U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-440/90014(DRP)

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

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

Facility Name: Perry Nuclear Power Plant

Inspection At: Perry Site, Perry, Ohio

Inspection Conducted: June 11 through July 30, 1990

Inspectors: P. L. Hiland

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Reactor Projects Section 3B

8/16/90
Date

Inspection Summary

Inspection on June 11 through July 30, 1990 (Report No. 50-440/90014(DRP))
Areas Inspected: Routine, unannounced safety inspection by resident inspectors of licensee action on previous inspection items; licensee event report followup; monthly surveillance observation; monthly maintenance observations; operational safety verification; onsite followup of events; evaluation of self-assessment capability; and plant status meeting.
Results: Of the eight areas inspected, two violations were identified and five non-cited violations were identified. The first violation was identified in the area of followup to previously identified items (Paragraph 2.a.). That violation concerned an extended maintenance outage on a safety related support component. The second violation was identified in the area of onsite followup of events (Paragraph 8.b.(2)). That violation concerned the failure to initiate adequate corrective action after a surveillance test failure. Both of these violations were receiving appropriate licensee attention at the close of the report period.

The five non-cited violations were identified in the area of licensee event report followup (Paragraph 3). All five of those violations met the test of 10 CFR 2, Appendix C, Section V.A; therefore, a Notice of Violation was not issued.

For this report period, the area of plant operations was considered adequate based on the inspectors observations of plant evolutions and response to events. The area of maintenance and surveillance was considered a weakness due to initial delay in documenting and initiating corrective actions for a surveillance test failure. The inspectors considered the licensee's actions, once the test failure was properly documented and reported, to be appropriate. The area of safety assessment/quality verification was considered a strength based on the inspectors review of activities performed by the onsite review committee. Of particular note was the detailed investigation into a rod scram time failure that occurred late in the report period.

In general, the inspectors found the areas of security and emergency preparedness to be a strength based on routine observations. The area of radiological controls was considered adequate; however, continued licensee management attention appears warranted to improve housekeeping in general and improve radiological practices at entries to contaminated areas. The inspectors noted that senior licensee management personnel were addressing the concerns in this area.

DETAILS

1. Persons Contacted

   a. Cleveland Electric Illuminating Company (CEI)

      \# M. Lyster, Vice President, Nuclear-Perry
      #*R. Stratman, General Manager, Perry Nuclear Power Plant (PNPP)
      M. Gmyreck, Operations Manager (PNPP)
      #*M. Cohen, Manager Maintenance Department (PNPP)
      \# V. Higaki, Manager, Outage Planning Section (PNPP)
      \# D. Cobb, Operations Superintendent (PNPP)
      #*S. Kensicki, Director, Perry Nuclear Engineering Department (PNED)
      #*V. Concel, Manager, Technical Section, (PNED)
      \# F. Stead, Director, Perry Nuclear Support Department (PNSD)
      #*H. Hegrat, Compliance Engineer (PNSD)
      #*R. Newkirk, Manager, Licensing and Compliance Section (PNSD)
      #*E. Riley, Director, Perry Nuclear Assurance Department (PNAD)
      *W. Coleman, Manager, Perry Nuclear Assurance Department (PNAD)

   b. U. S. Nuclear Regulatory Commission

      \# A. Davis, Regional Administrator, RIII
      \# J. Zwolinski, NRR, Assistant Director for RIII Reactors, NRR
      #*P. Hiland, Senior Resident Inspector, RIII
      *G. O'Dwyer, Resident Inspector, RIII
      \# D. Calhoun, Reactor Engineer, RIII

   \# Denotes those attending the management meeting on July 17, 1990.

   * Denotes those attending the exit meeting held on August 1, 1990.

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

   a. (Closed) Unresolved Item (440/90005-02(DRP)): Safety-related screenwash pump extended out-of-service time.

      As documented in Inspection Report 50-440/90005, Paragraph 9, dated May 4, 1990, the inspectors noted an extended maintenance outage for screen wash pump OP49C002B. At the conclusion of that inspection effort, this item remained unresolved pending the inspectors review of the reasons for the extended maintenance work.

      During this report period, the inspectors reviewed licensee Condition Report (CR) 90-071, dated July 24, 1990. The investigation detailed in that report noted the following sequence concerning maintenance activities on the subject safety-related pump:

      - November 16, 1989: Screenwash pump OP49C002B was removed from service to replace O-rings and repair oil leak. Work Priority 4B (Routine-Preventive Maintenance).

- November 24, 1989: After disassembly and inspection, it was identified that major pump components (sleeve, shaft, and packing ring), were degraded and required replacement.

- March 2, 1990: Screenwash pump 0P49C002B work order priority upgraded to 2C (Urgent-May deteriorate into Technical Specification). This upgrade occurred after the inspectors questioned the continued component outage.

- March 8, 1990: Sixteen additional parts (pump components, cap screws, bolts, etc.) were identified as needed to complete repair. These additional components were required since the original parts had been lost due to improper storage.

- May 4, 1990: Screenwash pump 0P49C002B was returned to service and the associated work order closed.

The unavailability of screenwash pump 0P49C002B on April 3, 1990, resulted in the licensee necessarily declaring both Division 1 and 2 emergency service water systems inoperable and making an ALERT declaration when the 100 percent redundant screenwash pump 0P49C002A was lost due to a mechanical failure (ref.: IR50-440/90005, Paragraph 11.b.(2)).

The cause for the extended maintenance outage was evaluated by the licensee and the following root causes were detailed in CR 90-071 as follows:

(1) Maintenance Program Deficiencies

As noted in the above timeline, following the discovery of needed additional components on November 24, 1989, the work order priority remained at a "Routine" level until continued inoperability was questioned by the inspectors. In addition, parts control was not adequate when additional components were found missing on March 8, 1990.

(2) Inadequate Management Response to Degraded Plant Equipment

Although the subject screenwash pump was identified on the licensee's monthly equipment out-of-service report for January, February, and March 1990, plant management did not elevate the work order priority on their own initiative. In addition, management attention was not adequate after the prolonged outage was questioned by the inspectors to expedite repairs or complete detailed support system impact prior to the loss of the redundant pump on April 3, 1990.

For the above identified programmatic problems, the licensee initiated the following corrective actions to prevent recurrence:

(1) All system engineers were required to review CR 90-071 with regard to the effects of subsystem component operability. The area to be stressed was the effect of prolonged inoperability without increased awareness.

(2) All operations personnel were required to review CR 90-071 with regard to the effects of subsystem component operability. The area to be stressed was the effects of prolonged inoperability of components without increasing awareness.

(3) All section managers were distributed copies of CR 90-071 for review of lessons learned.

10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that measures be established to assure that defective material and equipment are promptly identified and corrected. Between November 24, 1989, and March 2, 1990, the measures established to identify and correct out-of-service screenwash pump OP49C002B were not properly implemented and is a Violation (440/90014-01(DRP)).

