U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-352/87-18

Docket No. 50-352

License No. NPF-39

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

Facility Name: Limerick Generating Station, Unit 1

Inspection At: Limerick, PA

Inspection Conducted: July 1 - August 5, 1987

Inspectors: E. M. Kelly, Senior Resident Inspector
S. D. Kucharski, Resident Inspector

Approved by: James C. Linville Jr. 8/25/87
James C. Linville, Chief, Projects Section date

Inspection Summary: Routine daytime (121 hours) and backshift (53 hours including weekends) inspections of Unit 1 by the resident inspectors consisting of: followup on license conditions and outstanding items; walkdown of the CRD system and the emergency diesel generators using PRA guidance; plant tours including fire protection and security measures; observation of refueling activities; maintenance and surveillance observations; evaluation of modifications and outage planning; and review of LERs and periodic reports.

Inspections were conducted that evaluated the results of steam setpoint pressure test failures of the Target Rock SRVs. Events followed up included control rod blade damage to CRD 30-47 on July 3; a Group 1 isolation signal on July 15; dropping of the steam dryer assembly during reinstallation into the reactor vessel on July 23; and the apparent unauthorized exchange of several inverter indicating lamp lens caps on August 2. Meetings were attended onsite during the period, including routine PORC meetings; the SALP report management meeting on July 2; a public meeting for offsite emergency preparedness evacuation plans on July 15; a discussion of Target Rock SRV test failures on July 20; and a tour of Unit 1 by the NRC Region I regional administrator on July 22.

No violations were identified. An unresolved item was initiated for evaluation of the bypass of emergency diesel engine protective trips under accident conditions (Detail 9). The inspectors noted well organized and successfully conducted surveillance testing, as well as good coordination of refueling floor activities by health physics personnel during the inspection period.

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DETAILS

1.0 Principals Contacted

Philadelphia Electric Company

J. Doering, Superintendent of Operations
W. Boyer, Electrical Engineering
R. Dubiel, Senior Health Physicist
G. Edwards, Technical Engineer
J. Franz, St. ion Manager
M. Gallagher, Reactor Engineer
J. Harding, Field Engineer
D. Helwig, Mechanical Engineering
J. Law, Outage Planning
J. Spencer, Superintendent of Services

Also during this inspection period, the inspectors discussed the plant status and operations with other supervisors and engineers in the PECO, Bechtel and General Electric organizations.

2.0 Followup on Unresolved Items

2.1 (Closed) Unresolved Item 87-12-01

The inspector reviewed Revision 7 to Engineering and Research Department Procedure (ERDP) 3.2 for maintaining, amending, and revising the Limerick Unit 1 project Q-List which became effective on July 30, 1987. The procedure was amended to meet commitments by the licensee that were associated with NRC Generic Letter 83-28, in order to administratively assign responsibility for Q-List revisions to the licensee's Mechanical Project Engineer or his designee (including, for example, an outside contracting agency). Revision 7 to ERDP-3.2 also provides for consistency in amending the Peach Bottom and Limerick Q-Lists. The inspector discussed the procedural changes with engineering personnel, concluded that the licensee had met their commitment to control the Unit 1 Q-List, and will assess the control of such changes in future inspections.

3.0 Review of Plant Operations

3.1 Summary of Events

The Unit 1 core remained fully offloaded from June 6 through July 8. Core reloading was commenced on July 9 and completed on July 18. Operational Condition 4, Cold Shutdown, was entered on July 28 following reactor vessel reassembly and head tensioning. An operational hydrostatic test of the reactor coolant pressure boundary at operating pressure was begun on August 3 and remained in progress for the rest of the inspection period.

3.2 Operational Safety Verification

3.2.1 Control Room and Refuel Floor Activities

The inspectors toured the control room daily to verify proper manning, access control, adherence to approved procedures and compliance with technical specifications. The inspectors reviewed shift superintendent, control room supervision, and licensed operator logs and records covering the entire inspection period. On July 29 and 31, the backshift inspections were between the hours of 2:00 a.m. and 6:00 a.m.

The inspectors reviewed logs and records for completeness, abnormal conditions, and significant operating changes and trends. Other records reviewed included: Reactor Engineering and STA books, night orders, radiation work permits, the locked valve log, maintenance request forms, temporary circuit alterations, and ignition source control checklists. The inspectors also observed shift turnovers during the period. Operations activities were observed for conformance with Administrative Procedure A-7. No unacceptable conditions were noted.

