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REGION I

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Facility Name:	Limerick Generating Station, Units 1 and 2
Inspection Period:	June 7 - July 18, 1992
Inspectors:	T. J. Kenny, Senior Resident Inspector L. L. Scholl, Resident Inspector D. T. Moy, Reactor Engineer M. J. Buckley, Reactor Engineer

Approved by: Clifford J. Anderson 7/30/92

Clifford J. Anderson, Acting Chief
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Date

Inspection Summary: This inspection report documents routine and reactive inspections during day and backshift hours of station activities including: plant operations; radiation protection; surveillance and maintenance; and safety assessment/quality verification.

EXECUTIVE SUMMARY
Limerick Generating Station
Report No. 92-17 & 92-17

Plant Operations

The startup of Unit 1 following the refueling outage was very well controlled. Procedural use and professionalism was excellent. The control room operators and the fire brigade performed well in response to a fire on the 1B reactor feed pump lagging. (Section 1.2.)

Surveillance and Maintenance

Weaknesses in the surveillance testing program were identified in that the suppression pool cooling mode of the residual heat removal system was not being tested properly (Unresolved Item 50-352/92-17-01 and 50-353/92-17-01) and the stroke time test of the automatic depressurization system valves did not specify acceptance criteria. (Sections 1.3 and 3.3.) The response to a 10 CFR 50 Part 21 report regarding Rosemont transmitters was prompt and thorough. (Section 3.1.)

Engineering and Technical Support

The PECo response to NRC Bulletin 92-01 regarding Thermo-Lag Fire Barriers was found to be prompt and comprehensive. (Section 4.1.)

Safety Assessment and Quality Verification

PECo's response to the NRC Information Notice 92-30 regarding the falsification of plant records at another facility was found to be thorough. (Section 6.2.)

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DETAILS

1.0 PLANT OPERATIONS (71707)[1]

The inspectors conducted routine entries into the protected areas of the plant, including the control room, reactor enclosure, fuel floor, and drywell (when access was possible). During the inspections, 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 evaluated the licensee's compliance with 10 CFR, Technical Specifications, License Conditions and Administrative Procedures. During this period, the inspectors performed 7 hours of deep backshift inspections.

1.1 Operational Overview

Unit 1

The fourth refueling outage was completed and a plant startup was performed on July 4, 1992. Full power operation was achieved on July 16, 1992.

Unit 2

Unit 2 continued to operate at or near 100 percent power and at the end of this report period had attained 410 days on line.

1.2 Reportable Events

Unit 1

Unusual Event (Fire)

On June 13, 1992, at 7:10 p.m., an unusual event was declared resulting from a fire beneath the insulation of 1B reactor feed pump (RFP) turbine. The fire was discovered by an operator at 6:42 p.m. The fire brigade responded and the fire was reported extinguished at 7:05 p.m.

A review of station logs by the resident inspector and discussions with operations personnel yielded the following event sequence.

- At the beginning of the second shift operators noticed a strong odor near the 1B RFP. This odor generally accompanies a plant heat-up. However, as the shift progressed the odor became stronger. Observations were made more frequent in the area.

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[1]The NRC Inspection Procedures used as guidance are listed parenthetically throughout this report.

- At 6:42 a.m., smoke was observed above the shield wall and the operator called for Health Physics (HP) personnel and a key to enter the area.
- The HP, with the key, arrived at 6:42 p.m. and the door was opened. Flames were observed around the thrust bearing area of the turbine.
- The control room was informed immediately and Unit 1 power was decreased to 73 percent and 1B RFP was turned off.
- The fire was saturated with CO_2 which extinguished the visible flames. The insulation was removed with a crow bar and the insulation was sprayed with fire water. The turbine causing was not sprayed with water.
- The unusual event was terminated at 7:52 p.m.
- The operations personnel checked the entire area including below the RFP area to see that no other equipment had been affected. The only damage noted was a burnt wire for the vibration probe of the 1B RFP.

Conclusion:

PECo personnel responded rapidly to this event and possibly prevented further damage to equipment. Good judgement was shown by the "on scene" fire brigade leader by not immediately spraying the hot turbine casing with cold water. The fire was effectively put out with minimum damage to equipment and the surrounding area.

The burnt cable was repaired and the turbine was returned to service on July 15, 1992. The cause of the fire appeared to be oil under the insulation. Although no oil leaks were found on the turbine after the fire, it is suspected that the oil may have got on the lagging when maintenance was performed during the recent outage on Unit 1.

Unit 2

Inoperable Fire Barriers

On June 30 a one-hour non-emergency notification was made as a result of inoperable fire barriers. See Section 4.2 for details.

Blowout Panel Opening

On June 21, 1992, at 8:31 p.m., the North wall blowout panel at elevation 217 feet of the reactor building (Unit 2) opened due to a pressure in the reactor enclosure (RE). This event was reported to the NRC via the ENS as an event that would have compromised the ability to control radioactive releases if an accident occurred. This required the operators to place the unit in a four hour action statement, which is to repair the breech of secondary containment or be in hot shutdown within the next 12 hours. The panel was restored in 2 hours and 59 minutes. The release to the environment was less than 0.01 percent of regulatory limits.

The events leading up to the panel opening are as follows:

- A monthly flow surveillance test of the standby gas treatment system (SGTS) and the reactor enclosure recirculation system (RERS) was in progress on June 21, 1992 2031 hours (ST-6-076-250-2).

- During the test the chief operator observed a failure of the "OA" SGTS fan to maintain Unit 2 reactor enclosure (RE) differential pressure. The shift supervisor (SS) directed the start of "OB" SGTS fan. The "OB" fan could not maintain the required differential pressure. An investigation by PECo operators found the blowout panel opened.

- The surveillance test requires the starting and stopping of all the fans within the systems described above. Flow traces, reviewed by the inspector, showed that the RE did not exceed 0.15 inches of water pressure (.025 pounds per square inch (PSI)). The panel setpoint is .25 PSI. All tested ventilation systems worked properly.

Because of finding the blowout panel open, PECo engineers began an investigation to ascertain why the panel opened prematurely. The following are PECo's conclusions:

- The panel was possibly stressed during an earlier event and this condition was not noticed by PECo personnel. A surveillance is performed every 18 months (RT-1-076-900-2) to determine the condition of all blowout panel washers. The test was performed on July 13, 1990, satisfactorily. The surveillance is a visual inspection of the convex washers around the bolt fasteners to ensure that they are not flattened. This condition of the washers would lower their threshold for breaking as designed.

- Several ventilation transients were experienced after the last inspection of the blowout panels in the RE.