As discussed above, the licensee's evaluation of the cause and corrective actions taken, as documented in Condition Report 90-071, appeared to have been accurate and reasonable. Therefore, the inspectors have no further concerns and the subject Unresolved Item and the noted Violation are considered closed.

b. Maximum allowed out-of-service time for one of two fully redundant traveling screens

As previously documented in Inspection Report 50-440/90005, Paragraph 9.b, dated May 4, 1990, the inspectors requested the NRR staff to evaluate the acceptability of plant operation with one of the two redundant Emergency Service Water traveling screens having degraded support equipment.

NRR memorandum J. Zwolinski to E. Greenman (RIII), dated June 5, 1990, responded to the inspector's request as follows:

Inspector's Question

With traveling screen-B out of service, a single failure in Division 1 components (e.g., traveling screen-A) would render the traveling screen support function inoperable. What is the allowed out-of-service time for one of the two 100 percent redundant traveling screens?

NRR Answer

The licensee made a determination that system operability was not affected by having one of two redundant 100-percent capacity traveling screens out of service so long as the other screen remained operable. The staff concurs with this determination. The requirement under GDC 44 related to system operability as contained in the Technical Specifications is to provide redundancy in loops for maintaining system flowpaths. Loss of one traveling screen for an indefinite period of time does not impact the ability of the plant to maintain redundant flowpath loops. Thus, no time limit is strictly required for loss of one traveling screen.

Based on the above NRR staff evaluation, the inspectors concluded that the licensee had complied with the associated Technical Specification "Operability" requirements for systems supported by the Emergency Service Water (ESW) traveling screens between November 1989 and May 1990 during which time ESW traveling screen B had degraded support equipment. However, the inspectors noted that in addition to meeting system Operability requirements imposed by 10 CFR 50.36 (Technical Specification), the quality assurance criteria contained in 10 CFR 50, Appendix B, must be considered. Therefore, as noted above in paragraph a., the extended maintenance outage of the ESW traveling screen support equipment was a violation of 10 CFR 50, Appendix B.

One violation was identified.

3. Licensee Event Report Followup (92700)

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

a. (Closed) LER 88001-00/88001-01: Reactor scram from intermediate range neutron monitors upscale trip due to excessive feedwater flow with manual control of a turbine feedwater pump.

On January 3, 1988, while in Operational Condition 2 (Startup), during the conduct of a reactor shutdown for a planned maintenance outage, plant operators attempted to control feedwater flow to the reactor using a turbine driven feedwater pump. As reactor pressure decreased below the minimum setpoint of the startup level controller, about 865 psig, the operating turbine driven feedwater pump continued to discharge at a rate causing reactor water level to increase. In response to the system transient, plant operators placed the turbine driven feedwater pump in manual and reduced the turbine speed which resulted in an expected reduction in feedwater flow and a decrease in reactor water level. Subsequent manual operation of the turbine driven feedwater pump to increase feedwater flow caused an injection of cold water and a power increase as indicated by the intermediate range monitors. The operator "at-the-controls" was unable to range-up the intermediate range monitors rapidly enough to prevent an automatic reactor scram when upscale trips were received on two intermediate range monitors.

Licensee's Evaluation of Caus^ and Corrective Action

Root Cause

(1) The root cause was identified to have been an inadequate instruction; whereas, the integrated operating instructions utilized for performance of a plant shutdown did not provide guidance to utilize the available motor driven feedwater pump for the plant shutdown evolution.

(2) The root cause was also determined to be a design deficiency since the startup level controller (initially controlling turbine feedwater pump speed) and the low flow controller did not provide an adequate overlap when changing from one mode of feedwater control to the other.

Corrective Actions

- Integrated Operating Instruction (IOI)-3, "Power Changes," was revised to include a note stating that the motor driven feedwater pump was preferred in order to minimize power transients.

- Integrated Operating Instruction (IOI)-4, "Shutdown," was revised to include a statement that the motor driven feedwater pump was preferred during the shutdown evolu' ,n.

- The low speed stop for the turbine driven feedwater pump, when on the startup level controller, was lowered from 3500 rpm to 3300 rpm.

- The turbine driven feedwater pump manual speed controllers were changed to reduce sensitivity by installing a potentiometer with a greater number of turns over the control band.

- The low flow controller was recalibrated to increase the span allowing 2,000 gpm flow through the low flow control valve.

Inspectors' Review

The inspectors concluded that the event received prompt management attention. The licensee's root cause determination and corrective actions to prevent recurrence of this event appeared to be accurate and reasonable. The inspectors review of documentation found that corrective actions as stated by the licensee had been completed with the exception of "fine-tuning" the low flow controller which was still considered an open work order pending system availability. Based on the completion of corrective actions as stated in the event report, this item is closed.

b. (Closed) LER 88018: Failure to recalibrate level instruments following design change results in Technical Specification Violation.

On May 5, 1988, the licensee identified that a design modification which changed instrument condensing chamber elevations in 1987 did not provide instructions for revising instrument calibration requirements. After installation of nozzle inserts in reactor vessel level instrument Channel D reference legs, the associated condensing chamber was reinstalled at an elevation of +7/16 inch over the original value. Although the increased elevation had been identified at the time of initial work completion, a use-as-is disposition was provided based on the elevation effect being within the allowable drift for the <u>wide</u> range level instruments. However, that use-as-is disposition <u>did</u> not consider the effect on <u>narrow</u> range instruments using the same reference leg. As stated <u>in the</u> event report, the licensee identified Channel D reactor high level scram setpoint to be above its allowable value by +0.06 inch.

In addition, the licensee identified that Channel B of the scram discharge volume level instrument provided a rod block at a setpoint outside the allowable value. Investigation identified that in 1986 a one-inch error was identified in the original head correction factor. The affected surveillance instruction was revised; however, the instrument was not recalibrated using the correct values.

<u>Licensee's Evaluation of Cause and Corrective Actions</u>

<u>Root Cause</u>

The root cause for both instruments being found outside their allowable value was attributed to personnel error. First, engineering personnel had overlooked the effect condensing chamber elevation changes had on <u>narrow</u> range reactor level instruments which resulted in Channel <u>D</u> operating outside allowable values. Second, the system engineer who discovered the one-inch head correction error in 1986 for the scram discharge volume level instruments failed to recognize the need to reperform the initial instrument calibration after revising the associated surveillance instruction.

<u>Corrective Actions</u>

- Reviewed as-left values for other instruments affected by the above changes. No additional problems were identified.

- Other instruments utilizing condensing chambers were reviewed to identify similar changes. None were identified.

- Design drawing as-build elevations were incorporated.

- Surveillance instructions for all reactor level instruments were revised to account for changes in condensing chamber elevations.

- All pre-1987 surveillance instructions were reviewed to assure revisions to those instructions did not affect plant instrumentation.

- Program change was incorporated to assure future changes in physical configuration of instruments to be evaluated for impact on instrument calibrations.

- Engineering staff was trained on the need to identify changes affecting instrument elevation.

Inspectors' Review

The inspectors noted that the corrective actions stated in the subject event report had been completed. The inspectors concluded that the licensee's root cause determination and corrective actions were accurate and reasonable.

