U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-440/88009(DRP)

Docket No. 50-440/88009 License No. NPF-58

Licensee: Cleveland Electric Illuminating Company
Post Office Box 5000
Cleveland, OH 44101

Facility Name: Perry Nuclear Power Plant, Unit 1

Inspection At: Perry Site, Perry, Ohio

Inspection Conducted: April 20 through June 30, 1988

Inspectors: K. A. Connaughton

G. F. O'Dwyer

RC Knop for

Approved By: Richard Cooper, Chief
Reactor Projects Section 3B

7/21/88
Date

Inspection Summary

Inspection in April 20 through June 30, 1988 (Report No. 50-440/88009(DRP))

Areas Inspected: Routine, unannounced inspection by resident inspectors of previous inspection items, operational safety, nonroutine events, maintenance, surveillance, engineered safety features, containment closeout, Licensee Event Reports, onsite review committee activities, physical security, and radiological controls.

Results: Of the 11 areas inspected, one violation was identified in one area (failure to identify entry into a technical specification LCO as required by procedure - Paragraph 5). Seven unplanned reactor scrams occurred during this inspection, including three which resulted from operating personnel error. Initial followup inspection activities were conducted to verify identification and correction of root causes prior to facility startup after each scram. Additional inspector reviews will be conducted following issuance of Licensee Event Reports associated with the scrams. NRC Region III management met with licensee management on June 29, 1988, at the Perry site to discuss the recent negative trend in operating personnel errors and licensee initiatives to turn around this trend.

DETAILS

1. Persons Contacted

   a. Cleveland Electric Illuminating Company

      +Alvin Kaplan, Vice President, Nuclear Group
      C. M. Shuster, Director, Nuclear Engineering Department (NED)
      *+M. D. Lyster, General Manager, Perry Plant Operations Department (PPOD)
      +R. A. Stratman, Manager, Operations Section, (PPOD)
      *H. N. Kelly, Acting Senior Operations Coordinator (PPOD)
      *D. R. Green, Manager, Electrical Design Section (NED)
      +V. K. Higaki, Manager, Outage Planning Section (PPOD)
      +M. Cohen, Manager, Maintenance Section (PPOD)
      +F. R. Stead, Director, Perry Plant Technical Department (PPTD)
      +W. R. Kanda, Manager, Technical Section (PPTD)
      +S. F. Kensicki, Technical Superintendent (PPTD)
      L. L. Vanderhorst, Radiation Protection Section (PPTD)
      +E. M. Buzzelli, Manager, Licensing and Compliance Section (PPTD)
      *G. R. Dunn, Supervisor, Compliance (PPTD)
      +R. A. Newkirk, Manager, Technical Section (PPTD)
      S. J. Wojton, Manager, Radiation Protection Section (PPTD)
      +E. Riley, Director, Nuclear Quality Assurance Department (NQAD)
      *W. E. Coleman, Manager, Operations Quality Section (NQAD)
      T. A. Boss, Supervisor, Quality Audit Unit (NQAD)
      D. J. Takas, Manager, Mechanical Maintenance Quality Section (NQAD)

   b. U.S. Nuclear Regulatory Commission

      +R. C. Knop, Chief, Projects Branch 3
      +R. W. Cooper, II, Chief, Projects Section 3B
      *+K. A. Connaughton, Senior Resident Inspector
      *+G. F. O'Dwyer, Resident Inspector

   *Denotes those attending the exit meeting held on June 30, 1988.
   +Denotes those attending the June 29, 1988 plant status meeting.

2. Licensee Action on Previous Inspection Findings (92701 92702)

   a. (Closed) Open Item (440/86999-01(DRSS)): Unusual Event declaration following January 31, 1986, earthquake. This item was entered into the NRC Region III open item tracking system to track followup inspection activities by NRC emergency preparedness specialists. Subsequently, the NRC Inspection Manual was revised to require emergency preparedness inspector reviews of all events which resulted in activation of the licensee's emergency plan. These reviews were to be conducted during routine, periodic emergency preparedness inspections. This item therefor serves no useful purpose and is hereby administratively closed.

   b. (Closed) Open Item (440/86999-02(DRSS)): Unusual Event declaration due to offgas system charcoal adsorber combustion event. The basis for closure of this item is identical to that cited for Open Item (440/86999-01(DRSS)) in Paragraph 2.a. above.

c. (Closed) Open Item (440/86999-03(DRSS)): Charcoal adsorber reignition following suspension of nitrogen purge. The basis for closure of this item is identical to that cited for Open Item (440/87999-01(DRSS)) in Paragraph 2.a. above.