The inspectors observed portions of core reload activities and reactor vessel reassembly; attended various outage planning and work control meetings including PORC Meeting No. 87-064 to review plant conditions prior to refueling under Operational Condition 5; periodically verified proper provision of shutdown cooling decay heat removal and refuel cavity water level; and the maintenance of secondary containment refuel floor integrity and neutron monitor operability during core alterations. The inspectors noted well coordinated activities on the refuel floor, particularly by maintenance and health physics personnel, and acceptable housekeeping and loose object control. Proper communication between the refuel bridge and main control room was maintained. No violations were identified.

3.2.2 Security

During entry to and egress from the Unit 1 protected area and vital areas, the inspectors observed that access controls, security boundary integrity, search activities, escorting and badging were all in accordance with Security Plan implementing procedures and guard force instructions. The inspectors also observed the availability and operability of security systems such as search equipment, perimeter detection devices, and security computer alarms. The inspectors verified that the minimum number of armed guards required by the Security Plan to be onsite were present on selected shifts by review of duty rosters, discussion with licensee Shift Security Advisors, and observation of guard force turnovers.

3.2.2.1 Handgun Discovery in Search Train

On July 3, plant security personnel identified a loaded weapon in a gym bag during a routine x-ray search prior to entering the protected area. The weapon was a semi-automatic .380 Astra caliber handgun, fully-loaded, with a round in the chamber and two spare clips of ammunition. The owner was a contract boilermaker who had been badged for unescorted access since 5/29 and was subsequently escorted offsite and turned over to local Limerick Township police authorities. The owner of the gun did not have a permit for its use. The NRC was notified via the ENS on 7/3.

3.2.2.2 Drug Allegations

On July 1, the senior resident inspector was informed by the licensee's Nuclear Security Specialist of a contractor who had been terminated on June 30 for suspected drug use. The contractor was badged on June 9 as a fire watch person. The licensee's security staff had noticed drug paraphernalia (i.e. razor blade) in the contractor's wallet and, upon interviewing on June 30, the contractor admitted to drug use, refused a urinalysis test, and resigned. The contractor was terminated from employment and escorted offsite on June 30.

NRC Region I received an anonymous allegation (RI-A-87-0083) on July 17 concerning drug use by six contract radwaste workers. Five of the six workers were temporarily suspended pending the outcome of urinalysis testing. One worker never returned to work after receipt of the allegation, and was terminated. Two of the five tested workers showed positive results, one for marijuana and one for methamphetamines. The worker with test results of marijuana (badged since 4/87) was terminated; the worker with methamphetamines (badged since 4/86) was retested since the chain of custody for his urine sample was lost. The results of the retest were negative and the worker was reinstated. The employees whose test results were negative were reinstated during the week of July 27, but remain under fitness-for-duty observation by their supervision. The senior resident inspector was informed of the test results on 7/31 and will follow the progress of the licensee's investigation.

An anonymous allegation (RI-A-87-0062) was received by NRC Region I on June 5 concerning a contract employee alleged to use drugs and to be selling fireworks inside of the Unit 1 protected area. NRC Region I management turned over the allegation to licensee security investigators who encountered the employee in question on June 9, patted-him-down (no unusual objects were found), requested that he submit to a urinalysis (which he refused), and subsequently escorted him offsite after he resigned rather than submit to the drug screening test. The inspector observed the licensee's pat-down search, and ascertained that the contractor's badge had been active for approximately six weeks prior to his termination. No further concerns were identified.

3.2.3 Radiological Controls

The inspectors observed the availability and use of radiation monitoring equipment, including portal monitors and portable friskers. The inspectors also observed health physics (HP) supervision and technicians in plant activities involving potentially significant radiological conditions. Radiation work permits (RWPs) were selectively reviewed to determine that appropriate job controls, protective clothing, dosimetry and HP support were prescribed, in use, and understood by workers involved.

Radiological controls for refueling floor were assessed as part of review of RWP-041-87-0530M. Proper surveys and contamination clothing were prescribed. Radiological conditions were discussed with HP technicians and workers who signed-in under the RWP. Because of heat stress considerations, stay-times for refuel cavity work were limited to 60-90 minutes and ice-vests were employed. The inspector had no further concerns, and identified no violations.