Failure of the licensee to properly calibrate the Channel D narrow range reactor water level instrument and the scram discharge volume rod block instrument resulted in plant operations in violation of Technical Specifications 3.3.1 and 3.3.6, respectively (440/90014-02(DRP)). This violation was a "licensee identified item" which meets the tests of 10 CFR 2, Appendix C, Section V.G; therefore, a Notice of Violation will not be issued. Based on the inspectors review of completed corrective actions, this item is closed.

c. (Closed) LER 88025-00/88025-01: Overtravel of reactor protection system (RPS) power transfer switch resulted in loss of power to both RPS busses and a full RPS actuation.

On June 18, 1988, while in Operational Condition 4 (Cold Shutdown), plant operators attempted to transfer the RPS bus A power supply from its alternate (transformer) power supply to its normal (motor-generator) power supply. During the transfer evolution, the transfer switch was rotated beyond the required position resulting in the loss of power. The break-before-make transfer switch was expected to deenergize the RPS bus A; however, the same switch also transfers the RPS B power supply. With the anticipated loss of RPS bus A and the loss of RPS bus B due to the overtravel during switch rotation, a complete loss of power to both RPS busses occurred.

Licensee's Evaluation of Cause and Corrective Action

Root Cause

The licensee attributed the cause for this event to be personnel error on the part of the plant operator manipulating the RPS bus transfer switch. A contributing factor was poor human factors design of the transfer switch.

Corrective Action

- A permanent label was installed above and below the RPS bus transfer switch cautioning operators on the potential for overtravel during switch operation.

- The system operating instruction was revised to add the same caution on potential switch overtravel during switch operation.

Inspectors' Review

The inspectors noted that a thorough and prompt evaluation of the cause for this event was performed by the licensee. The inspectors verified by direct field observation that the permanent labels were installed at the RPS bus transfer switch located in the main control room. The inspectors concluded that the licensee's root cause determination was accurate and the corrective actions taken appeared reasonable. This item is closed.

d. (Closed) LER 88034: Start of Division 1 diesel generator building ventilation system due to failure of power converter.

On September 3, 1988, an unexpected start of the Division 1 diesel generator building ventilation system occurred. At the time of event occurrence, the plant was operating at 100 percent power. The licensee identified the start signal for the system ventilation fans was generated from a failed 125/24 VDC converter in the diesel generator control circuitry.

Licensee's Evaluation of Cause and Corrective Action

Root Cause

The installed failed converter was discarded prior to a detailed failure analysis. However, the first replacement converter was discovered defective prior to its installation. The licensee requested the supplier to perform a failure analysis on the first replacement converter and evaluate reportability under 10 CFR 21.

The supplier of the failed "spare" converter responded to the licensee's request for a failure analysis stating that the most likely failure mechanism of the converter was an individual component (diode) failure due to age or random failure. The supplier stated that a review for reportability in accordance with 10 CFR 21 had been performed and it was determined not to be a reportable basic component failure.

Corrective Action

- The failed converter was replaced with a second spare.

- The first spare converter was returned to the supplier for failure analysis as discussed above.

Inspectors' Review

The inspectors concluded that the licensee's evaluation of root cause and corrective actions taken were adequate. Although the original failed component was lost, the licensee pursued a root cause failure analysis from a generic viewpoint once the first spare converter was found defective. The inspectors noted that improvements were made in the governing administrative procedures to reduce the likelihood of discarding failed components without considering the need for failure analysis. Based on the corrective actions taken by the licensee as stated above, this item is closed.

e. (Closed) LER 88040: Electrical fault in control power circuit resulted in loss of control room emergency recirculation mode of the heating, ventilation and air conditioning (HVAC) system and entry into Technical Specification 3.0.3.

On October 7, 1988, while in Operational Condition 1 (Power Operation) at 100 percent power, an electrical fault occurred in the control circuit for Train B chiller unit of the control room HVAC. At the time of discovery, the redundant Train A chiller was removed from service for a planned maintenance activity. With both trains of control complex chillers inoperable, the licensee declared both trains of the supported system, control room emergency recirculation, inoperable and entered Technical Specification 3.0.3. About one hour after initial entry into Technical Specification 3.0.3, temporary repairs were made to Train B chiller and Technical Specification 3.0.3 was exited.

Licensee Evaluation of Cause and Corrective Actions

Root Cause

The cause of this event was equipment failure. Degradation of control circuit wire insulation resulted in grounding the power supply. Wearing of the insulation was due to the wire contacting its vibrating metal enclosure.

Corrective Action

- The failed control wiring and its associated valve actuator were replaced.

- To prevent recurrence, the individual control wires at the actuator were wrapped with Varglass sleeving and secured with Raychem heat shrink material.

- In addition to the repairs made to the failed Train B chiller, similar repair work was completed on Train A chiller.

Inspectors' Review

The inspectors concluded that the licensee's root cause determination was accurate and the corrective actions appeared reasonable. Of note was the proactive approach to perform preventive repairs on the Train A chiller. Based on the inspectors review of completed corrective actions as stated above, this item is closed.

f. (Closed) LER 88041: Reactor core isolation cooling (RCIC) system containment isolation caused by failed leak detection transmitter.

On October 13, 1988, the RCIC system was automatically isolated when an associated leak detection system high steam flow transmitter failed low. At the time of event occurrence, the plant was in Operational Condition 1 (Power Operation) at 100 percent power. Initial response to this unexpected system isolation was to place RCIC in a secured state and investigate the cause for the automatic isolation. Following repair, the RCIC system was restored to service on October 14.

Licensee's Evaluation of Cause and Corrective Actions

Root Cause

The root cause for this event was attributed to "temporary offscale failure syndrome" exhibited by the Rosemount differential pressure transmitter used in the high steam flow leak detection circuit. As discussed in the event report, the "temporary offscale failure syndrome" was caused by microscopic conductive particles in the sensor module fill fluid shorting the transmitter.

Corrective Actions

As a result of a previous event, an engineering evaluation had been in progress at the time of this event which concluded that the RCIC isolation signal was not required for safe operation of the plant. The low pressure signal was removed from the RCIC trip circuitry and modified to provide a control room alarm function only.

Inspectors' Review

The inspectors concluded that the licensee provided a prompt determination of the apparent cause and corrective action for this event. The plant modification to remove the trip logic was initially processed as a "lifted lead and jumper." The inspectors review of the associated change approval documentation noted that the original design organization (General Electric) had concurred in the elimination of the low steam flow trip function. Based on the corrective actions taken as stated above, this item is closed.

g. (Closed) LER 88042: Local power range monitor (LPRM) failure results in reactor protection system (RPS) actuation.

On October 22, 1988, while in Operational Condition 4 (Cold Shutdown), LPRM 16-17D failed upscale. The average power range monitor (APRM) Channel A increased to about 17 percent, exceeding the Neutron Flux-High Setdown setpoint of 15 percent. The Channel A APRM trip resulted in a half-scram signal on RPS Channel A/C. Since RPS Channel B/D already had a half scram signal due to ongoing maintenance, a full RPS actuation was generated. However, no control rod movement occurred since all rods were fully inserted at the time of event occurrence.