d. (Closed) Open Item (440/87999-01(DRSS)): Unusual Event declaration due to high pressure core spray system automatic initiation and injection following a loss of feedwater and reactor scram. The basis for closure of this item is identical to that cited for Open Item (440/86999-01(DRSS)) in Paragraph 2.a. above.

e. (Closed) Open Item (440/87999-02(DRSS)): Alert declaration following momentary loss of control room annunciators. The basis for closure of this item is identical to that cited for Open Item (440/86999-01(DRSS)) in Paragraph 2.a. above.

f. (Closed) Violation (440/87016-02(DRP)): Failure to follow procedure during main steam isolation valve (MSIV) rework resulting in excessive MSIV leakage. This violation was issued against 10 CFR 50, Appendix B, Criterion V, the licensee's QA Plan, and Plant Administrative Procedure (PAP)-0905. PAP-0905 was cited because, at the time the Notice of Violation was issued, the inspector believed that inconsistencies between the MSIV rework Job Traveler (work instruction) and the work actually performed stemmed from a failure to revise the Job Traveler to specify that interface with Power Cutting Incorporated (PCI), a contractor brought in to grind the seat of MSIV 1B21-F028B, was required and to specify how the interface was to be accomplished. A requirement to make such a revision when work was to be performed by outside entities was contained in PAP-0905. The violation further stated that assurances of valve seating surface acceptability originally specified in the Job Traveler were not required to be performed following grinding of the valve seat by PCI. Valve seating surface irregularities therefore went undetected.

By letter dated January 8, 1988, the licensee responded to the Notice of Violation (NOV) and indicated that the requirement of PAP-0905 cited in the NOV was intended to apply to outside entities performing work under their own approved quality assurance program and procedures. Since PCI was brought in to perform work under the licensee's direct supervision and in accordance with the licensee's quality assurance program and procedures, the licensee took the position that the cited requirement of PAP-0905 did not apply. The licensee stated that the cause of the excessive MSIV leakage was apparent insufficient machining of the valve seat resulting from a failure to recognize that the seat contact point would move deeper into the seat as the MSIV was subsequently cycled. Corrective actions specified in the licensee's response included relapping the valve seat in accordance with requirements contained in the original Job Traveler in order to remove the identified defects, and development of a General Maintenance Instruction (GMI) to incorporate lessons learned during the MSIV rework.

Inspector review of the licensee's January 8, 1988 response letter determined that the licensee's position regarding the intent and applicability of PAP-0905 was reasonable. However, with regard to 10 CFR 50, Appendix B, Criterion V, and Section 5 of the licensee's QA Plan, the violation stood as written. Personnel performing the rework did not require/perform valve seat lapping and verification of valve seating surface acceptability originally specified in the Job Traveler (failure to follow procedure). Alternatively, the licensee did not revise the Job Traveler to address grinding of the valve seat by PCI (prescribe the activity by a procedure appropriate to the circumstances).

Specifically, the Job Traveler required: the valve seat be covered with a blue dye prior to each lapping period; lapping and polishing with No. 180 and, if necessary, No. 400 abrasive paper; removal of all blue dye during lapping as evidence that the valve seat was true and polished and; achievement of a defect-free valve seat with an approximate 32 RMS finish. Finally, QC inspection was to be performed, including measurement and documentation of the valve seat angle. According to the Job Traveler, these were the last activities to be performed affecting MSIV main seat surface quality prior to the performance of a blue check and MSIV local leakrate test.

Contrary to the Job Traveler, the PCI tool was a grinding tool which utilized a grinding stone as opposed to the abrasive paper specified in the Job Traveler and as utilized by the original lapping tool. According to licensee personnel interviewed by the inspectors, blue dye was not applied to the valve seat and verified completely removed by the PCI tool in order to ensure the specified seat finish and trueness were achieved by the grinding process. A defect-free seating surface was not achieved. The seat angle was not documented following the last QC inspection of the valve seat (Step 32. of Attachment 1A to the Job Traveler, Revision 5). Following the grinding process and prior to the blue check and local leakrate testing, originally specified seat lapping and polishing was not performed.

The inspectors discussed the January 8, 1988 response letter with licensee personnel on several occasions and expressed the concern that, notwithstanding the apparent inapplicability of the PAP-0905 requirement cited in the NOV, licensee personnel should have recognized the need to revise the Job Traveler in order to reconcile Job Traveler requirements with what they intended to accomplish utilizing the services of PCI. Based upon these discussions, the licensee agreed to submit a revised response to the NOV and did so by letter dated March 18, 1988. Inspector review of the revised response determined that it was responsive to the inspector's concern. In addition to the corrective actions specified in the original response, the licensee committed to revise Maintenance Administrative Procedure (MAP)-0203, "Conduct of Maintenance" to provide additional guidance to maintenance supervisors on the need to revise a Job Traveler when work scope or job conditions are altered.