3.3 Station Tours

The inspectors toured accessible areas of the plant throughout the inspection period, including: the Unit 1 reactor and turbine-auxiliary enclosures, the main control and auxiliary equipment rooms; battery, emergency switchgear and cable spreading rooms; and the plant site perimeter. During these tours, observations were made of potential fire hazards, radiological conditions, housekeeping, tagging of equipment, ongoing maintenance and surveillance, and the availability of required equipment. No unacceptable conditions were identified.

The inspector observed that the licensee's corporate fire protection engineers toured the Unit 1 power block and walked-down fire protection systems on July 30. The site visit was the first of what are expected to be monthly visits to better coordinate engineering support for various fire equipment and program needs. The fire protection staff has developed a matrix that identified items for corporate attention and possible action, including door improvements, combustible loading evaluations/recommendations, and various equipment improvements.

The inspector also noted that rolling stock throughout the plant has been evaluated for its potential effect on safety related equipment and, for example, 4kV ground trucks used to move replacement breakers have had their wheels pinned to prevent unwanted movement. The inspector noted that, during regular tours of the safeguard switchgear rooms, the subject ground trucks were no closer than two feet away from safety related switchgear. The inspector had no further concerns, and identified no violations.

At the end of the inspection period, the inspector noted the cleanliness and housekeeping levels of safety related battery rooms. In particular, the floor underneath the Division I and II cells was being refinished and readied for fresh painting in response to observations made by the NRC Region I Regional Administrator during his tour of Unit 1 on July 22. No further concerns were identified.

On July 31, the resident inspector discovered what appeared to be a makeshift bed underneath a stairwell on Turbine Enclosure elevation 239. The bed appeared to be used as a break location for concrete masons associated with outage activities, although no use of the bed was observed during regular surveillance through the end of the inspection period. Licensee shift supervision were made aware of the inspector's concerns and also monitored possible use of the bed through the remainder of the period; none was observed. Licensee management removed the makeshift bed shortly after the end of the inspection period.

The inspector also reviewed a revision to a diesel generator engine lubricating instruction for the speed changer/governor. The revision was based on a question concerning potential overfill of the governor oil. The inspector verified that the engine operating manual specified the proper fill of the governor oil. The governor was filled properly but the instruction guide did not relate the correct information. Therefore the licensee corrected the instruction. The inspector had no further questions.

3.4 Engineered Safeguards Features Verification

The inspector independently verified the operability of the control rod drive (CRD) system including hydraulic control unit (HCU) operability by performing a detailed walkdown of the accessible portions of the system, and confirmation of the following items:

-- Review of CRD system Technical Specifications, FSAR, System Operating Procedures and P&IDS.

-- Identification of equipment conditions and items that might degrade performance.

-- System check-off list S47.9 (COL) equipment and operating procedures consistent with plant drawings.

-- Valves and breakers properly aligned, including appropriate locking devices.

-- Instrumentation properly valved in and functional.

-- Control room switches, indications, and controls satisfactory.

-- Surveillance procedures and adequately implement Technical Specification requirements.

Within the scope of the inspection, no unacceptable conditions were noted.

4.0 Onsite Followup of Events

The inspector performed onsite followup of the following events that occurred during the inspection period. The events were evaluated for proper notification of the NRC, reactor safety significance, licensee efforts to identify cause and propose effective corrective action, and verification of proper system design response.

4.1 Control Blade Damage

On July 3, 1987 the double blade guide was removed from control rod 30-47 without first withdrawing the control rod due to a core component transfer authorization sheet (CCTAS) error. The double blade guide was subsequently re-oriented to support LPRM replacement and placed in the cell. The control blade then leaned over and the double blade guide was inserted over the control rod blade.

On July 7, control rod 30-47 was realigned and visually inspected by the licensee. The inspection showed a minor scratch on the fuel support piece and bottom blade wing tips. Friction testing was satisfactorily performed by reactor engineers with a double blade guide in place. The licensee concluded that the probability that a problem will occur because of a bent control rod blade on CRD 30-47 at power which would require the control rod blade to be inserted (and the unit to be degraded) would be small.