Licensee Evaluation of Cause and Corrective Action

Root Cause

The cause for this event was component failure. Troubleshooting identified the detector had shorted. At the time of this event the licensee had experienced 3 similar failures out of 164 installed detectors.

Corrective Actions

- Immediate actions accomplished was to bypass the failed LPRM detector and reset the RPS Channel A/C.

- During the licensee's Spring 1989 refueling outage, the failed detector was replaced.

Inspectors' Review

The inspectors concluded that the licensee had performed a prompt evaluation of the cause for this event and that appropriate short term and long term corrective action had been taken. This item is closed.

h. (Closed) LER 88043: High pressure core spray pump room cooler inadvertent start.

On October 30, 1988, while in Operational Condition 4 (Cold Shutdown), an unexpected start of the high pressure core spray (HPCS) room cooler occurred following maintenance testing. With the HPCS pump breaker "racked out" to its test position, maintenance personnel cycled the breaker to verify satisfactory completion of repairs. As designed, when the breaker was closed, the HPCS pump room cooler received a start signal.

Licensee's Evaluation of Cause and Corrective Action

Root Cause

The root cause for the unexpected start of the HPCS room cooler was attributed to a procedural deficiency. The work planner failed to verify the functions of the HPCS pump breaker auxiliary switches prior to release of the work package.

Corrective Action

- The work planner involved with this event was counseled to ensure interlock functions are identified and controlled during maintenance.

- Guidelines to verify interlock functions were added to generic electrical instructions for breaker maintenance.

Inspectors' Review

The inspectors concluded that the licensee had identified the cause for this event and that corrective actions were appropriate. This item is closed.

i. (Closed) LER 88045: Sodium Pentaborate concentration in the standby liquid control (SLC) system exceeds Technical Specification limits.

On November 21, 1988, while in Operational Condition 1 (Power Operation) at 100 percent power, both trains of the standby liquid control (SLC) system became inoperable when the concentration of Sodium Pentaborate was found to be above Technical Specification limits. The as-found concentration was 14.0 percent, by weight, and the Technical Specification limit was 13.8 percent, by weight. The concentration was restored to within Technical Specification limits by adding 125 gallons of water to the SLC storage tank.

Licensee's Evaluation of Cause and Corrective Action

Root Cause

The cause for this event was a program deficiency that allowed the concentration of Sodium Pentaborate to exceed Technical Specification limits before corrective actions were taken. Prior samples of the Sodium Pentaborate concentration had shown an increasing trend; however, action was not required to be initiated since those previous concentrations were below Technical Specification limits.

Corrective Action

The sampling procedure (SVI-C41-T1026) was revised to require Sodium Pentaborate concentrations to be maintained within administrative limits that were more restrictive than the Technical Specification limits.

Inspectors' Review

The inspectors concluded that the licensee's evaluation of the cause for this event was accurate and appropriate corrective actions had been taken. This item is closed.

j. (Closed) LER 88046: Inadequate design of containment pressure instrument resulted in a condition outside the updated safety analysis report.

On December 1, 1988, following a review of NRC Information Notice (IN) 88-76, the licensee identified that pressure instruments did not account for temperature-induced gradients inside or outside the secondary containment. As a result, the Perry Secondary Containment Annulus may not have been maintained at a uniform negative pressure with respect to the outside atmosphere. At the time of initial receipt of IN 88-76, the licensee increased the setpoint for the Secondary Containment differential pressure to greater than 0.75 inch vacuum water gage (WG). An engineering evaluation was performed that concluded a value of 0.66 inch vacuum WG was required to maintain negative pressure within Secondary Containment with low outside temperatures. The licensee concluded that at some time prior to identifying the subject instrument inaccuracy, the plant could have operated with Secondary Containment pressure exceeding design limits.

Licensee's Evaluation of Cause and Corrective Actions

Root Cause

The cause for this event was an inadequate instrument setpoint design calculation. The original design calculation did not consider the effect of temperature on pressure gradients inside and outside Secondary Containment.

Corrective Action

- The Secondary Containment pressure setpoint was recalculated and associated design documents revised.
- The system operating instruction was revised to maintain the Secondary Containment pressure within the new calculated values.

Inspectors' Review

The inspectors noted that the licensee had taken prompt action upon initial receipt of NRC Information Notice 88-76. Increasing the normal operating pressure range while performing the necessary engineering review was a conservative measure. The inspectors concluded that the licensee's stated cause for the event was accurate and the corrective actions taken were appropriate. This item is closed.

k. (Closed) LER 88047: Degraded flow switch resulted in loss of control room heating, ventilation, and air condition (HVAC) systems and entry into Technical Specification 3.0.3.

On December 8, 1988, while in Operation Condition 1 (Power Operation) at 100 percent power, Train A of the control room HVAC became inoperable when a cooling water flow switch failed to operate properly. At the time of that failure, the redundant Train B of control room HVAC had been removed from service for planned maintenance activities. With both trains of control room HVAC inoperable for their emergency recirculation modes Technical Specification 3.0.3 was entered. About 30 minutes after initial entry, Technical Specification 3.0.3 was exited upon restoring control room Train A HVAC to an operable status.

Licensee Evaluation of Cause and Corrective Actions

Root Cause

The cause of this event was equipment failure. While attempting to start the Train A control complex chiller, the cooling water flow switch failed to reset the low cooling water flow interlock. A contributing problem was a malfunctioning valve actuator which had limited normal cooling flow.

Corrective Actions

- The flow switch was verified to be capable of functioning properly with increased flow provided by repairing the normal cooling water flow valve actuator.

- The cooling water flow operating range was increased to assure chiller restart in the event of interruption in cooling water flow.

- Methods for improving system performance were investigated by the independent safety engineering group (ISEG).

Inspectors' Evaluation

The inspectors concluded that the licensee's evaluation of cause for this event was adequate. The corrective actions taken appeared to have been effective based on subsequent system (i.e., chiller start) performance. The inspectors noted that the study on improving system reliability was an open ISEG item (No. 89-007) awaiting a third party review. Based on the completed corrective actions as discussed above, this item is closed.

1. (Closed) LER 89002: Radiation monitor declared operable prior to completion of maintenance resulted in not taking required samples.

On January 13, 1989, the licensee identified that the liquid radwaste discharge radiation monitor had been declared operable prior to completion of post-maintenance testing following a design change. With that radiation monitor inoperable Technical Specification 3.3.7.9-1, Action 110, required that two independent samples be analyzed prior to a liquid release and that two individuals verify release rate calculations and valve lineup.

On January 10 and 12 liquid radwaste discharges were made without obtaining the backup analysis or performance of valve lineup verification prior to discharge.

Licensee's Evaluation of Cause and Corrective Actions

Root Cause

The cause for this event was inadequate procedural guidance. In addition, personnel failing to follow existing procedures was a contributing factor.

Corrective Actions

- Plant Administrative Procedure (PAP)-0905, "Work Order Process," was revised to improve the control of work completion including required post-maintenance test activities.

- All work planners were trained on the required information for each block in a work order.