A subsequent inspection of the panels by PECo revealed several deformed washers. The washers were replaced.

As documented in combined inspection report 50-352/92-15 and 50-353/92-15, the blowout panels were upgraded on Unit 1 to a blowout pressure of 0.5 psig. This modification is scheduled to be incorporated in Unit 2 during the next refueling outage. The inspector considers PECo's actions adequate and has no further questions at this time.

The NRC received reports of the above events via the Emergency Notification System (ENS). The inspectors determined that the licensee's initial response and corrective actions were appropriate. The root cause analysis of the event and the need for additional/long-term corrective action will be reviewed upon issuance of the Licensee Event Reports as part of the routine inspection program.

1.3 Startup Testing Following Refueling (Cycle 5)

The cycle 5 startup was conducted according to procedure GP-2, Normal Plant Startup, Rev. 38, dated June 10, 1992. This procedure outlined the steps in the startup, set initial conditions and prerequisites, specified calibration or surveillance procedures at appropriate points in the sequence, and referenced detailed tests and data collection in separate test procedures.

Review of Cycle 5 Shutdown Margin Requirement

The inspector independently reviewed shutdown margin predicted values and acceptance criteria obtained from PECo cycle management report, cycle 5, and Limerick Unit 1 Technical Specification (TS). The areas evaluated and documented in the cycle management report include: the end-of-cycle 4 assumptions, the full core loading for cycle 5, the control rod patterns (including thermal performance) and shutdown margin demonstration test data. PECo used the shutdown margin data in surveillance test, ST-6-107-875-1, to determine the shutdown margin for cycle 5. The inspector reviewed the results and verified that the calculated shutdown margin (1.73 percent delta K/K) was in excess of the TS requirements (0.38 percent delta K/K).

Cycle 5 Startup Testing

The inspectors reviewed calibration and functional test results to verify the following:

- The procedures contained sufficient detailed instructions;
- The technical content of the procedures was sufficient to provide satisfactory component calibration and test data; and
- The acceptance and operability criteria obtained were in compliance with TSs.

The following tests were reviewed or witnessed by the inspectors:

ST-2-074-610-1, IRM Channel C Functional Test, Rev. 12.
This functional test validates once per 7 days testing required by TS Table 4.3.1-1 Items 1.a and 1.b by verifying that the IRM Drawer C, and associated trip functions are operational. No significant observations were made.

ST-6-107-885-1, Thermal Limits Determination for Two Recirc Loop Operation, Rev. 25.
This test verifies the thermal limit of average planar linear heat generation rates (APLHGR), minimum critical power ratio (MCPR) and linear heat generation rate (LHGR). The readings

taken by the operator at control panel 10C608 were within TS requirements. No significant observations were made.

ST-6-055-230-1, HPCI Pump, Valve and Flow Test.

The High Pressure Coolant Injection (HPCI) Pump, Valve and Flow Test was performed on July 6, 1992 with the reactor power at approximately 6 percent. The surveillance test (ST) was witnessed by the inspectors from the control room. The test was unsuccessful due to a problem with the HPCI tappet assembly associated with the overspeed mechanism. The requirement of TSs 4.5.1.b.3 was not met and PECo declared the HPCI system inoperable. The oil pressure in the 38 psig oil supply line to the tappet assembly was maintaining a higher pressure than normal. The higher pressure caused an overspeed condition and the overspeed trip mechanism to actuate. Maintenance performed work on the supply line and found foreign material in the 3/32" orifice line to the sump and in the trip tappet assembly. The PECo Metallurgical Testing lab in Valley Forge performed an analysis of the material and concluded that it was paint chips. The supply line was cleaned and restored. The second run of the ST was conducted and completed satisfactorily on July 8, 1992. The root cause determination and actions to prevent recurrence will be reviewed upon submittal of the LER.

During the review of the ST an inspector concern was brought to the attention of PECo. This concern was that a prerequisite in the procedure would not have been met if the ST had been completed during the first performance attempt. The prerequisite required that diagnostic testing be performed in conjunction with the ST. PECo was using this step to remind the technician that a previously submitted LER committed to test the valve on a quarterly basis. PECo issued a temporary procedure change prior to the second attempt at performing the ST. This concern was adequately addressed.

The test and operating shift personnel performed satisfactorily. They adhered to procedures and demonstrated adequate communication when verifying data from operators located in the plant. The inspectors concluded that this test was satisfactorily performed.

ST-5-097-355-1, Core Post - Alteration Verification.

This procedure verifies that the core was loaded according to the approved core load procedure. The inspectors reviewed the procedure and portions of the video of the final core configuration and found them satisfactory. There was a concern with a missing retaining ring from a globe valve in the shutdown cooling return line. During reassembly of the valve, maintenance personnel discovered that a retaining ring was missing. The first three to four feet of piping on either side of the valve and around the valve body was inspected with mirrors. The part could not be located. A lost parts analysis was performed to evaluate the safety significance of the part remaining in the reactor vessel. The inspector reviewed the safety analysis, that had been approved by the Plant Operations Review Committee (PORC), and concluded that the part would be restricted in the jet pump nozzle and would not come in contact with the fuel. Further analysis showed that flow through the jet pump would not be affected significantly enough to cause a concern. The inspector has no further questions regarding this lost part.

S32.1.A, Synchronizing Main Generator to Grid
The Synchronizing Main Generator to Grid procedure was reviewed and no significant observations were made. The main generator was synchronized to the grid at 11:53 a.m., Thursday, July 9, 1992.

ST-6-049-230-1, Reactor Core Isolation Cooling Pump, Valve and Flow Test
During the performance of the Reactor Core Isolation Cooling (RCIC) Pump Valve and Flow Test, the minimum flow valve HV-049-1F019 cycled partially closed and then opened during the minimum flow operation of step 6.3.37. This was attributed to fluctuations in the flow indication due to air in the transmitter. The RCIC Pump ran at set flow for two hours and fulfilled all tes equirements. All indications were normal. A temporary change was initiated to co..ect and evaluate new baseline reference data following an 18 month maintenance overhaul.

ST-3-107-790-1, Control Rod Scram Timing
During the review of Control Rod Scram Timing, performed on June 2, 1992 the average insertion scram times for the three fastest control rods in each group of four were verified for each group including control rod 50-47. This rod had a scram ins on time slightly greater than the average that is required by echnical Specification's. Control rod 50-47 scram time was within the limit for an individual control rod. The inspector identified a typographical error in the procedure which PECo then corrected by the issuance of a change to the procedure. The error did not affect the performance of the test.

