U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-440/95008

FACILITY

Perry Nuclear Power Plant, Unit 1

License No. NPF-58

LICENSEE

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

DATES

September 2 through October 20, 1995

INSPECTORS

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APPROVED BY

R. D. Lanksbury, Chief 12/14/95
Reactor Projects Branch 2 Date

AREAS INSPECTED

A special announced inspection of operations, engineering, maintenance, and plant support was performed. Safety assessment and quality verification activities were routinely evaluated. Follow-up inspection was performed for non-routine events, for certain previously identified items, and Temporary Instruction 2515/128, Revision 1.

RESULTS

Assessment of Performance

OPERATIONS: Overall plant operations were conducted well. Command and control related to the scrams, and oral communications were excellent. Two concerns were identified pertaining to operator errors. One of the three recent scrams was caused by multiple operator errors and weaknesses in the implementation of self-checking practices were noted. This event was more than of minor safety significance and a violation was issued. (Section 1.3.2)

MAINTENANCE: Overall, work was performed in a thorough and professional manner. Management decisions were conservative and excellent briefings were provided to the craft and operators. Some weaknesses were identified with the Infrequently Performed Test or Evolution (IPTE) process. However, the inspectors concluded the process had improved over the last 2 years and is now generally good. One task was identified as a potential IPTE candidate by the inspectors, but was not worked under the IPTE process. (Section 2.1) Some weaknesses were also identified with planning and review. A violation was issued for inadequate verification of a component and falsification of records associated with that verification, that occurred during the 1994 refueling outage. (Section 2.4)

ENGINEERING: Support to other departments was excellent. Engineering response was thorough and prompt to the inspector identified RHR snubber that was attached to the piping at greater than the maximum specified angle. (Section 3.2) Systems Engineering responded promptly and thoroughly to a supplement to Information Notice 94-66 concerning Reactor Core Injection Cooling (RCIC) turbine governor valve stem corrosion which led to the identification of an inoperable RCIC turbine governor valve. (Section 3.4) However, a weak evaluation of reactor pressure vessel level changes (Section 3.3) and RHR piping snubber damage (Section 3.2) did not support the licensee's conclusions. This issue had been previously identified as an Unresolved Item.

PLANT SUPPORT: Overall, activities related to radiation protection, contamination control, and emergency preparedness were well conducted. However, some concerns were identified. The inspectors identified a minor computer error with electronic dosimeter (MG) set points and noted that a Potential Issues Form (PIF) was not generated until prompted by the inspector. (Section 4.1.1) The inspectors observed fluid leaking from a Reactor Feed Booster Pump at two locations and reported this to the Radiation Protection Technician at the control point. However, nothing had been done to contain the previously reported leaks until it was identified to the plant manager 2 days later. (Section 4.1.2) An example of ineffective corrective actions was identified when the warehouse office building phone lines used for emergency plan radios were inadvertently cut. Although this had strong management support resulting in prompt documentation and development of corrective actions, 3 weeks later the radio cable was cut again. (Section 4.2)

SAFETY ASSESSMENT/QUALITY VERIFICATION: Overall performance was well conducted. Action on previous inspection findings and activities related to the reactor vessel water level reference leg backfill modification were good. (Section 7.0) The self-assessment of engineering and technical support was conducted by an independent and objective team, and the scope and depth of the audit was good. (Section 3.1) Some weaknesses were noted in the identification and correction of plant problems. Several adverse conditions were not documented with a PIF until questioned by an inspector and leaks were not properly documented and controlled. The inspectors remained concerned that over the past year some individuals appeared to be narrowly focused on their assigned tasks and not observant of adverse conditions around them. (Section 5.1) A Quality Assurance audit indicated that progress had been made on reducing the large backlog of corrective actions, however some problems remained with timely and effective corrective actions. This was highlighted by the slow and inadequate evaluation of an unexplained reactor pressure vessel water level change and repetitive problems with the tone alert system. (Section 5.3)

<u>Summary of Open Items</u>
<u>Violations</u>: Identified Sections 1.3.2 and 2.4 in this report
<u>Unresolved Items</u>: Not identified in this report
<u>Inspector Follow-up Items</u>: Not identified in this report
<u>Non-cited Violation</u>: Identified in Section 1.3.1 of this report

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INSPECTION DETAILS

1.0 OPERATIONS

NRC Inspection Procedures 71707, 71500, and 92901 were used to perform an inspection of plant operations activities. One cited violation was identified for failure to follow a procedure. No deviations were identified.