- All work supervisors were trained on the need for verbatim compliance with procedures.

- The event was reviewed by plant operators during requalification training emphasizing review of work closure documents.

Inspectors' Review

The inspectors concluded that this event had received prompt evaluation by licensee management personnel. The root cause evaluation appeared thorough and corrective actions were appropriate. Based on the successful post-maintenance test conducted January 16 and the fact that a sample had been drawn and analyzed for the liquid discharges performed on January 10 and 12, the inspectors noted that the event was not safety significant. However, positive control of plant systems during liquid discharges is necessary and the failure to maintain that control was significant.

The failure of the licensee to comply with Action Statement No. 110 of Technical Specification 3.3.7.9-1 during the conduct of liquid discharges on January 10 and 12, 1989, is a Violation (440/90014-03(DRP)). This violation was a "licensee identified item" which meets the test of 10 CFR 2, Appendix C, Section V.G; therefore, a Notice of Violation will not be issued. This item is closed.

m. (Closed) LER 89004-00/89004-01: Loss of auxiliary building ventilation resulted in reactor water cleanup (RWCU) system isolation.

On February 3, 1989, while in Operational Condition 1 (Power Operation) at 100 percent power, the RWCU system automatically isolated on high room differential temperature. Plant operators verified proper system isolation and discovered inadequate auxiliary building ventilation flow as the reason for increased differential RWCU pump room temperatures. The RWCU system was returned to service on February 4.

Licensee Evaluation of Cause and Corrective Actions

Root Cause

The cause for this event was a failed air control valve on the auxiliary building ventilation air discharge damper. The component failure caused inadequate air flow in the auxiliary building with the subsequent increase in RWCU pump room temperatures.

Corrective Action

- The failed solenoid valve was replaced.

- Since the same valve had experienced a similar failure earlier (ref.: LER 88010), additional corrective actions included verifying system application for valve design, check of voltage supplied at the solenoid, inspection of similar valves used in the same system, and the failed valve was reinspected six months after the event. No anomalies were identified.

Inspectors' Review

The inspectors concluded that the licensee had performed a prompt evaluation of the root cause for this event. The corrective actions performed, when considering a known previous failure, appeared reasonable. Based on completion of corrective actions as stated above, this item is closed.

n. (Closed) LER 89005-00/89005-01: Drywell head holddown bolts found detensioned.

On February 24, 1989 during the removal of Drywell head holddown bolts, the licensee identified that the bolts were loose. The expected bolt tightness was the as-installed condition prior to plant operation of about 500 foot-pounds torque.

Licensee Evaluation of Root Cause and Corrective Actions

Root Cause

The cause for this event was too small an initial preload in the bolts. The original design did not establish bolt installation torques to envelope added bolt tension forces resulting from all external loading conditions.

Corrective Actions

The bolt installation torque was recalculated based on a new value of initial bolt prestress equating to about 38,000 pounds preload per bolt.

Inspectors' Review

As documented in Inspection Report 50-440/89014(DRS), dated July 6, 1989, a special safety inspection was performed by a NRC Region III specialist inspector as a followup to the licensee's investigation. Based on the NRC staffs' review of the investigation of root cause and corrective action, as documented in Inspection Report 50-440/89014(DRS), this item is closed.

o. (Closed) LER 89007: Mode change made without completing required surveillance test.

On February 27, 1989, during the conduct of changing from Operational Condition 4 (Cold Shutdown) to Operational Condition 5 (Refueling), source range monitor signal to noise ratios were not obtained. Technical Specification 4.0.4 prohibits entry into an Operational Condition unless the surveillance requirements have been performed.

Root Cause

The cause for this event was personnel error on the part of plant operators. The operators erroneously assumed the required surveillance was to be performed one shift prior to control rod withdrawal or core alterations.

Corrective Actions

- Personnel involved in the event were counseled on the need for attention to detail.
- Plant logs were revised to eliminate confusion.
- The event was reviewed by licensed operators during requalification training.

Inspectors' Evaluation

The inspectors concluded that the licensee had performed a prompt evaluation of the cause for this event with appropriate management attention. The corrective actions taken appeared reasonable to prevent recurrence.

Failure of the licensee to perform required surveillance testing on source range monitors prior to changing from Operational Condition 4 to Operational Condition 5 on February 7, 1989 is a Violation (50-440/90014-04(DRP)). This violation was a "licensee identified item" which meets the test of 10 CFR 2, Appendix C, Section V.G; therefore, a Notice of Violation will not be issued. This item is closed.

p. (Closed) LER 89008: Automatic start of the control room ventilation system in its emergency recirculation mode during surveillance testing.

On March 1, 1989, while in Operational Condition 5 (Refueling), Train A of the control room ventilation system unexpectedly started in its emergency recirculation mode. After verification that an actual initiation signal was not present, control room operators restored Train A of the control room ventilation system to a standby readiness condition.

Licensee's Evaluation of Cause and Corrective Actions

Root Cause

The cause for this event was a procedural deficiency. A surveillance instruction being performed at the time of event occurrence, did not provide a caution that resetting isolation signals would automatically start the ventilation system.

Corrective Actions

- The appropriate surveillance and operating instructions were revised to include the caution on system start when initiation signals, inserted for test purposes, are reset.

- This event was reviewed by licensed operators during requalification training.

Inspectors' Review

The inspectors concluded that the licensee had performed a prompt evaluation of the cause of this event and appropriate corrective actions had been implemented. Based on completion of corrective actions as stated above, this item is closed.

q. (Closed) LER 89009: Disabling control room alarm function for effluent radiation monitor resulted in failure to perform Technical Specification Action statement.

On March 21, 1989, while in Operational Condition 5 (Refueling), the licensee identified that the control room alarm function for the Unit 2 Vent Sampler Flow Rate Monitor had been made inoperable on March 11, 1989. Technical Specification 3.3.7.10, Action 123 required flow rate estimates to be made every four hours when the flow rate monitor was not operable.

Licensee Evaluation of Cause and Corrective Actions

Root Cause

The cause for this event was personnel error on the part of plant personnel assigned to remove Unit 2 (not operational) nuisance alarms that were a distraction to control room operators in Unit 1. When disabling Unit 2 nuisance alarms, the involved personnel overlooked the Unit 2 Vent Sampler Flow Rate Monitor which was a required instrument for Unit 1 operation.

Corrective Actions

- An independent review of Unit 2 alarms needed for Unit 1 control room staff was performed. No additional anomalies were identified.

- Personnel involved were made aware of the error and the prudent use of an independent reviewer.

Inspectors' Review

The inspectors concluded that the licensee had performed a prompt evaluation of the cause for this event and the corrective actions taken were appropriate.

Failure of the licensee to comply with Technical Specification 3.3.7.10, Action 123 between March 11 and March 21, 1989, is a violation (440/90014-05(DRP)). This violation was a "licensee identified item" which meets the test of 10 CFR 2, Appendix C, Section V.G; therefore, a Notice of Violation will not be issued. This item is closed.

r. (Closed) LER 89012: Containment isolation valve unexpected closures during maintenance activities.