ST-6-050-760-1, Automatic Depressurization System Valve Exercising
During the review of the Automatic Depressurization System (ADS) Valve Exercising procedure, it was noted that no acceptance criteria was included for the opening times for the five ADS valves. The procedure was signed, reviewed and completed satisfactorily by the test engineer without comment on July 5, 1992. Limerick's Administration Procedure A-80.7, Inservice Testing, requires maximum stroke times be specified, based on vendor supplied data and re rified as results from subsequent inservice tests are available. After discussions with Pl ngineers regarding timing of ADS Valves, according to the ASME Code, a test change issued to add the acceptance criteria.

GP-2, Normal Plant Startup
GP-2 is intended as guidance and requires that steps be performed as written. This procedure is applicable for startups after short duration shutdowns with minimal maintenance performed, after maintenance outages, or after refueling/long term outages. Required plant systems were specified and instructions for the performance of the procedure are complete to the extent necessary to ensure that procedure objectives are met. Special precautions for personnel and equipment safety are specified.

2.0 SURVEILLANCE/SPECIAL TEST OBSERVATIONS (61726)

During this inspection period, the inspector reviewed in-progress surveillance testing and completed surveillance packages. The inspector verified that surveillances were done according to PECo approved procedures and plant Technical Specification requirements. The inspector also verified that the instruments used were within calibration tolerance and that qualified technicians did the surveillances.

Surveillance testing observed and/or reviewed included:

ST-6-020-813-1	D13 Diesel Generator Fuel Oil Analysis
ST-6-020-233-1	D13 Diesel Generator Fuel Oil Transfer Pump, Valve and Flow Test
ST-6-047-750-2	CRD Accumulator Pressure Check
SP-S-079	Demonstration of Unit 1 Main Steam Line Drains as an Alternate Decay Heat Removal Method
RT-1-076-900-2	Blowout Panel Inspection

The activities observed by the inspectors were acceptable.

3.0 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 PECo's Quality Assurance (QA) program.

The inspector reviewed the following procedures used during this evolution.

WO C0087539	Replacement of Blowout Panel Washers
WO C0086436	1B Residual Heat Removal Heat Exchanger Cleaning

The activities observed by the inspectors were acceptable.

3.1 Rosemont Transmitters

On July 9, 1992, PECo became aware of a 10 CFR Part 21 report delineating that the end caps of Rosemont transmitters models 1152, 1153 and 1154 may not be properly torqued to 200 inch pounds. PECo store room personnel conducted a search of records to identify how many of the questionable transmitters were purchased. PECo had purchased and received nine such transmitters of which five were installed in Unit 1 and four were in the store room.

PECo checked the end cap torque settings and found that the five in the plant were correct and the ones in the store room were also found correct. PECo's installation procedures reference the checking of the torque settings on the installed transmitters.

The resident inspector considers this matter closed.

3.2 Residual Heat Removal System Heat Exchanger Fouling and Corrosion

As previously discussed in NRC Inspection Report 50-352/92-15 the 1A Residual Heat Removal (RHR) System heat exchanger failed a heat transfer test performed in March, 1992. When chemical cleaning did not significantly improve the heat transfer results the heat exchanger was opened to permit inspection and cleaning of the tubes. The heat exchanger was found to have a layer of biological and non-organic material coating the insides of the tubes. The material was cleaned off by high pressure hydrolasing of the tubes and then eddy current testing was performed. The eddy current results showed there was pitting of the tubes caused by under deposit corrosion and microbiologically inducted corrosion (MIC). Two tubes were removed to permit additional testing and examination in the PECo Metallurgical Laboratory. The Nuclear Engineering Department (NED) evaluated the results of the testing and recommended that all tubes that had pitting indications greater than 80 percent through the tube wall be plugged. This resulted in a total of 37 tubes plugged out of a total of 530.

The 1B heat exchanger had passed the heat transfer test, however, it also was opened for inspection and cleaning. Following the hydrolasing of the tubes, eddy current testing was performed with similar results to the 1A heat exchanger. Using the same 80 percent through wall pitting criteria, thirty-five tubes required plugging.

Heat transfer testing was performed on the 1A and 1B heat exchangers following the cleaning and tube plugging and the results of both tests were satisfactory.

On June 25, 1992, representatives from PECo met with the NRC staff at the NRC Region I Headquarters to discuss the heat exchanger issue. Copies of the slide presentation and attendance list are included as Attachment A to this report.

Items presented were:

- Results of the testing and non-destructive evaluations.
- Root cause determination.
- Minimum wall thickness determination.
- Tube plugging.
- Heat exchanger operability.

- Corrosion monitoring program.
- Chemistry control program.
- Potential generic considerations.

PECo concluded that based on the testing and analysis performed there is adequate tube wall remaining to ensure continued heat exchanger operability throughout the next operation cycle. The following actions are being taken to minimize tube corrosion:

- A non-oxidizing biocide is being used as a treatment prior to placing the heat exchanger in a lay-up condition with demineralized (demin) water. The heat exchangers are flushed and placed in a demin water layup condition after each use.

- Spray pond chemistry is being monitored more frequently to ensure conditions which could cause fouling or scaling are not present.

- Procedures have been established for chemical treatment of the spray pond following a Loss of Coolant Accident (LOCA).

- Corrosion monitoring equipment will be installed on the 1A heat exchanger so that a sample tube is subjected to the same environment as the heat exchanger and, thereby, permit a more accurate assessment of corrosion during the operating cycle.

Based on PECo's actions taken, and an NRC evaluation of the information and test results, there is adequate assurance that the heat exchangers will fulfill their safety function. The resident inspectors will continue to monitor PECo's actions to ensure reliable heat exchanger operation is maintained.

The Unit 2 heat exchangers have been tested to ensure adequate heat transfer capabilities. Both tests were satisfactory. Unit 2 heat exchanger inspection and cleaning is planned for the next refueling outage scheduled for January, 1993. The Unit 2 heat exchangers have been in operation for a much shorter time than the Unit 1 heat exchangers and, therefore, are predicted to have less corrosion.

3.3 Residual Heat Removal System Heat Exchanger Bypass Valve Leakage

The inspector reviewed flow test data obtained on the 'A' loop of the RHR system operating in the suppression pool cooling mode. The purpose of the testing was to determine the flow rate through the RHR heat exchanger (Hx). Technical Specification 4.6.2.3.b requires that the suppression pool cooling mode of the RHR system be demonstrated to be operable by verifying that the RHR pumps develop a flow of at least 10,000 gallons per minute (gpm) on recirculation flow through the RHR heat exchanger, the suppression pool and the full flow test line when tested pursuant to Specification 4.0.5. Technical Specification 4.0.5 specifies the surveillance requirements for the American Society of Mechanical Engineers (ASME) Code Class 1, 2 and 3 components.