1.1 Operations Summary

The plant was in hot shutdown in a forced outage at the beginning of the inspection period. On September 2, 1995, during power ascension following plant startup, an automatic scram was caused by personnel errors. Startup from that scram was completed on September 4 and the plant operated at power until September 11 when another automatic scram was caused by an instrument bus inverter failure. Startup from the second scram was completed on September 14 and the plant operated at power levels up to 100 percent for the rest of the inspection period.

1.2 Operator Control of Routine Plant Operations Was Good

The inspectors observed routine plant operations and concluded that overall performance was good. Oral communications among operators continued to be excellent.

1.3 Reactor Scrams

Two automatic reactor scrams occurred during this inspection period and another scram had occurred at the end of the previous inspection period. The response of the operators was at times excellent, but was not consistent. One of the three scrams was caused by multiple operator errors. Four of the last five inspection reports had identified several operator errors with minimal safety significance. These errors demonstrated weaknesses in the implementation of self-checking practices.

1.3.1 Reactor Scrams Caused by Equipment Failure

On August 31, 1995, a relay in the Division 2 Instrument Power Supply failed. This caused two reactor water level transmitters to falsely indicate low. As designed, the reactor core isolation cooling (RCIC) system initiated, the main turbine and reactor feedwater pump turbines tripped, the Motor Feed Pump (MFP) auto started, and the reactor scrammed. Command and control of operator activities and oral control room communications were excellent. Performance of plant equipment after the scram was improved over previous scrams. All safety equipment called upon functioned as expected except for two items. An outboard containment isolation valve for a containment radiation monitor failed to close (caused by a stuck actuator or linkage) and the RCIC pump failed to trip when reactor vessel level reached the high level setpoint (caused by the failed power supply).

After the scram there were two Technical Specifications that required compliance with Action Statements within 1 and 2 hours (T.S. 3.8.1.1.b, Division 2 Diesel Generator inoperable, and T.S. 3.3.2 containment isolation valves inoperable respectively, both caused by the failed instrument power supply). The operators used the time that they identified the conditions as the starting times for compliance with the action statements. However, since the actual time of the failure of the instrument power supply was known and was earlier, that was the time that should have been used. Therefore, the operators did not complete the actions within the time constraints. The failures to meet the time constraints constituted a violation of minor safety significance and are being treated as examples of a non-cited violation, consistent with Section VII of the NRC Enforcement Policy.

Investigation of the Division 2 Instrument Power Supply failure revealed a failed relay resistor caused by excessive current and heating over time. The excessive current was caused because the relay was designed for use with AC circuits and not for use in DC circuits. This failure mechanism had been identified in 1984 and the failed power supply was thought to have been upgraded with a DC relay. One corrective action included in Licensee Event Report (LER) 95-005 was an investigation of why the AC relay was not replaced with a DC relay when other power supplies were upgraded. The inspectors will review the results of that investigation during final review of the LER.

On September 11, the power supply replaced after the August 31 scram failed when a capacitor failure degraded circuit performance. This caused a reactor scram similar to the one on August 31. The operators and the plant performed well with no significant problems or errors. The subsequent investigation was thorough and revealed that the capacitor failure was an unavoidable early life failure. The licensee's response to the same conditions that occurred following the scram on August 31 demonstrated that its corrective actions were appropriate for the minor violation identified in the preceding paragraph.