This item was previously reviewed by the inspectors as documented in Inspection Report 50-440/90008, Paragraph 2.j. At the conclusion of that inspection, Unresolved Item 440/89007-04 was closed based on the inspectors review and conclusion that the licensee had identified the root cause for this event and completed appropriate corrective actions. Based on the inspectors previous review documented in Inspection Report 440/90008, this item is closed.

s. (Closed) LER 89013: Automatic start of emergency service water pump during post-maintenance testing.

On April 15, 1989, while in Operational Condition "At All Times," (fuel removed from reactor), an unexpected start of emergency service water (ESW) Pump A occurred during post-maintenance testing of emergency closed cooling (ECC) Pump A.

Licensee's Evaluation of Cause and Corrective Actions

Root Cause

The cause of this event was personnel error on the part of a plant technician. While attempting to install a test jumper, the technician inadvertently touched a conductor in the start circuit for the reactor core isolation cooling (RCIC) system. As designed, ESW-A automatically started in response to the RCIC start signal.

Corrective Action

The involved technician was counseled on the need for caution when working on or around energized equipment. In addition, all technicians were directed to inform their supervision if significant risk was involved during jumper installation.

Inspectors' Review

The inspectors concluded that the licensee's cause evaluation was prompt and accurate. The stated corrective actions were appropriate. This item is closed.

t. (Closed) LER 89014: Automatic start of Division 3 diesel generator.

On April 25, 1989, while in Operational Condition "At All Times," an automatic high pressure core spray (HPCS) initiation signal was inadvertently initiated during troubleshooting activities. Since the HPCS pump was removed from service it remained idle. However, the associated Division 3 emergency diesel responded to the signal and automatically started.

Licensee's Evaluation of Cause and Corrective Actions

Root Cause

The cause for this event was personnel error due to miscommunications between personnel performing troubleshooting activity and control room personnel. Between the time that approval to commence the troubleshooting work effort and its actual performance, plant conditions were changed in preparation for a maintenance test of the Division 3 emergency diesel. The change in plant conditions (i.e., diesel control switch was taken out of the "pull to lock" position) allowed the troubleshooting to start the emergency diesel.

Corrective Actions

The shift crew was counseled on the need for knowing plant status and the details of planned work during a refueling outage. In addition, this event was reviewed by all operators during requalification training.

Inspectors' Review

The inspectors concluded that the licensee had conducted a prompt and accurate evaluation of the cause for this event. The corrective actions taken as stated above were appropriate. This item is closed.

u. (Closed) LER 89015: Error in head correction for narrow range reactor water level instruments resulted in trip setpoints outside the Technical Specification allowable values.

On May 12, 1989, while in Operational Condition "At All Times," the licensee identified an error made in the head correction factor for three reactor water level transmitters. The error resulted in a difference of 2 inches between the actual and indicated reactor water level. One instrument, 1B21-N080B, was one of four inputs in the reactor protection system logic that generate a reactor trip signal at level 3 (low water level). Another instrument, 1B21-N095B, provided a confirmatory level 3 (low water level) signal to the Automatic Depressurization System (ADS). The third instrument, 1C34-N004B, provided a level 8 (high water level) trip signal to the feedwater pumps and main turbine.

The 2 inch error resulted in a conservative trip value for the level 8 high water level trip signals and a non-conservative trip signal for the level 3 trip signals.

Licensee's Evaluation of Cause and Corrective Actions

Root Cause

The cause for this event was a design deficiency. During preparation of design input values, the wrong drywell penetration elevation was used to determine density correction factors used in instrument calibrations. A value of 633.09 feet was initially used instead of the correct value of 647.00 feet.

Corrective Actions

- The affected instruments were recalibrated with the correct head adjustment factor.
- All other reactor water level instrument calculations were reviewed to assure proper elevations had been used. No additional errors were identified.

Inspectors' Evaluation

The inspectors concluded that the licensee had performed a prompt and accurate evaluation of the cause for this event. The non-conservative errors were in trip logic systems that had redundant instrumentation properly calibrated; therefore, the inspectors noted that the safety significance of this event was low.

In addition, the licensee performed an evaluation assuming the low water level trips occurred at the non-conservative water level and verified no significant change in peak clad temperature. The inspectors also concluded that the corrective actions taken, as stated above, were appropriate.

Failure of the licensee to maintain the subject reactor water level instruments within the Technical Specification allowable ranges during the first operating cycle is a Violation (440/90014-06(DRP)). This violation was a "licensee identified item" which meets the test of 10 CFR 2, Appendix C, Section V.G; therefore, a Notice of Violation will not be issued. This item is closed.

Five non-cited violations (NCV) were identified.

4. Plant Zebra Mussel Program (92701)

At the request of Region III management, the inspectors reviewed the licensee's planned activities to control zebra mussels (Dreissena polymorpha).

As previously documented in Inspection Reports 50-440/89026, Paragraph 6.b.(6), dated December 15, 1990; 50-440/89028, Paragraph 9, dated January 12, 1990; and 50-440/90002, Paragraph 10, dated March 20, 1990, the licensee had actively investigated and evaluated the potential impact the infestation of zebra mussels could have on the Perry plant.

Licensee research and monitoring

Two ongoing studies were in progress. First, the licensee was continuing to evaluate the effectiveness of the installed plant chlorination system to control veliger (larval mollusk in the stage when it has developed the velom). The normal chlorination system injects sodium hypochlorite into the plant's service water systems every twelve hours for thirty minutes. Settlement monitors were located at the service water pumphouse, circulating water pumphouse, radwaste heat exchanger, and in the turbine building closed cooling system. Effective treatment was to be evaluated based on settlement at the monitor locations.

Second, the licensee was testing the ability to control adult mussels in pressurized water systems. Specifically, the fire protection system which was maintained at 130-140 psi. Following test results on the adult zebra mussels, the licensee was extending the test to evaluate control of veligers. The test vessel was installed at the discharge of the fire protection system jockey pump.

Electric Power Research Institute (EPRI) Research

The licensee contracted with EPRI to conduct a joint research project in conjunction with Davis-Besse and the Bay Shore (non-nuclear) plant. At Perry the research that was to be performed included an evaluation of current chemical technologies and development of treatment strategies. Studies were to be conducted in the sidestream equipment located in the service water pumphouse.

Perry 1990 Control Program

Following approval from the Ohio Environmental Protection Agency on June 15, 1990, the licensee initiated a chemical treatment program utilizing "Betz Clam-Trol CT-1." The treatment program started in July with the implementation of Temporary Instruction (TXI)-0102, "Zebra Mussel Treatment." Additional treatments were scheduled to be conducted in late August and late October. Each treatment was to inject the Betz Clam-Trol at the plant's raw water intake for a period of si: to twelve hours. Prior to discharge back to Lake Erie, the effluent stream was to be detoxified by the addition of bentonite clay.

No violations or deviations were identified.

5. Monthly Surveillance Observation (61726)

For the below listed surveillance activities the inspectors verified one or more of the following: testing was performed in accordance with procedures; test instrumentation was calibrated; limiting conditions for operation were met; removal and restoration of the affected components were properly accomplished; 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 managemert personnel.