The RHR system configuration includes a flow bypass valve that can be used to bypass a portion of system flow around the RHR Hx. The bypass valve is normally closed in the suppression pool cooling mode and is primarily used when the RHR system is used in the shutdown cooling mode. In this mode adjusting bypass flow permits the operator to better control the cooldown rate of the reactor.

The flow test was performed with the bypass valve shut and the results showed a total loop flow (Hx flow plus bypass valve leakage) of approximately 10,600 gpm. This data was obtained from a computer point reading installed plant instrumentation.

A portable ultrasonic flow measuring device was installed on the piping to measure the flow rate through only the Hx. This data showed a flow rate of 9600 gpm through the Hx indicating there is leakage past the closed bypass valve.

Since the data showed there was less than 10,000 gpm through the Hx, the 'A' RHR loop was declared inoperable and a 72 hour action statement was entered in accordance with TS 3.6.2.3a. To increase heat exchanger flow to 10,000 gpm, total loop flow was increased by removing an orifice plate in the cooling loop. The 1B, 2A and 2B suppression pool cooling loops were also tested with similar results and their orifice plates were subsequently removed to increase heat exchanger flow above the TS requirement.

PECo's engineering department subsequently performed heat removal capability calculations which showed that adequate cooling was available even though prior to the orifice plate removal flow through the HX's were less than 10,000 gpm.

PECo is submitting a LER to address why the TS surveillance was not properly performed in the past. Initially, it appears that PECo misinterpreted the TS and was only verifying that total loop flow was greater than 10,000 gpm rather than flow only through the heat exchanger. This issue is an unresolved item (50-352/92-17-01; 50-353/92-17-01). The inspector will review the LER during the next inspection period.

4.0 ENGINEERING AND TECHNICAL SUPPORT (37700)

4.1 NRC Bulletin 92-01 - Thermo-Lag Fire Barrier System

On June 24, 1992, the NRC issued NRC Bulletin No. 92-01 to notify licensees of failures of fire endurance testing associated with the Thermo-Lag 330 fire barrier system. The Thermo-Lag system is used to protect safety systems that are required for the safe shutdown of the reactor.

NRC Information Notices (INs) 91-47 and 91-79 were issued on August 6, 1991 and December 6, 1991, respectively and provided information to licensees regarding Thermo-Lag testing and installation deficiencies. As a result of these INs, Texas Utilities instituted a fire endurance test program to qualify its Thermo-Lag 330 electrical raceway fire barrier systems for the Comanche Peak Steam Electric Station.

During this testing, test failures occurred on mock-up installations using Thermo-Lag 330 to protect small diameter conduit and wide cable trays. Based on these failures NRC Bulletin 92-01 required licensees to take the following actions:

1) For those plants that use either 1- or 3-hour pre-formed Thermo-Lag 330 panels and conduit shapes, identify the areas of the plant that have Thermo-Lag 330 fire barrier material installed and determine the plant areas that use this material for protecting either small diameter conduit or wide cable trays (widths greater that 14 inches) that provide safe shutdown capability.

2) In those plant areas in which Thermo-Lag fires barriers are used to protect wide cable trays, small conduits, or both, the licensee should implement, in accordance with plant procedures, the appropriate compensatory measures, such as fire watches, consistent with those which would be implemented by either the plant technical specifications or the operating license for an inoperable fire barrier.

3) Each licensee, within 30 days of receiving the bulletin, is required to provide a written notification stating whether there is or is not Thermo-Lag 330 fire barrier system installed in its facilities. Each licensee who has installed Thermo-Lag 330 fire barriers is required to inform the NRC, in writing, whether it has taken the above actions and is required to describe the measures being taken to ensure or restore fire barrier operability.

PECo's actions in response to the bulletin are as follows:

1) On June 29, 1992, an engineering review was completed which identified where the Thermo-Lag fire barrier material was installed at Limerick. Although Bulletin 92-01 specifically addresses only small diameter conduits or wide cable tray installations, PECo chose to include all Thermo-Lag installations in the scope of their response to the bulletin. This review completed Action 1 of Bulletin 92-01.

2) On June 30, 1992, continuous and roving fire watchers were established in all of the plant areas that had Thermo-Lag 330 installations. The establishment of the fire watches implemented Action a. of TS 3.7.7 for inoperable fire rated assemblies and completed required Action 2 of Bulletin 92-01.

To minimize fire watch personnel radiation exposure, PECo requested authorization from the NRC, Office of Nuclear Reactor Regulation (NRR) to use a closed circuit television in one area and a mirror in another area. This permitted the fire watches to observe the areas without entering high radiation fields. The permission was granted on July 2, 1992 as documented in a letter from Mr. R. Clark, NRR to Mr. G. Beck, PECo .

On June 30, 1992, a one-hour non-emergency notification was made to the NRC via the ENS as a result of the inoperable fire barriers. This notification was made to:

1) report a condition outside of the design basis of the plant

2) report a condition prohibited by TS, and

3) report the failure to maintain the provisions of the approved Fire Protection Program.

The ENS notification was followed by the submittal of Licensee Event Report (LER) 1-92-012 dated July 10, 1992. The written response required by Action 3 of Bulletin 92-01 is being prepared and is expected to be submitted per the 30 day requirement of the bulletin.

The resident inspectors concluded that PECo took prompt, comprehensive actions in response to Bulletin 92-01 and have no additional questions at this time.

5.0 RADIOLOGICAL PROTECTION (71707)

During the report period, the inspector examined work in progress in both units including health physics procedures and controls, ALARA implementation, dosimetry and badging, protective clothing use, adherence to radiation work permit (RWP) requirements, radiation surveys, radiation protection instrument use, and handling of potentially contaminated equipment and materials.

The inspector observed individuals frisking in accordance with HP procedures. A sampling of high radiation area doors was verified to be locked as required. Compliance with RWP requirements was reviewed during plant tours. RWP line entries were reviewed to verify that personnel provided the required information and people working in RWP areas were observed as meeting the applicable requirements. The activities observed by the inspectors were acceptable.

6.0 SAFETY ASSESSMENT/QUALITY VERIFICATION

6.1 Management Changes

On June 22, 1992, PECo announced the rotation of executives within its nuclear group. Mr. David Helwig will replace Mr. Graham Leitch as Vice President, Limerick Generating Station effective July 20, 1992. Mr. Leitch will retire from the company on August 1, 1992.