Additional friction testing with the four fuel bundles in place after core reload was successfully performed, indicating that the rod is trippable at all times. The inspector discussed the event with the reactor engineer, observed successful CRD 30-47 stroking, and identified no further concerns.

4.2 Group 1 Isolation Signal

A Group 1 isolation signal occurred on July 15. All main steam isolation valves (MSIVs) were closed already for maintenance and leak rate testing. One steam line drain valve was open at the time for leak testing, and closed upon receipt of the isolation signal. The signal was caused when maintenance personnel stroked the main turbine stop valves closed as part of an electrohydraulic control (EHC) system flush. The isolations were reset, fuses in the isolation logic were pulled to prevent recurrence, a PORC-approved procedure was developed to resume the EHC work, and the event was reported to the NRC.

The inspector reviewed the revised procedure to conduct the EHC flushing, discussed the event with the plant maintenance supervision and the operations superintendent, and observed subsequent successful stroking of the stop valves using a temporary circuit alternation to prevent the recurrence of an isolation signal. In a memorandum to the plant manager dated July 17, the operations superintendent evaluated the cause of the event as inadequate work controls by test engineers and licensed operators over mobile maintenance personnel. The inspector had no further concerns, and no violations were identified.

4.3 Steam Dryer Drop

After lowering the steam dryer into the reactor vessel on July 23, the dryer settled and then dropped about 2-4 inches. The drop occurred about 30 seconds after the dryer strongbacks were removed from the lifting assembly. The dryer was fully seated on one of the vessel lugs, partially seated on two of the lugs, and not seated on the remaining fourth lug. The PORC subsequently approved a procedure to lift the dryer and remove it from the vessel, and subsequently reinstalled the dryer successfully on July 25. The inspector discussed the circumstances of the dropped dryer with licensee management, attended a meeting onsite on July 24 to discuss the event, and reviewed an evaluation of the structural damage performed by licensee engineers (See Detail 7.4). No violations were identified.

4.4 Exchanged Inverter Lens Covers

On August 3, the resident inspectors were notified by the licensee's superintendent of operations of the discovery by a nonlicensed operator on his rounds of light covers which had been apparently interchanged on four uninterruptible power supplies in a vital area of the control enclosure. The operator discovered blue and white lens caps interchanged at 11:45 p.m. on August 2, the same operator had previously observed normal light configurations approximately 16 hours earlier. The lights indicate internal and external synchronization references, and do not affect inverter operation. The inverters supply power to the reactor protection system, and the process and security computers.

The licensee's security investigators began review of vital area access records and interview of personnel present in related vital locations on August 2.

On August 4, an additional similar incident was identified to station management concerning breaker position indication light covers for the number 10 Auxiliary Bus switchgear. The lens covers had been switched (green and red) for one breaker, and were missing on another. The condition discovered did not affect the operation of the 13.2 kV switchgear.

The inspector performed walkdowns of safety related equipment areas during the period August 3 and 4 and identified no further similar instances. Plant operators performed similar tours at the end of the inspection period, and security force members conducted vital area sweeps during the same time frame. No further unacceptable conditions were noted, and the inspector will follow the progress of the licensee's investigation.

5.0 Licensee Reports

5.1 In-Office Review of Licensee Event Reports

The inspector reviewed Unit 1 LERs submitted to the NRC Region I office to verify that details of the event were clearly reported, including the accuracy of description of the cause and adequacy of corrective action. Where multiple causes are suspect, or may be different than reported in the LER, this is indicated below. The inspector determined whether further information was required from the licensee, whether generic implications were involved, and whether the event warranted on-site followup. The following LERs were reviewed:

LER Number	Report Date	Cause	Subject
87-21	7/1/87	Personnel error by field engineer	Containment isolations due to a blown fuse during relay replacement
87-22	7/9/87	Procedure deficiency	Testable ring assembly not included in LLRT procedure