1.3.2 Reactor Scram Caused by Personnel Errors

Another automatic reactor scram occurred from 15 percent power on September 2, 1995, when reactor vessel water level reached the low level setpoint. Feedwater was being provided to the vessel by the MFP. Operators, as part of a normal startup, were attempting to place one of two Reactor Feed Pump Turbines (RFPT) in standby. System Operating Instruction SOI-N27, "Feedwater Control," Revision 10, Temporary Change 1, effective August 25, 1995, was being used for this activity. Multiple errors by the operators, as listed below, caused an unintended reduction in feedwater flow to the reactor vessel:

Operator #1 failed to follow the procedure when instructed to place the RFPT in manual control. He did not verify that the turbine control was in "manual."

Operator #2 failed to verify that the RFPT was in manual control prior to changing the system's status.

Operator #2 failed to adequately observe plant parameters after making changes in the plant configuration and left the control room.

Operators #3 and #4 failed to diagnose and take corrective action for the cause of decreasing reactor vessel water level during the approximately 60 seconds between the low level alarm and the reactor scram.

The failure to verify that the RFPT was in manual control prior to changing the system's status, as required by step 4.11 of SOI-N27, was a violation of Technical Specification 6.8.1 (50-440/95008-01(DRP)) which required that written procedures or instructions be implemented for applicable activities recommended in Appendix A of Regulatory Guide 1.33.

Because there were multiple operator errors associated with the scram, the inspectors reviewed operator working hours for the 6 weeks prior to the scram. The data from this review and a licensee Human Performance Enhancement System report indicated that fatigue may have contributed to the multiple personnel errors. Based on the 6 weeks of data reviewed, the inspectors concluded that most operators typically worked in excess of 40 hours a week during normal operations. Perry's Technical Specifications required that administrative procedures be developed and implemented to limit the working hours of unit staff who perform safety-related functions in accordance with the NRC Policy Statement on working hours (Generic Letter No. 82-12). The Generic Letter and Perry's administrative procedure state that enough operating personnel should be employed to maintain adequate shift coverage without routine heavy use of overtime. The inspectors concluded that the 6 weeks of data reviewed indicated that while extensive used overtime was not made routinely by the licensee, some operators on occasion, such as the operator involved in this scram, had worked an extensive amount of overtime. Discussions with plant management personnel revealed that the licensee considered itself to be in compliance with the Technical Specifications and the NRC Policy Statement and had no short term plans to change its shift staffing policies. Further review of this issue will be made by the inspector to determine if the heavy reliance on the use of overtime is more wide spread and to evaluate any corrective action the licensee plans to minimize the affect of fatigue on operator performance.

1.4 Main Turbine Trip Caused by Equipment Degradation

On September 12, 1995, about 2.5 hours after main generator synchronization, the main turbine tripped due to an indication of low bearing oil pressure. This condition occurred during normal testing of the lube oil system, which started and stopped various lube oil pumps. When the turning gear oil pump was stopped a pressure perturbation in the oil piping caused the main turbine to trip. Further testing using additional test equipment confirmed the cause of the trip. Although individual system components had operated within the vendor's specifications, the licensee's thorough evaluation indicated that a pressure perturbation experienced during the testing was sufficient to be detected as a low pressure condition. The operating configuration of the lube oil system was changed to prevent short term recurrence. The licensee plans to refurbish various lube oil system components during the next refueling outage to prevent long term recurrence.

2.0 MAINTENANCE AND SURVEILLANCE

NRC Inspection Procedures 62703, 61726, and 92902 were used to perform an inspection of maintenance and testing activities. One no-response violation was identified for failure to independently verify equipment status and falsification of the associated record during the 1994 refueling outage. No deficiencies were identified.

2.1 Infrequently Performed Tests or Evolutions (IPTE)

The purpose, as defined in Perry Administrative Procedure PAP-1121, of designating a test or evolution as an IPTE is to provide management oversight and control so that the plant's level of safety is maintained within acceptable limits. Overall use of the IPTE process has improved over the last 2 years and is now generally good.