Mr. Helwig is currently Vice President, Nuclear Engineering and Services. Nuclear Engineering and Services will be reorganized into two groups with Mr. Gerald Rainey as Vice President, Nuclear Services and Mr. Gregory Cranston as General Manager, Nuclear Engineering.

6.2 NRC Information Notice 92-30 - Falsification of Plant Records

On April 23, 1992, the NRC issued the above information notice which identified that plant personnel were falsifying plant logs at several facilities. Record falsification is contrary to the requirements of 10 CFR 59.9(a) which states that information required by statute or by the Commission's regulations be complete and accurate in all material respects.

The Nuclear Network, used by PECo to keep up with events at other stations, announced the falsification of plant logs at Millstone on April 8, 1992. The Station Manager issued a memo to all operators on April 16, 1992, highlighting the seriousness of record falsification. The memo went on to discuss the expectations of PECo management.

PECo performed a series of zone traces (security system computer readouts of security zone entries) for the following departments: Operations, Health Physics, Chemistry, Maintenance and I&C. The traces tracked certain individuals, tasked with assignments, throughout the facility. The times picked for performing the traces was limited to off hours, weekends and holidays. PECo began these traces before the Information Notice was issued and went back to January 1, 1992 for their assessment The following documents the results of these surveys.

OPERATORS

Thirty-five sets of rounds were selected over 5 random days between January 20, - April 22, 1992, falling on backshift during holidays and/or weekends.

Three hundred seventy-five security monitored access locations.

Twenty different operators were monitored

Results

One non licensed operator was found to have missed non-safety related readings. However, the individual did take his TS required readings.

The individual's badge was revoked immediately and more of his rounds were investigated. No more missed readings were identified. The individual was counselled and a written reprimand was put into his service record. The individual has been returned to duty and is being monitored by supervision. The individual had no excuse for missing the readings and admitted the wrongdoing.

LUBRICATIONS

Fourteen lubrication tasks were reviewed.

Results

All lubrication tasks were performed, however, discrepancies were discovered on the dates signed for on the surveillance sheet. Lubrication tasks are assigned ahead of time (e.g., monthly). Some operators were found to have performed the task, but had signed the surveillance as completed on a different date. The operators have been counseled and PECo management has instructed that the date the lubrications are performed is the date of record.

HEALTH PHYSICS/RADWASTE OPERATORS

Ten surveillances, 10 routine tests and 5 instrument source checks were investigated by PECo.

Results

No problems were identified.

CHEMISTRY

Eleven surveillance tests were investigated by PECo.

Results

No problems were identified.

MAINTENANCE

Supervisor reverification of valve manipulations on Hydraulic Control Units.

Results

Three were in question, however, after interviews with the foremen who stated that although they were not signed in on the RWP they could observe the actions by maintenance from outside the radiological zone. The inspector went to the observation area and determined that the operator can be seen performing the valve manipulations, however, foremen have been instructed to accompany the operator in the future.

I&C

Fifteen surveillance tests were investigated by PECo.

Results

No problems were identified.

HOURLY FIRE WATCHES

Six individuals were monitored over several randomly selected days.

Results

No problems were identified.

PECo is continuing to monitor tasks within the QA and QC department. Because of PECo's intensive investigations they have identified some poor work practices such as:

- Tailgating
- Work practices of signing off logs before the work is completed - even though the work does get done subsequently.
- Misunderstandings of managements expectations of certain tasks.

After discussions with PECo management and a review of PECo documentation the inspector concluded that: PECo did a thorough job of investigating a diverse group of individuals over a representative span of time; all people that were identified in wrongdoing were disciplined appropriately; no TS type readings were in question; PECo is correcting their identified concerns; there were no safety related concerns identified by the investigations and; no licensed operators were involved in any wrongdoing.

The inspector notes that as a result of the investigation PECo has decided to include similar investigations as part of their self monitoring program. The inspector considers this matter closed.

7.0 REVIEW OF LICENSEE EVENT REPORTS (LERs), ROUTINE AND SPECIAL REPORTS (90712, 92700)

7.1 Licensee Event Reports (LERs)

LERs are 30 day reports submitted to the NRC, by PECo, as required by 10 CFR 50.73. These reports document: the major occurrences present during an event, including all component or system failures; a clear, specific, narrative description of what occurred; plant operating conditions before the event; status of contributors to the event; dates and approximate times of contributing factors; the causes and failure modes; personnel errors if applicable; procedural deficiencies if applicable and the short-term and long-term corrective actions taken to prevent recurrence. The Resident Inspector routinely reviews these documents and performs follow-up to PECo's actions regarding the disposition of corrective initiatives. In his review, the inspector validates the above and determines whether events are described accurately and whether corrective and compensatory actions have been properly addressed. Unless otherwise delineated below, the following LERs met all the requirements discussed above.

LER 1-92-009, Event Date: May 15, 1992, Report Date: June 9, 1992
Missed Surveillance
This LER reported a quarterly surveillance test on the emergency service water system which was not completed in its entirety within the required test interval. The test was subsequently completed satisfactory.

LER 1-92-010, Event Date: May 5, 1992, Report Date: July 2, 1992
North Stack Radiation Monitor Inoperable
On June 5, 1992, PECo discovered that the North Stack Wide Range Accident Monitor (WRAM) had been inoperable since May 5, 1992. During Unit 1 electrical testing power was removed to the WRAM. When the power was restored, the WRAM began using the Standby Gas Treatment System (SBGT) flow rate in its release rate calculation instead of the desired North Stack process flow. This default to SBGT flow was a result of the radiation monitoring system central pressure malfunctioning during the electrical transient caused by the test. When power was restored to the WRAM, a "Loss of Process Flow Alarm" annunciated in the main control room. When the alarm could not be cleared, the operator consulted Procedure RMMS-402, "Determining Monitor or Channel Status at the RM-11 Color Console," which directed to not declare the WRAM inoperable but to contact the system engineer. The system engineer improperly diagnosed the condition as a minor problem and the WRAM was considered to be operable. Subsequently on June 5, 1992, operations personnel again contacted the system engineer and questioned the status of the WRAM operability. With the assistance of the previous WRAM system engineer the system was properly diagnosed as being inoperable.

The isolation valves which receive isolation signals from the WRAM were closed during the entire period of the WRAM inoperability and thus no unmonitored release occurred.

PECo has revised procedure RMMS-402 and provided additional training on the system for the system engineer.