LER Number	Report Date	Cause	Subject
87-23	7/15/87	Component failure	Secondary containment isolations and SGTS/RERS initiation due to battery charger failure
87-24	7/13/87	Personnel error by test engineer	Refuel floor ventilation isolation due to lifting an incorrect wire during trouble-shooting
87-25	7/13/87	Personnel error by reactor engineer	Inoperable control rod block logic due to refuel platform limit switch lifting
87-26	7/13/87	Inadequate procedure	Scram signal, MSIV isolation and CRD insertion caused by radiography
87-27	7/15/87	Unknown (could not be duplicated)	Isolation logic signal due to UPS/RPS inverter power supply transfer
87-28	7/10/87	Personnel error by fire watches	Failure to perform hourly compensatory fire watches
87-29	7/16/87	Incomplete design review	Isolation logic signals due to de-energized relays caused by dis-connection of a neutral wire from common connections during a modifica-tion

LER Number	Report Date	Cause	Subject
87-30	7/17/87	Personnel error by ST coordinator	Failure to perform surveillance test of refuel area ventilation radiation monitors
87-31	7/16/87	Personnel error by licensed operators	Hot maintenance shor ventilation operated without operable radiation monitor

5.2 Onsite Followup of Licensee Event Reports

For those LERs selected for onsite followup, the inspector verified that the reporting requirements of 10 CFR 50.73 and Technical Specifications had been met, that appropriate corrective action had been taken, that the event was appropriately reviewed by the licensee, and that continued operation of the facility was conducted in accordance with Technical Specification limits.

5.2.1 LER No. 87-25; Refuel Platform Limit Switches

The inspector discussed the proposed corrective action for the subject LER with licensee representatives, and questioned why a technical specification change had not been proposed as part of the response to LER 87-25. A primary cause of the event was the misleading technical specifications for refueling operations with the reactor core completely offloaded; a condition wherein control rod block logic is insignificant. The inspector also noted that LER 87-25 failed to evaluate the roles of the refuel platform vendor and the senior licensed operator in charge of refueling bridge activities, as well as the potential training deficiency exhibited by a staff engineer for the use of administrative controls for troubleshooting and circuit alterations. The inspector will follow the resolution of the above questions in the licensee's response to NRC inspection item 87-09-01.

5.2.2 LER No. 87-028; Fire Watch Violation

On June 10, between the hours of 6:00 a.m. and 8:00 a.m., during a security computer outage the hourly firewatch inspections of the 254' and 239' elevations of the Control Enclosure were not performed within one hour of the previous inspection. The cause of the event was the failure of the roving firewatch and the security guard to adequately follow their instructions for performing a firewatch during a security computer outage. In the event of a computer outage, the firewatch is unable to enter vital areas to perform firewatch inspections. The roving firewatch had been instructed to proceed to elevation 289 of the control structure and wait for the designated security guard. When the security guard failed to arrive within the designated time specified in the Security Post Orders, the roving firewatch left to continue the inspections in accessible plant areas. The shift firewatch coordinator was informed of the problem and sent a second firewatch to begin inspection on the 200' elevation. The roving firewatch eventually met with the designated security guard, and they were able to complete the required inspections.

In reviewing this event the licensee noticed a discrepancy in time indications between the firewatch log and the security access log. On two occasions the records were falsified to show the inspections were completed within the required time. When asked the firewatch admitted to the falsification. The licensee terminated the firewatch.

In an effort to prevent recurrence of this problem the licensee issued a letter on June 29, 1987, to all contracted firewatch and firewatch supervision outlining the duties of the roving firewatch during computer outages and normal conditions. This information was then discussed with the firewatches during training sessions held on July 6, 1987. Also revised was Security Post Order #39 to shorten the arrival time of the designated security guard to the firewatch point of contact. Also, the assigned guard will sign out a set of vital area keys at the beginning of each shift.

This change in post orders was discussed with the security guards before each shift and a sign off sheet outlining the appropriate steps to be taken during a security computer outage was completed by each guard. Due to the fact that this event has occurred several times in the past the NRC inspectors will be following planned computer outages for future inspection for compliance with the revised procedures.

5.2.3 LER No. 87-031; Maintenance Shop Ventilation

On June 14, at 3:05 a.m., while performing the daily surveillance log surveillance test, a reactor operator discovered that the hot maintenance shop ventilation exhaust radiation monitor had not been functioning and the compensatory sampling had not been initiated as required, the Hot Maintenance shop stopped operations immediately and the ventilation was shutdown. Further investigation identified that the radiation monitor experienced a loss of sample flow on June 13, at 9:13 a.m. and had been out of service since. During this time period there was a continuously running air sampler present in the work area and an analysis of the air sample provided by the monitor showed activity in the shop to be less than the lower limit of detection. Also, there was an Iodine Noble Gas monitor at the entrance of the shop which was operating and did not alarm during the time period involved.