The inspectors observed an IPTE which involved replacement of an instrument air valve gasket. Failure of this air supply would have caused a plant scram with many normally available systems out-of-service. Management decisions associated with this IPTE, such as bringing two additional air compressors on site, were conservative. During task preparations, several challenges to conservative operation were introduced. For example:

- Fire hoses were originally planned for use to connect the air compressors to the air system;

- Potential problems with the tie-down of temporary air hoses in the plant existed during the inspector's walkdown; and

- Use of existing air system piping, not normally used, did not originally include a blowdown to remove particulates.

All of these challenges were appropriately identified by the licensee and corrected. However, these challenges to management could have been avoided by more thorough early planning and review.

The inspectors also observed an excellent IPTE briefing prior to RCIC post maintenance testing. There was good participation in the briefing.

The inspectors reviewed work on the Inclined Fuel Transfer System (IFTS) in preparation for the upcoming refueling operation. The work was not designated as an IPTE, but qualified as an IPTE in six of eight criteria provided in PAP-1121. The work involved significant personnel safety hazards including radiation exposure and potential contamination due to flooding of the IFTS pool. The infrequently performed nature of the work and inadequate planning caused a siphon break to be misplaced such that water was being drawn into the IFTS pool from the storage pool over the drywell. A temporary pump was being used to maintain the IFTS pool in a drained condition until the source of this additional water was identified and stopped. Although the licensee had reviewed this work during the planning process, use of the IPTE process might have identified the problems prior to beginning the work.

2.2 Replacement of Division 2 Instrument Power Supply

On September 1, 1995, upon energizing the Division 2 Instrument Power Supply that replaced the power supply that caused the August 31, 1995, scram, Emergency Closed Cooling (ECC) Pump "A" started automatically. The licensee had considered the possibility that components might receive an inadvertent start signal when the power supply was energized. However, the review done to identify potential start signals was not thorough enough to identify the start signal to the ECC pump. This event had no safety significance. Licensee Event Report (LER) 95-006 discusses this event. The inspectors will evaluate the licensee's corrective actions during their review of the LER.

2.3 Verification of DC Relays in Installed Power Supplies

The licensee determined that the cause of the August 31, 1995, scram was a failed AC relay used in DC service in a Division 2 Instrument Power Supply inverter. Licensee equipment documentation indicated that the installed Division 1 and Division 2 Instrument Power Supplies had DC relays. However, the licensee chose to visually verify the presence of DC relays. Each power supply uses two inverters with one on top of the other. The DC relay in the upper inverter is easily verified. The DC relay in the lower inverter can only be verified with a boroscope. The licensee procured a boroscope from off site. The inspectors observed instrument and control technicians perform the difficult boroscope verification in a thorough and professional manner.

2.4 Improper Verification of Equipment Prior to Work

On April 9, 1994, with all fuel offloaded from the reactor, Perry Work Order 93-3056 was being used by contractors (NPS Energy Services) to control removal of a blank flange from Residual Heat Removal (RHR) "A" full flow test line, part of a safety-related system. RHR "A" and "B" were both out of service and drained at that time. Work order step 010, required, in part, that the work superintendent, responsible foreman and craftsman independently verify that the component to be worked was the item identified in the work order. Each was required to sign the work order after he had verified the identity of the component. The NPS Energy Services (NPS) superintendent mistakenly identified the RHR "B" full flow test line as the correct component and signed the work order. The responsible NPS maintenance foreman and the craftsmen did not independently verify the location of the blank flange (1E12-D003A) in the "A" RHR full flow test line prior to proceeding with the work. When the craftsmen removed what they expected to be a blank flange they observed that it was a flow orifice instead. The work was stopped, the control room was notified, and the licensee began an investigation. The licensee determined that the maintenance foreman falsified the work order by later signing that he had performed the independent verification when he had instead relied on the superintendent's verification. 10 CFR 50.9 required that information required by the Commission's regulations be complete and accurate in all material respects. Neither of the craftsmen had signed the work order verification step prior to starting work. 10 CFR Part 50, Appendix B, Criterion V, required that activities affecting quality be prescribed by documented instructions and procedures and be accomplished in accordance with those instructions and procedures. Maintaining inaccurate information on the work order and failure to work in accordance with the work order resulted in a violation (50-440/95008-02(DRP)) of 10 CFR 50.9 and 10 CFR Part 50, Appendix B, Criterion V. The inspectors verified that the licensee had trained NPS personnel on the triple verification requirement and that the individuals involved were aware of the requirement. The inspectors verified that the licensee took prompt and effective corrective actions. The individuals involved were promptly disciplined. Additional training was provided to appropriate individuals. Because the licensee's corrective actions were prompt and effective, no response to this violation is required.