The inspectors verified by review of GMI-0056, "MSIV Disassembly, Repair and Reassembly Instructions", Revision 0, dated February 9, 1988, Temporary Change Notice (TCN-3) to MAP-0203, dated May 2, 1988, and documentation of training conducted on April 19, 1988 that the licensee had satisfactorily completed corrective actions specified in the licensee's response letters. The inspectors have no further concerns regarding this matter.

3. 10 CFR Part 21 Report Followup (92700)

(Closed) 10 CFR Part 21 Report (440/88002-PP): Defective upper connecting rod bearings for General Motors (GM) Electro Motive Division (EMD) 645 engines in nuclear service. The subject 10 CFR 21 report was submitted by letter dated April 14, 1988 by Morrison Knudson Company Inc., the exclusive worldwide distributor for GM EMD diesel engines for nuclear standby service. In January and February 1988, a limited number of Clevite manufactured upper connecting rod bearings were replaced. These bearings, which would have been shipped to licensees in either February or March 1988, were determined not to be acceptable for use. The inspectors verified through discussions with licensee personnel that the licensee had been informed of the problem and that eight of the suspect bearings had been received by the licensee. The bearings had not been installed by the licensee. On June 6, 1988, the licensee issued Nonconformance Report PRCS-3345 to identify the suspect bearings as nonconforming. The bearings were segregated and placed on hold in the licensee's warehouse facilities. The inspectors have no further concerns regarding this matter.

4. Operational Safety Verification (71707)

The inspectors observed control room operations, reviewed applicable logs, and conducted discussions with control room operators during this inspection period. The inspectors verified the operability of selected emergency systems and verified tracking of Limiting Conditions for Operation associated with affected components. Tours of the intermediate, auxiliary, reactor, and turbine buildings were conducted to observe plant equipment conditions including potential fire hazards, fluid leaks, and excessive vibrations, and to verify that maintenance requests had been initiated for certain pieces of equipment in need of maintenance. The inspectors by observation and direct interview verified that the physical security plan was being implemented in accordance with the station security plan. The inspectors observed plant housekeeping/cleanliness conditions and verified implementation of radiation protection controls.

At various times from May 1 to 4, 1988, the inspectors selected from the tag-out records for the previous 60 days the following tagouts for the High Pressure Core Spray (HPCS) System:

1-88-1280	1-88-1282	1-88-1284	1-88-1263
1-88-1281	1-88-1283	1-88-1285	1-88-1264

The inspectors then verified that equipment tagged out of service had been properly returned to service.

These reviews and observations were conducted to verify that facility operations were in conformance with the requirements established under technical specifications, 10 CFR, and administrative procedures.

No violations or deviations were identified.

5. Followup of Nonroutine Events at Operating Power Reactors (93702)

a. General

For each of the events discussed in Paragraphs 5b through 5i below, the inspectors performed onsite followup inspection activities to gather factual information, to assess the events for safety significance, and to evaluate diagnostic and remedial actions taken by the licensee in response to each of the events. As applicable, these followup inspection activities included interviews with licensee personnel; review of operating, maintenance, and surveillance test records; pertinent plant data; documented licensee event evaluations; and, associated corrective action documentation.

b. April 27, 1988 Loss of Feedwater and Reactor

On April 27, 1988, at approximately 10:05 p.m., while operating at 100% power, non safety-related 120 A.C. Electrical Bus V-1A experienced an electrical transient as a result of a switching error which resulted in Inverter DB-1-A (normal supply) being supplied solely by Reserve Battery Charger FD-12A. As a result of the electrical transient, the hot surge tank level controller output dropped to zero and hot surge tank level control Valve 1N21-F230 went fully closed. Four minutes later, the feedwater booster pumps tripped on low hot surge tank level and the main feedwater pumps, which are interlocked with the feedwater booster pumps, also tripped. The reactor subsequently scrammed on low reactor water level and, as reactor water continued to decrease, the Reactor Core Isolation Cooling (RCIC) system initiated and began injecting water to the reactor vessel. The high pressure core spray system was out of service for planned maintenance at the time of the event and therefore did not actuate. The RCIC system restored reactor water level to within normal limits approximately 25 minutes after its initiation. The minimum reactor water level achieved (wide range) was approximately 88 inches above the top of active fuel. The electrical switching error was corrected and reactor water level control utilizing the motor driven feedwater pump was established at approximately 10:50 p.m. The Senior Resident Inspector arrived at the site at approximately 12:50 a.m., April 28, 1988 to verify that plant conditions were stable, to obtain factual information regarding the event, and to observe initial licensee investigative efforts. Prior to return to power operation, the inspector verified that the licensee had completed an evaluation of hot surge tank level controller response, inspections of affected electrical loads, and functional testing of Inverter DB-1-A.