L.ER 1-92-011, Event Date: June 16, 1992, Report Date: July 10, 1992
Cable Separation Criteria Violation
This LER reported a condition in which the TS actions were not taken for a main steam line radiation monitor that was inoperable as a result of inadequate electrical separation. The condition was discovered during a panel inspection being performed by a PECo quality assurance inspector. The condition has most likely existed since the time of initial construction and was corrected by installing fiberglass sleeving on approximately 12 inches of cable. The quality assurance department has inspected all panels that have had work performed as a result of plant modifications and also plans to inspect 5 percent of all other panels on a quarterly basis.

LER 1-92-012, Event Date: June 11, 1992, Report Date: July 10, 1992
Inoperable Fire Barriers

This event is discussed in detail in Section 4.2 of this report.

LER 2-92-005, Event Date: June 5, 1992, Report Date: June 26, 1992
Inadvertent Emergency Diesel Generator Start
This LER reported an inadvertent emergency diesel generator start during surveillance testing. The cause was personnel error in that the operator inadvertently missed a step in the procedure. There were no detrimental effects to any plant equipment as a result of this event. The inspector verified the installation of operator aids that were posted as an effort to prevent recurrence.

LER 2-92-006, Event Date: June 21, 1992, Report Date: July 16, 1992.
Reactor Enclosure Secondary Containment Blowout Panel Opening
This event is discussed in detail in Section 1.2 of this report.

7.2 Routine and Special Reports

Routine and special reports are submitted by PECo to inform the NRC of routine operating conditions and other noteworthy occurrences that are reportable due to requirements in 10 CFR 20, technical specifications and other regulatory documents. The inspector reviews these reports for information and confirms the accuracy of the reports. The following reports were reviewed and unless otherwise delineated below, satisfied the requirements for which they were reported.

Monthly Operating Report for May, 1992, dated June 10, 1992.
Monthly Operating Report for June, 1992, dated July 10, 1992.

The resident inspector had no questions regarding the above listed reports.

8.0 FOLLOWUP OF PREVIOUS INSPECTION FINDINGS (92702)

(Update) Unresolved Item Nos. 50-352/90-02-01 and 50-353/91-04-01 pertaining to unsupported data used in calculation of high range radiation monitor accuracy.

Calculation EE-5506, Revision 2, dated March 21, 1991, determines the error caused by Post-LOCA insulation resistance (IR) degradation of instrument loop components associated with the Primary Containment Post-LOCA Radiation Monitoring System (PCPL-RMS). The EE-5506 calculation quantifies the IR degradation for the "worst case" PCPL-RMS loop per unit. The results of calculation EE-5506 indicate that at a temperature of 340°F, the cable insulation resistance degradation, in conjunction with the logarithmic amplifiers offset voltage, yields a worst case leakage current in the signal cable in the downscale direction. During post-LOCA high temperature conditions, cable IR degradation allows leakage currents to ground, thereby, introducing a bias error into the performance of the PCPL-RMS. This bias error is significant at the lower range of the PCPL-RMS, but becomes less significant and eventually negligible towards the upper range of the PCPL-RMS. The EE-5506 analysis further demonstrates that, during post-accident conditions, the PCPL-RMS satisfies the Regulatory Guide 1.97, Revision 2, factor-of-two accuracy requirements for the range of radiation values during which the operator must take corrective or mitigating actions. Since the PCPL-RMS provides indication of gross radioactivity in the primary containment to allow operators to assess plant conditions and initiate corrective action, plant personnel at Limerick do not take any action based on the primary containment radiation levels until a nominal 100 R/hr is detected. The range of indication for which channel accuracy is not within a factor of two for the four channels is from 43.8 R/hr for channels B and C to 75.0 R/hr for channel A.

The inspector reviewed PECo's calculation results and other documentation supporting their conclusion that the RMS was acceptable with the postulated worst case degradation.

PECo has indicated this information will be made available to plant operators. However, at the present time there is an operator aid posted in the main control room which gives the operator a correction factor to apply to the readings based on primary containment temperature. It is not certain if this operator aid is correct based on these latest calculations. This issue has been discussed with station personnel who agreed to review the question and take actions to ensure the correct information is conveyed to the operators.

These items remain open pending assurance that the plant operations personnel have adequate guidance.

9.0 MANAGEMENT MEETINGS

9.1 Exit Interviews

The NRC Resident Inspectors discussed the issues in this report with PECo representatives throughout the inspection period, and summarized the findings at an exit meeting with the Technical Superintendent, Mr. J. Muntz, (Acting Plant Manager) on July 20, 1992. No written inspection material was provided to licensee representatives during the inspection period.

9.2 Additional NRC Inspections this Period

The Resident Inspector also attended the following exit interviews during the report period:

Date	Inspector	Report	Subject
June 8-12, 1992	A. Della Ratta	50-352/92-18 50-353/92-18	Security
June 15-19, 1992	A. Finkel	50-352/92-19 50-353/92-19	Fire Protection
June 22-26, 1992	R. McIntyre	50-352/92-201 50-353/92-201	Procurement and dedication of Commercial Grade Items

9.3 SALP Meeting

A meeting was held on June 26, 1992, to discuss the Initial Systematic Assessment of Licensee Performance (SALP) report which was issued by the NRC on June 19, 1992. At this meeting PECo management stated that they concurred with the assessment and presented their plans for improving performance in several areas.

Attachment A

Meeting with PECo on RHR Heat Exchangers
June 25, 1992

Nuclear Regulatory Commission

Christopher J. Bennet
Richard J. Clark
James A. Davis
Wayne M. Hodges
Herbert J. Kaplan
Thomas J. Kenny
Jeffrey J. Lyash
Michael C. Modes
Larry L. Scholl
Joseph G. Schoppy

Philadelphia Electric Company

Robert Dickinson
Steve Dietch
R. John Diletto
Tony Gryscavalie
Ed Hosterman
John Hufnagel
Rod Krich
George Licina
Oscar Limpias
Jim Muntz
Glenn Stewart