The inspector reviewed the licensee's action to prevent recurrence which included the following:

- A memo from the Operations Engineer to all Operations personnel discussing the importance of following procedures when responding to alarms.

- Plant Procedure RMMS-402 "Determining Monitor Status at the RM-11 Color Console" was revised to provide additional guidance in responding to Hot Maintenance Shop Alarms.

- An operator aid has been installed an RM-11 terminal in the control room which will refer the operator to Procedure RMMS-402 when an alarm is generated.

The inspector had no further questions.

5.3 Review of Periodic and Special Reports

Periodic or special reports submitted by the licensee were reviewed by the inspector. The reports were reviewed to determine that the report included the required information, that test results and/or supporting information were consistent with design predictions and performance specifications, and whether any information in the report should be classified as an abnormal occurrence.

The following reports were reviewed:

-- Monthly operating report for June 1987

-- PECo response dated July 17, 1987, to NRC Inspection Report No. 50-352/87-09

-- Amendment No. 6 to the Technical Specifications, dated July 8, 1987

-- PECo response dated July 2, 1987, to NRC Inspection Report No. 50-352/86-25

The reports were found acceptable.

6.0 Surveillance Activities

6.1 Test Observations

The inspector observed the performance of and/or reviewed the results of the following tests:

-- ST-2-074-600; SRM/IRM Operability

-- ST-2-049-603; RCIC Steamline Pressure Test

-- RT-1-049-331-1; RCIC Turbine Overspeed Trip Test

-- ST-1-LLR-511 and 521; Drywell Chilled Water Leak Rate Tests

-- ST-1-LLR-092 and 084; Feedwater Leak Rate Tests

-- ST-6-047-760; CRD Stroke Timing

-- ST-1-072-103-1; Division III Isolation Logic System Functional Test

-- ST-1-050-101; Division 1 ADS Logic System Functional Test

The tests were observed to determine that surveillance procedures conformed to Technical Specification requirements; proper administrative controls and tagouts were obtained prior to testing; testing was performed by qualified personnel in accordance with approved procedures and calibrated instrumentation; test data and results were accurate and in accordance with Technical Specifications; and equipment was properly returned to service following testing.

No unacceptable conditions were noted.

6.2 Safeguards Bus Logic Testing

The inspectors observed portions of the logic system functional and safety system actuation testing of the 4kV diesel safeguards busses under procedures ST-1-092-111 through 114. The inspectors noted well prepared and coordinated testing, as evidenced by: The number and qualifications of the personnel assigned to perform the tests; the detail and use of an extensive test procedure; and, the successful completion of all tests. Personnel were knowledgeable of annunciated conditions throughout the test, and conducted the testing with minimal impact on plant operations in spite of the complexity of the testing. No unacceptable conditions were noted.

6.3 SRV Testing

All 14 main steam line safety relief valves (SRVs) were bench-tested at Wyle Laboratories from June 25 - July 14. The as-received test results found 10 of the 14 valves in excess of the 1% setpressure range required by technical specifications. Mean setpoint drift was 1.945%. Two valves did not lift within the 1250 psig capacity of the test stand. The tested valves had been installed on Unit 1 from July 1986 through May 1987.

Spare SRVs, which were refurbished at Wyle and had been installed on Unit 1 prior to July 1986, were installed during the inspection period for the next cycle of operation. The SRVs are two-stage Target Rock Model 7567F design. The licensee is preparing an LER on the test failures. The Resident Inspectors attended a July 20 PORC meeting to address the cause of the failures and implications for Cycle 2 operation. The cause is currently under evaluation by a BWR owners group and is thought to be due to a stellite corrosion phenomenon on the pilot disc. The inspectors will review the results and conclusions of the licensee in the forthcoming LER.

