA is revised and submitted as required in GL 88-20.

10.0 ASCO Solenoid Valve Problems at River Bend

River Bend Station has experienced problems with ASCO dual coil Model NP 8323 solenoid valves sticking after being in operation for several months. At River Bend these valves are used in the instrument air supply line for the main steam isolation valves (MSIVs). It was determined that the failure of these solenoid valves could result in the loss of the isolation capability for the MSIVs. The inspectors reviewed this concern with the PECo technical staff and determined that no ASCO dual coil Model NP 8323 solenoid valves are in use at Limerick.

11.0 Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable, deviations or violations. One unresolved item was identified and discussed in paragraph 6.3.1.

12.0 Exit Interview (30703)

The NRC resident inspectors discussed the issues in this report with the licensee throughout the inspection period, and summarized the findings at an exit meeting held with the site Vice President, Mr. G. M. Leitch and the plant manager, Mr. M. J. McCormick, Jr., on December 29, 1989. No written inspection material was provided to PECo representatives during the inspection period.