Other

James Xocher, Conco Services

Final

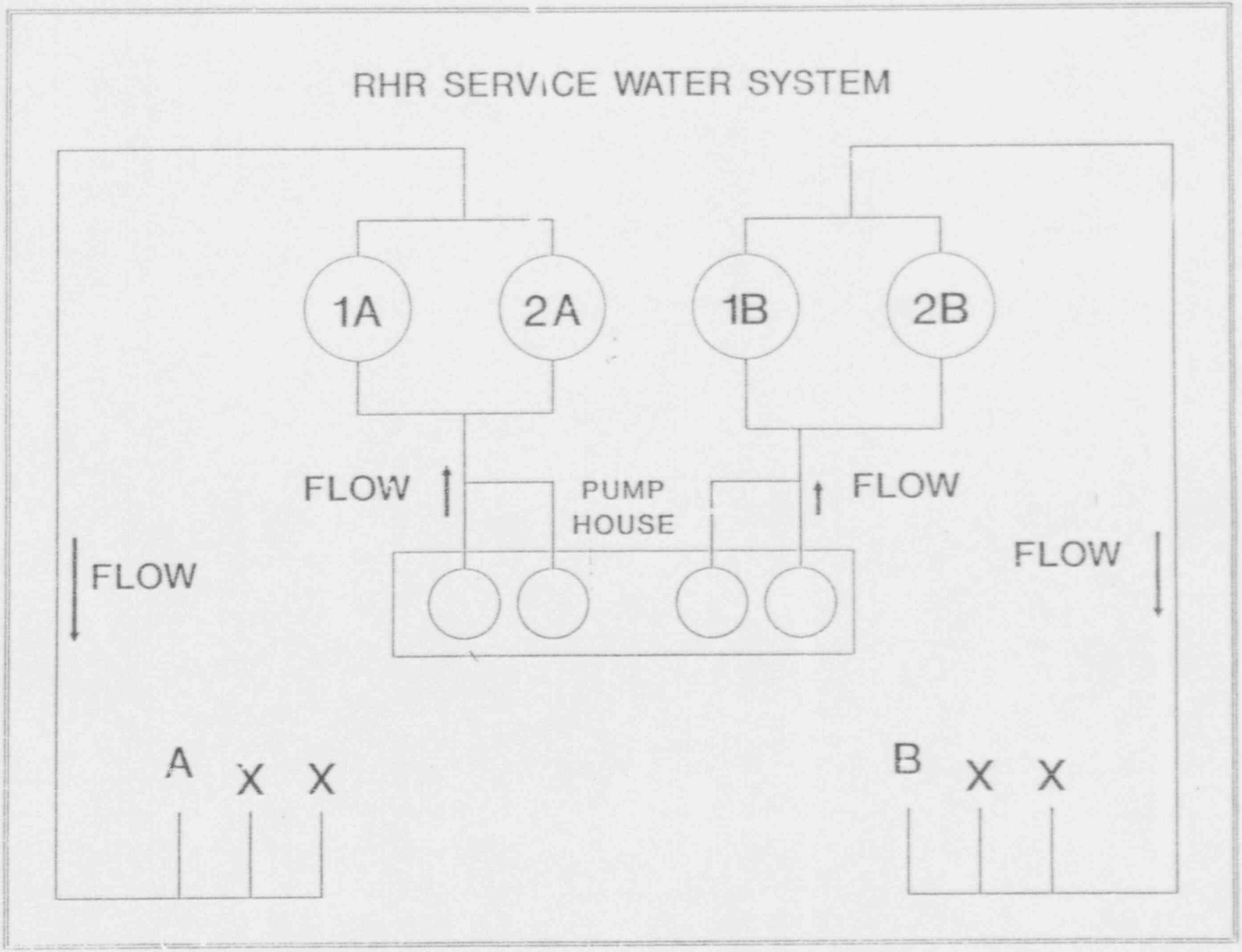
LGS

1A RHR

HEAT EXCHANGER

BOB DICKINSON

6/25/92


RHR SERVICE WATER SYSTEM
1A
2A
1B
2B
FLOW
PUMP
HOUSE
FLOW
FLOW
FLOW
A
X
X
B
X
X

RHR HEAT EXCHANGER

CONSTRUCTION

Shell Side	-	RHR	450 psig rated
Tube Side	-	RHRSW	450 psig rated

530 U-Tubes
- 304L S.S.
- O.D. 1 inch
- Min Thickness .049 inch
- 6281 sq.ft. effective surface area

GL 89-13

" SERVICE WATER PROBLEMS AFFECTING
SAFETY RELATED EQUIPMENT"

REQUIRES HEAT TRANSFER OF ALL SAFETY
RELATED SERVICE WATER HEAT EXCHANGERS

- ESW
- RHRSW

TESTING OF RHRSW

18 MONTH FREQUENCY

FOULING FACTOR CRITERIA BASED ON MOST RESTRICTIVE HEAT REMOVAL CONDITIONS

- SUPP. POOL COOLING MODE
- MAXIMUM SPRAY POND TEMPERATURE

HEAT TRANSFER CALCULATED BASED ON PROCESS AND COOLING WATER SIDE (VALIDATED BY COMPARISON)

INITIAL TESTING

UNIT 1 TESTING PERFORMED APRIL '91

UNIT 2 TESTING PERFORMED JUNE '91

ESTABLISHED CRITERIA FOULING FACTOR < .0025

ALL RESULTS WITHIN ACCEPTANCE CRITERIA (<.0015)

1A RHR HTX. EVENTS

3/18 HEAT TRANSFER TEST - INVALID
MISMATCH BETWEEN HEAT TRANSFER RATES

3/21 UNIT 1 FOURTH REFUEL OUTAGE COMMMENCED

3/27 HEAT TRANSFER TEST - INVALID
S/D CLG. ABILITY PREVIOUSLY DEMONSTRATED

A RHR SYSTEM OUTAGE

5/04 HEAT TRANSFER TEST - VALID FAILURE (.0039)
TESTED WITH ADDITIONAL INSTRUMENTATION

1A RHR HTX. EVENTS (CONT.)

5/08 HI FLOW FLUSH (14000 GPM)

5/08 HEAT TRANSFER TEST - VALID FAILURE (.0039)
INSTRUMENTED TO DETERMINE BYPASS FLOW AND TEMPERATURE STREAMING
PERFORMED AT VARIOUS FLOW RATES

5/23 CHEMICAL CLEANING
30 HOUR TREATMENT TO REMOVE CALCIUM CARBONATE

5/24 HEAT TRANSFER TEST - VALID FAILURE (.0037)

1A RHR HTX. EVENTS (CONT.)