Surveillance Test No.	Activity
SVI-C61-T1104, Revision 5	"Accıdent Monitoring and Remote Shutdown Channel Checks"
SVI-D17-T0065C, Revision 3	"Containment/DW Plenum Radiation Monitor Functional for 1D17-K609C"
SVI-B21-T0077D, Revision 3	"Main Steam Line Low Condenser Vacuum Channel D Calibration for 1B21-N075D"

No Violations or Deviations were identified.

6. Monthly Maintenance Observation (62703)

Station raintenance activities of safety-related systems and components listed 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 this review: 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; radiological controls were implemented; and, fire prevention controls were implemented.

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

The following specific maintenance activity was observed:

W. O.	Subject
89-7823	Control Rod Hydraulic Control Units 54-15 and 06-15

No Violations or Deviations were identified.

7. Operational Safety Verification (71707)

General

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, reviewed tagout records 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.

No violations or deviations were identified.

8. Onsite Followup of Events at Operating Power Reactors (93702)

a. General

The inspectors performed onsite followup activities for events which occurred during the inspection period. Followup inspection included one or more of the following: reviews of operating logs, procedures, condition reports; direct observation of licensee actions; and interviews of licensee personnel. For each event, the inspectors reviewed one or more of the following: the sequence of actions; the functioning of safety systems required by plant conditions; licensee actions to verify consistency with plant procedures and license conditions; and verification of the nature of the event. Additionally, in some cases, the inspectors verified that licensee investigation had identified root causes of equipment malfunctions and/or personnel errors and were taking or had taken appropriate corrective actions. Details of the events and licensee corrective actions noted during the inspectors' followup are provided in Paragraph b. below.

b. Details

(1) Unexpected start of control room ventilation in the emergency recirculation mode

On June 21, 1990, while in Operational Condition 1 (Power Operation) at 100 percent power, both trains of the control room ventilation system automatically started in their emergency recirculation mode. Plant operators verified proper system operation; and, after concluding that an actual automatic start signal was not present, the control room ventilation trains were restored to a standby status.

The cause for this event was identified to have been personnel error on the part of an Instrument and Control (I&C) technician. While performing maintenance on a ventilation damper relay, the I&C technician lifted a neutral wire lead connected to the relay terminals interrupting the power supply to two chlorine monitors. That loss of power resulted in a high toxic gas alarm and, by design, an automatic start of control room ventilation in the emergency recirculation mode. Following additional review of system impacts, the relay maintenance was successfully completed.

The licensee reported this event to the NRC Operations Center via the ENS about 10:15 p.m. on June 21, 1990. In addition, Licensee Event Report (LER) 440/90014 was issued by the licensee on July 20, 1990, detailing this event occurrence, root cause, and corrective actions taken to prevent recurrence. The inspectors will perform a followup review of that LER after completion of licensee corrective actions.

(2) Momentary power excursion above licensed core thermal power limit

On June 24, 1990, while in Operational Condition 1 (Power Operation) at 100 percent power, the licensee momentarily exceeded the Perry licensed 100 percent thermal power limit. At the time of event occurrence, plant personnel were performing a routine monthly surveillance test (SVI-C71-T0039) of reactor protection system (RPS) trip signals generated on a partial closure of the main steam isolation valves (MSIVs). The test performance required the temporary installation of two volt-ohm meters on the affected logic circuit in order for test personnel to observe the anticipated trip signal as each MSIV was partially slow-closed.

During the test performance for the first MSIV, the temporary test meters failed to indicate the receipt of the expected RPS trip signal. As a result, the licensed operator actually performing the MSIV test closure continued to hold the closure test button until receipt of all average power range monitors (APRMs) upscale alarms (108% neutron flux). Upon receipt of the APRM alarms, the licensed operator released the MSIV test button allowing the MSIV to return to its full open position.

The investigation by personnel involved in the surveillance test activity identified the cause for failure to receive the expected RPS trip signal was defective test meters. One meter had a blown fuse and the second meter's test leads were not installed correctly. The problems with the test equipment were corrected and the surveillance test was successfully performed on all eight MSIVs.

Initially, personnel involved in the surveillance test did not consider the momentary power excursion significant enough to warrant the initiation of any corrective action document. The inspectors noted that no mention of the anomaly was made in the completed surveillance test report nor in any shift operating logs. On June 25, personnel involved in the test on June 24 reconsidered the need for corrective action and initiated licensee Condition Report (CR) 90-153. Upon review of that CR, the inspectors asked the licensee to evaluate the potential for exceeding licensed power limits during the momentary transient. On June 26, at about 10:45 a.m., the licensee informed the NRC Operations center via the ENS that their initial calculations indicated thermal power had increased to about 105% for about 2 seconds. Final calculations performed by the licensee and reported in LER 440/90015 stated that thermal power peaked at 103.8 percent, with power exceeding 102 percent for 5.7 seconds.

The maximum thermal power level limit authorized by the facility operating license (License No. NPF-58, Paragraph 2.C.(1)) is 3579 megawatts thermal (100% power). A uniform basis for enforcing maximum licensed power was promulgated in NRR memorandum E. Jordan to all Regions dated August 22, 1980. That staff position recognized brief excursions up to 2% above license limits provided the average power level over any eight hour shift is maintained no greater than the 100% limit. In addition, the analyzed transients detailed in Chapter 15 of the Perry Updated Safety Analysis Report envelope the momentary excursion experienced for the event.

The inspectors concluded that the event itself had <u>no</u> safety significance with regard to reactor thermal limits; however, the inspectors considered the response by personnel performing the subject surveillance test on June 24 to be inadequate for the following reasons:

- The surveillance test report contained no record of problems experienced during the initial test performance.

- Neither the plant logbook nor the Shift Supervisors logbook contained any entry of the initial test failure.

- Although "the event" was not considered to be safety significant by the inspectors, the operating license required in Paragraph 2.F that violations of the requirements contained in Section 2.C (Power Limit was contained in Section 2.C.(1)) be reported within 24 hours. That report was not made in a timely manner.

- The inspectors concluded that personnel performing the test were not aware of other plant indications that could have provided earlier indications of faulty test equipment.

- Previous escalated enforcement action had been taken, in part, for inadequate control of surveillance test activities (reference: Notice of Violation and Proposed Civil Penalty dated March 12, 1990).

The licensee's review of the cause for this event was completed during this report period. LER 440/90015, dated July 20, 1990, identified the following root causes:

"1. The technicians involved in this event failed to properly check the test equipment prior to performing this surveillance. Although not a proceduralized requirement, the need to verify proper equipment functioning is considered entry level knowledge for an I&C technician. Additionally, the technician did not obtain proper contact between test leads and relay terminals when he installed the second VOM.

2. The Supervising Operator manipulating the MSIV was not sure of the timing and setpoints at which valve indication and RPS initiation would be expected. Although approved licensed operator training programs do not require that an operator has this information committed to memory, it is expected that personnel familiarize themselves with such details prior to performing evolutions. This information is readily available in the control room. Additionally, the operator was observing only the valve position indicating lights and waiting for verbal notification from the technician that the VOM indicated relay actuation. Additional process instrumentation such as reactor power, pressure, and steam line flow was not adequately monitored during the test.