6.4 PSA-10 Snubbers

During the performance of snubber inspections and testing, the licensee has observed that a significant percentage of size PSA-10 snubbers exhibited acceleration rates between 0.02 and 0.04 g. The acceptance criteria is 0.04g, maximum, for snubbers located in nonsensitive areas and 0.02g, maximum, for snubbers located in sensitive areas. Of the 86 PSA-10 snubbers having acceptance criteria of 0.02g maximum 21 failed. Based on a review performed by the licensee similar problems were discovered at other plants. In 1985, 43 of the 84 PSA-10 snubbers tested at Peach Bottom did not meet the acceptance criteria of 0.02g, but all were less than 0.04g. Additionally, 31 of the 161 PSA-10 snubbers tested during the Susquehanna Unit 1 second refueling outage were found to have high acceleration values between 0.02g and 0.04g. The cause of this problem is excessive grease in certain internal parts. Of the 21 failed snubbers at the Limerick site 15 snubbers were disassembled and cleaned to remove the excess grease, retested and found acceptable. The remaining 6 required replacement of the Capstan spring and clutch spring in order to meet the acceptance criteria. The unacceptable conditions were noted.

7.0 Maintenance

The inspector observed selected maintenance activities on safety related equipment to ascertain that: the work was conducted in accordance with approved procedures; proper equipment permits and tagging were administratively controlled; craft performing the work were appropriately qualified and supported; and return-to-service of equipment included adequate post-maintenance testing and operational verification.

7.1 Work Observation

Portions of the following work activities were observed or reviewed:

-- MRF 878-1596 and 1597, Core Spray Check Valve repairs

-- MRF 87-1455, Tensioning of Reactor Head Studs

-- MRF 878-1305; EQ maintenance for HCU-1451

No violations were identified.

7.2 CRD EQ Maintenance

The inspector observed post-maintenance testing under special procedure SP-S-056 to return the control rod drives (CRD's) and their associated hydraulic control units (HCU's) to operability following the rebuilding of 20 CRD's (discussed in NRC Inspection No. 50-352/87-13) and environmental qualification (EQ) preventive maintenance performed for 34 HCU's. The EQ maintenance was controlled under an extensive procedure, PMQ 500-036, which governed the use of five separate procedural tasks. Interviews with maintenance personnel who supervised or performed the EQ work, and observations of work in progress, indicated that progress of the job was slower than anticipated due to the cumbersome nature of the procedures. Extensive QC coverage was provided the EQ work, as was HP support. The inspector reviewed maintenance data record forms for HCU 14-51 work performed under MRF 87-1305. Work scope included scram valve air solenoid overhauls and replacement of seals and other components. Preplanning and job preparation included practice on a spare HCU in the maintenance shop. The licensee plans to more efficiently approach similar future work and is expected to critically review the lessons learned from the HCU-EQ maintenance after startup from the outage. No violations were identified.

7.3 Reactor Vessel Reassembly

The inspectors witnessed portions of the below listed refueling activities associated with reassembly of the reactor vessel internals and head tensioning under the following maintenance (M-041) procedures:

-- 024; Reactor Vessel Assembly

-- 025; Steam Separator Installation

-- 026; Steam Line Plug Removal

-- 027; Steam Dryer Installation

-- 028; Reactor Head Stud Installation

-- 029; Reactor Head Installation

-- 030; Reactor Head Stud Tensioning

-- 031; Mirror Insulation Installation

-- 032; Reactor Cavity Shield Plug Installation

No violations were identified.

7.4 Steam Dryer Settling

On July 23, 1987, at approximately 7:30 p.m., during installation of the steam dryer, the dryer settled or dropped three to four inches within 30 seconds after the removal of the strongback lifting assembly. A visual examination using the vessel flange and the top of the dryer lifting lugs as reference, showed the dryer had landed unevenly with the low point being at the lifting lug situated between the 0 and 90 degree dryer support lugs and the high point being the opposite side between 180 and 270 degree dryer support lugs. The licensee generated a work instruction to address the removal and inspection of the dryer and vessel. The inspection also included monitoring the load being lifted by the crane during removal.

When the dryer was removed, the low dryer lifting lug located between the 0 and 90 degree azimuth rose 3" to 4" before the dryer assembly proceeded out of the vessel in a level position which indicated the dryer being in a slanted position. There was no indication of excessive crane load during removal which indicated the dryer was not stuck. During the removal, the four seismic brackets on the dryer were inspected and minor scrapes were found. Once the dryer was removed a visual inspection of the top of the separator, the guide rods and dryer indicated only one area of minor damage. At the 270 degree support lug, right hand side where the vertical face meets the horizontal place, a sliver approximately 1/16" wide by 5/8" long was observed remaining attached to the support lug.