c. May 6, 1988 Reactor Scram

On May 6, 1988, at approximately 2:37 p.m., while operating at 100% power, a reactor scram occurred. At the time of the event, reactor vessel steam dome pressure high Channel "A" functional testing was being performed. Per the test procedure, a test signal was inserted which resulted in an RPS channel A/C half scram. Due to a blown fuse in the RPS power supply circuit for the Control Rod Group 3B solenoids, the RPS channel A/C half scram resulted in the scramming of all Group 3 control rods. Reactor level decreased as a result of void collapse and, five seconds later, a full scram was received due to low reactor water level. Following the reactor scram, all systems functioned per design. Followup investigation by the licensee disclosed that per the test procedure, all scram pilot solenoid status lamps were verified to be lit prior to performance of the surveillance test. The blown fuse was replaced and the surveillance test was reperformed with satisfactory results.

During the reperformance, the licensee monitored voltage on the Control Rod Group 3B solenoid power supply circuit and did not detect any perturbation that would link the blown fuse to performance of the surveillance test. Following these investigative efforts a reactor startup was commenced on May 7, 1988. While not identified as a startup restraint, the licensee intended to perform a further analysis of the blown fuse to establish the cause of failure.

d. May 15, 1988 Declaration of Unusual Event

On May 15, 1988, at approximately 4:46 p.m., while operating at 100% power, the licensee discovered that a compressor guide vane contr linkage on the "B" control complex chilled water chiller was broken. Though the "B" chiller was in service and, for the time being, performing adequately, it was declared inoperable. This resulted in the "B" train of the Control Room Emergency Recirculation System (CRERS) also being declared inoperable. At 5:15 p.m., the "A" train of the CRERS was declared inoperable following the discovery of a blown fuse associated with the "A" train supply fan motor. Technical Specification 3.0.3 was entered at that time.

Based upon the loss of both CRERS trains and the impending shutdown required by Technical Specification 3.0.3, an unusual was declared at 5:50 p.m. By 8:05 p.m., with reactor power reduced to approximately 86%, the "B" control complex chilled water chiller was repaired, Technical Specification 3.0.3 was exited and, a return to 100% power was commenced. Following fuse replacement and troubleshooting of the "A" train supply fan, the unusual event was terminated at 9:50 p.m. All emergency plan notifications were carried out satisfactorily.

e. May 18, 1988 Reactor Shutdown

On May 18, 1988, at approximately 12:35 a.m. edt, while operating at 100% power, a float type level controller failed causing the main generator hydrogen seal oil system to back up, dumping seal oil into the main generator casing. At approximately 4:00 a.m., the licensee began an orderly shutdown to inspect the main generator with particular attention to the generator bushings. Previously, in June 1987, main generator bushings overheated to failure as a result of seal oil blocking the flow of hydrogen coolant to the bushings. Licensee investigation determined that a weld failure on the float of the level controller had resulted in the float filling with seal oil. Additionally, the licensee discovered a small stator cooling water system leak during inspection of the main generator. The generator bushings were undamaged. Following repairs to the float trap level controller and stator water cooling system, the licensee restarted the plant on May 30, 1988.

f. June 5, 1988 Reactor Scram

On June 5, 1988, at approximately 13:25 a.m. edt, while operating at 80% power with reactor recirculation flow control in flux manual, an operator attempted to increase recirculation system flow by manually raising flux controller demand output. The operator inadvertently bumped the flux auto pushbutton causing the flow control system to shift to the flux automatic mode. Due to an existing deviation between master controller output and flux controller output, the recirculation system flow control valves received an opening signal which caused them to open an additional 20% in approximately three seconds. This resulted in a neutron flux increase and reactor scram on Average Power Range Monitor (APRM) high flux. All systems functioned per design following the scram. The licensee proceeded to cold shutdown in order to effect repairs to a leakage collection shroud on Valve 1B21-F0559A. Cold shutdown was achieved on June 5, 1988 at approximately 4:29 p.m. edt. The licensee completed a post scram evaluation and repair activities and commenced a plant startup on June 6, 1988. In the future, recurrences of this event will be prevented because the automatic controller card for the flux controller will be removed until the card is properly tuned. When the card is tuned, a portion of the circuitry of the card will prevent a transfer into the automatic mode if the existing deviation error is large enough to cause a problem. Furthermore, operating procedures will be amended to require operators to maintain the deviation between the manual and automatic signals to the flux controller at zero.

g. June 8, 1988 Reactor Scram

On June 8, 1988, at approximately 9:23 a.m. edt, while operating at 23% power, the reactor scrammed following a loss of power to the scram pilot solenoids. The power loss resulted from inadvertent deenergization of non safety-related 13.8KV electrical Buses L11 and L12. Prior to the occurrence, the normal and alternate moisture separator/reheater (MSR) 2A drain valves failed closed, the main turbine tripped on high water level in MSR 2A, and the main generator tripped on reverse power. Balance-of-Plant house electrical loads automatically transferred from the unit auxiliary transformer to the startup transformer per design.