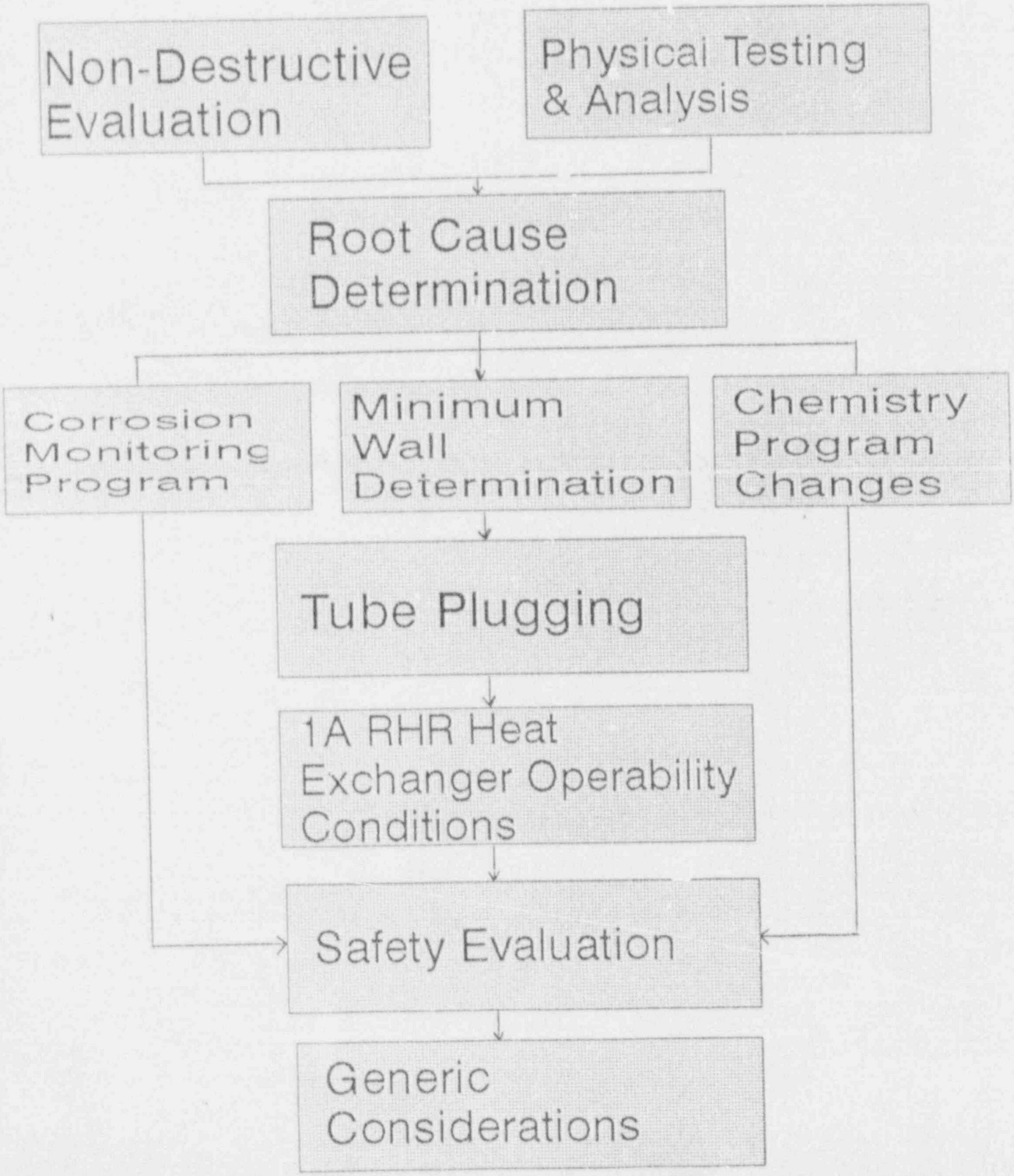
5/27 DISSASSEMBLED

5/29 HYDROLAZED

6/02 EDDYCURRENT TESTED
SIGNIFICANT PITTING SHOWN

Final

ANALYTICAL APPROACH



NON-DESTRUCTIVE (EDDYCURRENT) TESTING

- Initial examination of straight runs (PECo/CONCo)

- EPRI involvement
  - provided expertise for U-bend inspections
  - fabricated/provided calibration standards
  - provided technical guidance

- Examination of tube U-bends

- Re-analysis of straight run data (after re-calibration)

- Pit distribution quantified

PHYSICAL TESTING & ANALYSIS

- Two tube samples removed
- Pitting verificaiton and characterization
- MIC sampling
- Deposit analysis
- Burst pressure testing
- Collapse test
- Tensile stress testing

jgh3.rhr

TUBE PLUGGING CONSIDERATIONS

- Heat exchanger design data
- Heat transfer capability
- Flow considerations
- Structural integrity
- Code allowable limits
- Corrosion allowance

PITTING ROOT CAUSE DETERMINATION

- ROOT CAUSES

- Underdeposit corrosion caused by
  - Manganese deposition caused by oxidizing environment, microbes, and heat transfer surface
  - Oxygen concentration cells under manganese-rich deposits
  - Aggressiveness of environment increased by sodium hypochlorite

- Microbiologically Influenced Corrosion (MIC)
  - Microbes increase aggressiveness of underdeposit environment

MINIMUM WALL DETERMINATION

- Consultation with experts
  - EPRI
  - GE
  - UE&C
  - AEI
  - Bechtel
- Fracture mechanics analysis
- Simplified stress analysis for localized defects
- Calculations for burst/collapse pressures using standard methods
- Destructive Testing
  - burst, collapse tests
  - various pit depths
- Corrosion allowance
  - normal operation
  - post-accident (180 days)
- Conservatism

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CHEMISTRY PROGRAM CHANGES

- Eliminated use of oxidi. [illegible] de
- Eliminated corrosion cells (hydrolazing)
- Increased Spray Pond chemistry monitoring and control
- Implementing use of non-oxidizing biocide
- Evaluating use of dispersant

CORROSION MONITORING PROGRAM

- Simulate actual heat exchanger conditions
- Frequent corrosion rate checks
- Administrative corrosion limit established
- Confirmation of corrosion allowance assumptions

1A RHR HEAT EXCHANGER OPERABILITY CONDITIONS

- Establish improved layup practices
- Implement non-oxidizing biocide treatment during extended pump runs
- Establish chemistry controls for Spray Pond
- Establish procedures for post-LOCA Spray Pond treatment
- Installation/Operation of corrosion monitoring equipment
- Establish appropriate heat transfer test frequency

SAFETY EVALUATION

- Consequences of accidents not affected
- No new accident initiators created
- Probability of HX malfunction not increased
- Consequences of malfunction not increased
- No new malfunctions created
- No reduction in margin of safety

jgh10.rhr

GENERIC CONSIDERATIONS

- Unit 1 heat exchangers operated since early 1984; Unit 2 since late 1988

- Chemistry treatment, layup changes, and corrosion monitoring applied to both units

- 2A RHR heat transfer tested; 2B test pending

- Information drawn from Unit 1 and Unit 2 activities will be appropriately applied to Unit 2 heat exchanger assessment

- Other coolers reviewed for generic implications