The root cause of this event was inadequate procedures. The installation procedure did not require a visual confirmation that the dryer guide lugs are at the proper positions around the guide rods while the dryer is being lowered into the vessel. Review of available brackets indicate there is approximately 0.3" to 0.4" minimal radial clearance in this area. Inspection during the removal of the dryer indicated the dryer was positioned hard against the 180 degree guide rod, which impeded the proper setting of the dryer on the support lugs.

The licensee's installation procedures have been changed to include visual observation and measurements to ensure the centering of the dryer guide lugs in relation to the guide rods during installation. The inspector had no further questions.

8.0 Plant Modifications

The following modifications were evaluated to assess, in part, the: details and adequacy of the safety evaluation; proper consideration of Technical Specification changes; implementation under Administrative Procedure A-14; the status of completion of physical installation; effectiveness of modification acceptance testing; and, accurate update of operating and test procedures, as-built drawings, and operator training programs. The inspector verified that appropriate engineering design support and PORC review and approval were received; that Construction Division installation was in accordance with ERDP procedures including appropriate QC coverage and with a minimal effect on plant operations; and, that an operable system was returned to service with no apparent unreviewed safety questions. Within the scope of this inspection, no violations were identified.

8.1 SGTS Tie-In To Refueling Floor Zone

Modification (MDCP)-614 is intended to provide larger fans and new ductwork for the Standby Gas Treatment System (SGTS) to tie-in the systems to the refueling floor ventilation zone. Limerick License Condition No. 14 requires completion and test of modifications required to connect the refueling floor volume to the SGTS.

The inspector discussed the current status of the SGTS modification with the responsible test engineer assigned to coordinate the completion of Revision 4 to MDCP-614. New, higher capacity 8400 cfm fans have been installed and preliminary run-in at required design flows. An extensive modification acceptance test (MAT) has been developed to determine appropriate flow balancing, damper operation and SGTS logic system functioning. As of the end of the inspection period, licensee field engineers were performing final control system checkouts. The inspector noted that few difficulties have been experienced with implementation of MDCP-614, in spite of its complexity and extensive scope, due in part to the dedication of test engineers and a modification coordinator. The coordinator chairs weekly planning meetings with various site work groups involved with MDCP-614. The inspector noted that Amendment No. 6 to the technical specifications for the new SGTS configuration was issued during the inspection period, and is under review by station management prior to implementation. The inspectors will follow the completion of MDCP-614, including the final drawdown test under procedure ST-1-076-310, to demonstrate acceptable post-modification testing.

8.2 SRV Solenoid Replacements

Modification 86-5246 replaced the solenoid valve assemblies on all 14 mainsteam safety relief valves (SRVs) with dual three-way Target Rock solenoids. The new solenoids are capable of withstanding higher instrument gas pressures of up to 250 psi. The inspector reviewed the safety evaluation for MDCP-5246, discussed the modification with cognizant maintenance and construction engineers, and reviewed completed work packages. The inspector also ascertained that internal lubricants were properly considered in the initial installation of the solenoids, and had no further concerns.

8.3 RRCS Rework

The inspector observed portions of the post-modification acceptance testing of MDCP-0805 to revise programming of the redundant reactivity control system (RRCS) logic. Changes to RRCS include decreasing the recirculation pump trip time delay to 9 seconds, and increasing the time delay for automatic initiation of the standby liquid control system to 120 seconds. The inspector also reviewed the safety evaluation for MDCP-0805, discussed the changes with I&C engineers and technicians, and observed that a thorough PORC review was completed. The acceptance testing for the RRCS modifications was observed to be well staffed and supported, and carefully conducted. No violations were identified.

9.0 Exit Meeting

The NRC resident inspectors discussed the issues in this report throughout the inspection period, and summarized the findings at an exit meeting held with the Station Manager, Mr. John Franz on August 4, 1987. At the meeting, the licensee's representatives indicated that the items discussed in this report did not involve proprietary information. No written inspection material was provided to licensee representatives during the inspection period.