Following the transfer, an operator intending to place the control switches for the Bus L11 and L12 supply breakers from the unit auxiliary transformer in the open position inadvertently manipulated the control switches for the Bus L11 and L12 supply breakers from the startup transformer, deenergizing Buses L11 and L12. In addition to the reactor scram, this resulted in a loss of feedwater to the reactor. The RCIC system was manually started and water level was restored to the normal operating range. The minimum reactor water level achieved was approximately 145 inches; above the automatic initiation setpoint for the RCIC and HPCS systems.

By 9:55 a.m. edt, power was restored to Buses L11 and L12. Meetings between licensee management and personnel from all operating crews were being held to discuss this and other recent personnel error-related events. Following these meetings and completion of repairs to the MSR 2A drain valves, the licensee performed a plant startup on June 9, 1988.

h. June 16, 1988 Reactor Scram

On June 16, 1988, a reactor recirculation system flow increase resulted in a reactor scram when the average power range monitor neutron flux-high setpoint was exceeded. The transient was initiated when instrumentation and control technicians reinstalled the flux-auto controller printed circuit card into the reactor recirculation flow control system. At the time of the event, the recirculation flow control system was operating in the flux manual mode and perturbation to the manual flow control signal was not expected. Subsequently, the licensee contacted Foxboro, the control system equipment vendor, and was informed of the possibility of inducing a noise spike as a result of momentary improper grounding during reinsertion of the printed circuit card. With the reactor in cold shutdown, this root cause hypothesis was confirmed by repeating the evolution several times while monitoring recirculation flow control system signal output.

In order to prevent future similar occurrences, the licensee will take the following corrective action: (1) the recirculation flow control system will not be operated in the flux-auto mode until the control system is retuned; (2) prior to any future removal and reinsertion of any of the recirculation flow control printed circuit cards the hydraulic power units for the flow control valves will be locked up to prevent unexpected recirculation flow transients. Additionally, based upon discussions with Region III management, the licensee agreed to alert Foxboro to the fact that other Foxboro customers may experience similar problems since the Foxboro vendor manuals do not acknowledge the potential for this type of occurrence. By letter dated June 24, 1988, Foxboro notified all nuclear plants of the potential for similar problems and advised each licensee to evaluate plant-specific applications for potential reportability under 10 CFR 21. The facility was restarted on June 22, 1988.

i. June 23, 1988 Turbine Trip/Scram

On June 23, 1988, at approximately 9:45 a.m. edt, while operating at approximately 84% power, a main turbine trip occurred resulting in a reactor scram. All systems functioned per design following the scram. Approximately nine seconds prior to the occurrence, the licensee had completed an automated test sequence on the main turbine mechanical overspeed trip mechanism. By design, the mechanical overspeed trip mechanism is exercised with the trip function locked out. At the end of the test sequence, the overspeed trip mechanism is relatched and the trip function reinstated. During this particular test performance, control room instrumentation indicated that the test sequence was satisfactorily accomplished. Following completion of a post-scram evaluation and inspection of equipment associated with the trip mechanism, the licensee commenced a reactor startup with the intention of performing mechanical overspeed trip mechanism troubleshooting at low power with the main generator disconnected from the grid. Prior to achieving criticality, licensee personnel discovered that the mechanical overspeed trip latch rod was deformed and that the trip mechanism could not be latched. The reactor startup was terminated and the licensee proceeded to cold shutdown in order to determine why the trip latch rod was damaged and to effect repairs. Since the vendor had failed to specify and ensure proper installation tolerances, the trip latch rod was subjected to excessive wear. The trip latch mechanism was replaced with proper tolerances established. The licensee returned to power following satisfactory post-maintenance testing on June 27, 1988.