3. The I&C Technician heard relays inside the RPS cabinet change state at the expected time; however, because his VOM readings did not indicate actuation, he did not notify the operator who was positioning the valve.

4. SVI C71-T0039 includes a caution, warning test performers not to allow the MSIV to stroke more than 10 percent closed during the test. However, information necessary for the performers to properly evaluate valve performance, such as limit switch settings, sequence, and expected timing, is not provided in the instruction. Additionally, the critical implementing step in the procedure is written assuming proper operation of all equipment; the step directs the operator to close the valve until the VOM indicates actuation of the relay."

The following corrective actions to prevent recurrence were listed in LER 440/90015:

"1. Work planning activities are being initiated to install special test lugs on the affected RPS relays, improving the potential for adequate test lead contact.

2. Modifications to SVI C71-T0039 have been completed to provide expected indications, event sequencing and timing, and time limitations for valve stroking.

3. Plant Administrative Procedure (PAP-0201), "Conduct of Operations," will be revised to require operating personnel to monitor potentially affected significant process variables during the performance of operating evolutions.

4. Plant Administrative Procedure (PAP-1105), "Surveillance Test Control," will be modified to direct the Unit Supervisor to ensure that any anomalies experienced during the performance of a surveillance test are adequately documented in the completed surveillance package.

5. Direction has been issued to Operations personnel via the daily instructions regarding the use of ERIS data for the evaluation of operational events.

6. I&C supervision has issued a directive to all I&C personnel detailing various requirements to be completed prior to or during the performance of maintenance and surveillance activities, specifically including verification of test equipment operability.

7. The I&C technicians involved in this event have been counselled with specific emphasis on attention to detail, the need to verify operability of test equipment,

consequences of equipment failure, and the requirements for adequate documentation of surveillance activities. Additionally, one technician received disciplinary action.

8. All I&C technicians and supervisors will be trained to the details of this event during continuing training program activities.

9. The Operations personnel involved in this event have been counselled with respect to appropriate control of surveillance testing, the need to be adequately familiarized with the details of operational evolutions, the failure to recognize potential Operating License violations, and the failure to adequately document the circumstances of this event.

10. The details of this event will be discussed with all licensed operators as a part of the Licensed Operator Requalification Program. Specifically, this training will include discussion on the need to anticipate unexpected plant performance during surveillance testing, on MSIV limit switch development, the use of the computer setpoint list and details on recognition of power transient events."

10 CFR 50, Appendix B, Criterion XVI, requires in part, that conditions adverse to quality such as failures and malfunctions are promptly identified and corrected. Significant conditions adverse to quality require the cause and corrective action taken to be documented and reported to appropriate levels of management. Failure of personnel performing MSIV surveillance testing on June 24, 1990, to promptly document and report the initial surveillance test failure is a Violation (440/90014-07(DRP)).

Throughout the report period, the inspectors discussed with licensee management the cause for this event and the lack of prompt documented corrective action. The inspectors reviewed the licensee's root cause evaluation and corrective actions taken or planned to prevent recurrence. Both the cause evaluation and the corrective actions appeared to be appropriate; therefore, no further response to the noted violation is required. The inspectors will review completed corrective actions for this violation and LER 440/99015 in a subsequent inspection report.

(3) Automatic Shift of High Pressure Core Spray Suction

On July 10, 1990, while in Operational Condition 1 (Power Operation) at 100 percent power, the High Pressure Core Spray (HPCS) system automatically shifted from its "normal" suction lineup (Condensate Storage Tank (CST) to the "alternate" suction path from the suppression pool. Plant personnel verified system lineup to the suppression pool and identified the cause for the automatic suction shift to be a failed CST level transmitter. The licensee initiated Condition Report (CR) 90-168, dated July 10, 1990, to document the investigation of root cause for the CST level transmitter.

The licensee reported this event to the NRC Operations Center via the ENS at about 8:30 p.m. on July 10 in accordance with the requirements of 10 CFR 50.72.

One Violation was identified.

9. Evaluation of Licensee Self-Assessment Capability (40500)

The plant operations review committee (PORC) was the licensee's onsite review committee. The inspectors reviewed the below listed PORC meeting minutes published during the inspection period and determined that: the minutes were thoroughly documented; the minutes clearly denoted the topics discussed and the basis for any conclusions; the action items were clearly identified and followed up; and the committee reviewed safety-significant concerns that were not specifically required by technical specifications.

PORC Meeting No.	Dated
90-044	5/31/90
90-045	6/07/90
90-046	6/15/90
90-047	6/19/90
90-048	6/20/90
90-049	6/21/90
90-050	6/22/90
90-051	6/28/90

No violations or deviations were identified.

10. Plant Status Meeting (30702)

NRC Management met with CEI management on July 17, 1990, at the Perry plant, in order to discuss the current status of monthly performance indicators, planned activities to test chemical treatment of zebra mussels, status of planning for second refueling outage, status of recent technical specification change requests, and events of interest that had occurred since the last plant status meeting. Personnel in attendance at that meeting are designated by (#) in paragraph 1 of this report.

The licensee discussed planned treatment of plant service water systems to control infestation of zebra mussels (detailed status of licensee actions are presented above in paragraph 4). A discussion of events occurring since the last meeting was held with emphasis placed on the June 24 power spike (see paragraph 8.b.(2) above). The licensee provided a briefing on the status of planning for the Perry second refueling outage scheduled to start in September 1990. In addition, the licensee reviewed Technical Specification change requests that had been submitted for staff review and approval prior to the refueling outage scheduled to start in September.

NRC management acknowledged the licensee's plans and current plant status.

11. Violations For Which A "Notice of Violation" Will Not Be Issued

The NRC uses the Notice of Violation as a standard method for formalizing the existence of a violation of a legally binding requirement. However, because the NRC wants to encourage and support licensee's initiatives for self-identification and correction of problems, the NRC will not generally issue a Notice of Violation for a violation that meets the tests of 10 CFR 2, Appendix C, Section V.G. These tests are: 1) the violation was identified by the licensee; 2) the violation would be categorized as Severity Level IV or V; 3) the violation was reported to the NRC, if required; 4) the violation will be corrected, including measures to prevent recurrence, within a reasonable time period; and 5) it was not a violation that could reasonably be expected to have been prevented by the licensee's corrective action for a previous violation. Violations of regulatory requirements identified during the inspection period for which a Notice of Violation will not be issued were discussed in Paragraph 3.

12. Exit Interviews (30703)

The inspectors met with the licensee representatives denoted in Paragraph 1 throughout the inspection period and on August 1, 1990. The inspector 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.

During the report period, the inspectors attended the following exit interviews:

Inspector	Exit Date
G. Christoffer	6/15/90
M. Huber	7/13/90
M. Kunowski	7/13/90
A. Januska	7/13/90
J. House	7/13/90