6. Monthly Maintenance Observation (62703)

Station maintenance activities of safety-related systems and components described below were observed/reviewed to ascertain that they were conducted in accordance with approved procedures, regulatory guides and industry codes or standards, and in conformance with Technical Specifications. The following items were considered during these reviews: the limiting conditions for operation were met while components or systems were removed from service; approvals were obtained prior to initiating the work; activities were accomplished using approved procedures and were inspected as applicable, functional testing and/or calibrations were performed prior to returning components or systems to service; quality control records were maintained; activities were accomplished by qualified personnel; parts and materials used were properly certified; and radiological controls were implemented.

Work requests were reviewed to determine the status of outstanding jobs and to assure that priority was assigned to safety related equipment maintenance which may affect system performance.

From May 9 to 11, 1988, Limitorque maintenance on Valve 1M16-F010B ("B" isolation valve for the Drywell Vacuum Relief System) in accordance with Work Order (WO) 88-2441, Revision 1, was observed/reviewed.

Following completion of maintenance on valve 1M16-F010B, the inspector verified that this system had been returned to service properly.

From June 13 to 30, 1988, the inspector observed/reviewed WO-88-3953 which documented troubleshooting performed on the "A" Control Complex chiller.

The reason for the troubleshooting was documented by unit log entries on June 12 which stated that at 6:12 p.m., Chiller "A" was put into operation and at 8:20 p.m., it was determined that Chiller "A" was not chilling water, outlet temperature was 72 degrees F (normal temperature was 59 degrees F) and the chiller apparently needed refrigerant. Tables 9.4-23 of the Final Safety Analysis Report (FSAR) and the Updated Safety Analysis Report (USAR) are both entitled "Control Complex Chilled Water System Component" and both list the maximum temperature of chilled water leaving the chillers as 45 degrees F.

The Shift Supervisor on duty at the time told the inspector that the air temperature in the Control Room (CR) was increasing but that it was still below the 90 degree F limit of Technical Specification 3.7.2 and, therefore, he and other operations personnel felt that the chiller was still operable. However, no analysis was made to determine that the chiller would still allow the Control Room Ventilation (M25/26) System to perform its design function and maintain proper conditions in the CR in the Emergency Recirculation Mode (as specified in FSAR Table 3.11-5) for seven days following a design basis accident.

The aforementioned tables of the USAR and FSAR and the large difference (27 degrees F) between observed and specified maximum chiller outlet temperature indicated that the "A" chiller was inoperable and, therefore, the plant was placed in the LCO associated with Technical Specification 3.6.2 for the M25/26 system.

As a result of the failure to recognize that the "A" train of the M25/26 system was inoperable, operations personnel did not implement the licensee's LCO tracking system. Furthermore, the Unit Supervisor did not indicate on the Work Request that the problem resulted in entry into an LCO, as required by Step 11 of Attachment 1 to PAP-902, Revision 3, as amended by Temporary Change Notice (TCN)-3, which was in effect at the time. This failure to follow procedures is a violation (440/88009-01(DRP)).

7. Monthly Surveillance Observation (61726)

On April 23 and May 28, 1988, respectively the inspector observed Technical Specifications required Surveillance Instructions (SVI): SVI-D17-T0065-C, Revision 3, "Containment/Drywell Radiation Monitor Functional for 1D17-K609C" and SVI-T23-T1016, Revision 1, "Containment Airlock Interlock Verification." For the above mentioned surveillances the inspectors verified that testing was performed in accordance with procedures, that test instrumentation was calibrated, that limiting conditions for operation were met, that removal and restoration of the affected components were accomplished, that test results conformed with Technical Specifications and procedure requirements and were reviewed by personnel other than the individual directing the test, and that any deficiencies identified during the testing were properly reviewed and resolved by appropriate management personnel.

On June 11, 1988, at approximately 2:30 p.m, the inspectors observed Technical Specifications required testing performed in accordance with Surveillance Instruction (SVI)-C51-T0024, Revision 2, "Average Power Range Monitor (APRM) Gain and Channel Calibration" and verified that limiting conditions for operation were met, that removal and restoration of the affected components were accomplished, and that test results conformed with Technical Specifications and procedure requirements.

All testing was performed in accordance with procedures except that the Instrument and Calibration (I&C) Technician failed to record on Attachment 1 of the SVI the As-Found APRM meter reading with his initials as required by Step 5.1.2.1.b of the SVI. The box for the APRM meter reading was left blank and not noted in the comments section which is contrary to Step 6.4.1.2 of Perry Administrative Procedure (PAP)-1105, Revision 4, "Surveillance Test Control," which requires that "ALL blanks on surveillance tests and other supporting documentation shall be initialed, noted in the comments or marked "N/A" (Not Applicable)."

The note immediately preceding Step 6.4.2.5.c.8 of PAP-1105 states that the lead test performer's signature only signifies that he has reviewed the test results. Step 8 also states that the Unit Supervisor's review is only a review of test results. However, from verbal interviews with operations personnel as a matter of practice these reviews also seek to verify test completeness. The lead test performer and the Unit Supervisor also failed to discover the technician's omission which indicates a lack of attention to detail.

The inspectors observed a Supervising Operator (SO) (who was not the lead test performer but had only verified a calculation in the SVI) paging through the SVI, apparently reviewing the SVI for completeness (even though not specifically required to do so by PAP-1105). The inspectors observed that as the SO was about to glance at the page with the obvious omission (as he had glanced at the other pages of the SVI) he looked up at a Unit Supervisor in the front of the horseshoe who was loudly explaining a non-work related activity. When the SO looked back at the SVI he had already turned the page with the omission. This instance of a lack of professional control room decorum prevented detection of an omission, specifically illustrating the need for control room decorum.

The inspectors discussed the SVI with the Responsible Engineer (RE) who was required by Step 6.4.2.5.c.16. of PAP-1105 to review final packages of SVI-C51-T0024 for test completeness. The RE volunteered that he had not noticed the omission but asserted that the recording of As-Found Data for this SVI was meaningless because APRMs were too erratic for the trending of As-Found readings to be useful. The inspectors agreed that in the instant case the omission had no safety significance whatsoever and that the blank space may well have been marked "NA", since this SVI was done during power ascension and no meaningful trending of As-Found Data is possible. The inspectors feel, however, that meaningful trending of As-Found APRM readings is possible and recommended when the reactor is at steady state equilibrium conditions. The inspectors have no further concerns with this particular omission but will continue to monitor the attention to detail of personnel as well as to licensee's efforts to enhance professional decorum in the control room.

No violations or deviations were identified.

8. Engineered Safety Feature (ESF) Walkdown (71710)

During this inspection period, the inspectors performed a detailed walkdown of the accessible portions of Train "A" of the Residual Heat Removal (RHR) System. The system walkdown was conducted using Valve Lineup Instruction (VLI)-E12, Revision 3 and the controlled Piping and Instrumentation Diagrams (P&IDs) for the RHR System.

During the walkdown, the licensee identified the "A" train as operable. The inspectors took into account that during the walkdown the "A" train was in various modes of operation and therefore in various valve lineups.

During the system walkdown, the inspectors directly observed equipment conditions to verify that hangers and supports were made up properly; appropriate levels of cleanliness were being maintained; piping insulation, heaters, and air circulation systems were installed and operational; valves in the system were installed in accordance with applicable P&IDs and did not exhibit gross packing leakage, bent stems, missing handwheels, or improper labeling; and, that major system components were properly labeled and exhibited no leakage. The inspectors verified that instrumentation associated with the system was properly installed, functioning, and that significant process parameter values were consistent with normal expected values. By direct visual observation or observation of remote position indication, the inspectors verified that valves in the system flow path were in the correct positions as required by the various modes of operation that were required; power was available to the valves; valves required to be locked in position were locked; and, that pipe caps and blank flanges were installed as required.

No violations or deviations were identified.

9. Containment Closeout (61715)

The inspectors verified on May 27, 1988 through local visual observation the proper positioning of the following isolation valves associated with the following penetrations:

Associated Isolation Valve	Test Connection (if any)	Penetration
E21-F006	F519*, F250**	P112
E22-F005	F519*, F520**	P410
P43-F055	F698**	P310
P43-F215	F701**	P311
E12-F037A	F572*, F577**	P113

E51-F013	F066(check valve)	P123
E12-F105		P401

*Locked Closed
**Locked Closed and Pipe Capped

Blank Flange or Cap Installed	Associated Containment Penetration	Associated Dry Well Penetration
(on Integrated Leak Rate Test (ILRT) Penetration)		
1E61-Blank Flange	P120	PRB 2034
1E61-Cap	P119	PRB 2033
1E61-Blank Flange	P109	PRB 2032

No violations or deviations were identified.

10. Licensee Event Reports Followup (92700)

Through direct observations, discussions with licensee personnel, and review of records, the following event reports were reviewed to determine that reportability requirements were fulfilled, immediate corrective action was accomplished, and corrective action to prevent recurrence had been accomplished in accordance with Technical Specifications.

LER 87066-LL	Inadequate Surveillance Instructions Result in Technical Specification Violation for Diesel Generator Operability
LER 87067-LL	Improper Maintenance Results in the B Main Steam Line Penetration Exceeding Technical Specification Leakage Limit
LER 87068-LL	Residual Heat Removal Shutdown Cooling Isolation During Surveillance Testing Due to a Drifting Pressure Regulator
LER 87069-LL	Unexpected Annulus Exhaust Gas Treatment System Train B Auto Start Following Transfer of Operating Trains, Due to Indeterminate Cause
LER 87075-LL	Personnel Error Results in a Violation of Technical Specification Due to the Plant Exceeding 150 psig with the Reactor Core Isolation Cooling System Inoperable
LER 87076-LL	Deficient Surveillance Instructions Result in Main Steam Drain Line Isolations

LER 87076-1L Deficient Surveillance Instructions Result in Main Steam Drain Line Isolations

LER 87077-LL Personnel Error Results in Technical Specification Violation Due to Late Completion of a Technical Specification Action Requirement

LER 88007-LL Reactor Protection System Actuation Results From Inadvertent Isolation of Instrument Air to Containment During System Restoration

LER 87066-LL reported the failure of SVI-R43-T1327 and T1328 to independently test for each Division that the barring device being engaged would prevent a start of the respective Diesel Generator (DG) as required by Technical Specification 4.8.1.1.2.e.14.a. The SVIs attempted to start the DGs with the barring device engaged but also with the switch in the position, which by itself will prevent a start.

Upon discovery, the SVIs were revised (as verified by the inspectors) and the lockout features for the Division 1 and 2 DGs were successfully tested on September 28 and 30, 1987, respectively. In addition, the licensee completed on January 22, 1988, a review of SVI's to ensure incorporation of all Technical Specification surveillance requirements. Failure to adequately test all DG lockout features as required by Technical Specification 4.8.1.1.2.e.14.a. is a violation (440/88009-02(DRP)). This violation meets the tests of 10 CFR 2, Appendix C, Section V.a; consequently, no notice of violation will be issued and this matter is considered closed.

LER 87067-LL reported the excessive seat leakage of an MSIV due to improper maintenance which was the subject of violation (440/87016-02(DRP)). The corrective actions were evaluated as adequate in the closeout of this violation which is documented in Paragraph 2 of this report.

LER 87075-LL documented that a shift supervisor made the Reactor Core Isolation Cooling (RCIC) system operable 45 minutes late due to an incorrect recall of Technical Specification requirements. This was the subject of violation (440/87023-02(DRP)). The corrective actions were evaluated as adequate in paragraph 6 of that report and, as stated in the cover letter of that report, no written licensee response was required.

11. Onsite Review Committee (40700)

The inspectors reviewed the minutes of the Plant Operations Review Committee (PORC) Meetings No. 88-46 through 88-61, conducted prior to and during the inspection period, to verify conformance with PNPP procedures and regulatory requirements. These observations and examinations included PORC membership, quorum at PORC meetings, and PORC activities.

No violations or deviations were identified.

12. Physical Security Procedures for the Resident Inspector (71881)

During this inspection period, the inspectors observed/reviewed selected licensee activities for conformance with the approved physical security plan. The inspectors reviewed security personnel staffing levels and verified that individuals authorized by the physical security plan to direct security activities were provided for each shift. The inspectors observed that access control measures, including search equipment, protected area and vital area barriers, and security door locking devices were operational and in use. The inspectors observed that personnel and packages entering the protected area were properly searched in accordance with licensee procedures. The inspectors observed that persons granted access to the site were badged to indicate whether or not they had unescorted or escorted access authorization. Finally, by direct observation the inspectors determined that the effectiveness of detection assessment aids was maintained by the absence of obstructions in the isolation zone, adequate illumination of the protected area and protected area barrier, and operable video surveillance equipment.

No violations or deviations were identified.

13. Radiological Protection Procedures For The Resident Inspector (71709)

Through discussions with licensee management, supervisory, and health physics personnel, and observation of licensee work planning activities, the inspectors determined that licensee personnel were aware of the ALARA program and that ALARA considerations were routinely considered in the planning of activities which involved occupational radiation exposure. The inspectors also determined through monthly Plant Status Meetings, such as the one described in Paragraph 14. of this report, and through review of the licensee's internally generated Monthly Performance Reports, that the status of meeting ALARA goals and objectives is periodically assessed and disseminated to affected plant personnel.

No violations or deviations were identified.

14. Plant Status Meetings (30702)

On June 29, 1988, at the Perry site, NRC management met with CEI management to discuss the current status of the plant, recent events, and licensee initiatives to improve the quality of plant operating and maintenance activities. These meetings are being held on a periodic (initially monthly) basis.

15. Exit Interviews (30703)

The inspectors met with the licensee representatives denoted in Paragraph 1 throughout the inspection period and on June 30, 1988. The inspectors summarized the scope and results of the inspection and discussed the likely content of the inspection report. The licensee did not indicate that any of the information disclosed during the inspection could be considered proprietary in nature.