3.0 ENGINEERING

NRC Inspection Procedures (IP) 37550, 37551, 40501, and 92903 were used to perform onsite inspection of engineering activities. No violations or deviations were identified.

3.1 Engineering and Technical Support Self-Assessment

As allowed by IP 40501, the licensee performed a self-assessment of engineering and technical support (E&TS) under their Quality Assurance Audit No. PA 95-25. The NRC's in-process inspection of this effort concluded that the audit team's independence and objectivity were appropriate, the audit was consistent with previously presented scope, and the depth of the self-assessment was good.

The schedule, scope, effort level, and team qualifications of the self-assessment were delineated in the licensee's July 28, 1995, letter and further discussed with the NRC on August 15. The scope of the audit was comprehensive including all inspection requirements in IP 37550, and at the request of the NRC, assessed the materiel condition of the plant.

This later aspect was performed using plant walkdowns in conjunction with reviews of program efficiencies and trends for the number of "operator work-arounds," work requests, and non-conforming conditions. The depth of the audit was good, based on the level of detail in the team's questions. Also, the objectivity and independence of the audit team was considered appropriate based on the team's observed interactions with engineering personnel.

The audit team did not identify any operability issues; however, all audit findings were entered into the licensee's normal corrective action program for tracking, disposition, and evaluation. Corrective actions will be determined within the normal program and will consider the findings from other recent engineering self-assessments. This process was considered suitable for the current situation.

At the exit meeting, the audit team concluded that Engineering adequately performed routine and reactive activities and adequately provided technical support to other site departments. They also noted that several improvement initiatives had clearly demonstrated near term results compared to assessments in the recent past, and that the potential for long term strengths could be realized if these initiatives were effectively implemented. The licensee's management supported the findings and conclusions of the audit team and acknowledged the need for continued improvement.

The NRC considered this self-assessment effort to be good and will perform a final technical inspection after the licensee issues its final report.

3.2 Residual Heat Removal Snubber

During a partial walkdown of RHR systems prior to an HPCS outage, the inspectors identified a snubber that was attached to an RHR C pipe clamp at an angle greater than 5 degrees, the maximum specified angle. The licensee verified that the angle was about 6 degrees, initiated a PIF, and declared the snubber inoperable. Engineering promptly evaluated the snubber and determined that the allowable side load on the snubber had not been exceeded because the design snubber load was considerably lower than the maximum allowed. Since several previous licensee inspections had not identified the excessive angle, the licensee was concerned that the forces that caused the snubber to be at an angle may have damaged the snubber. Therefore, the snubber was replaced and the original snubber was tested. Testing of the snubber indicated it was not damaged.

Engineering and operations initial response to this issue was thorough and prompt. However, after repositioning of the pipe clamp the licensee identified a weakness in control of piping insulation which had no immediate safety significance.

3.3 Engineering Response to an Unexpected Reactor Water Level Change

The inspectors reviewed the licensee's final evaluation of issues that had previously been identified as an Unresolved Item (50-440/94011-02(DRP)). This item was generated after Reactor Pressure Vessel (RPV) level decreased 10 inches when, during the last refueling outage, operators opened a Residual Heat Removal (RHR) valve that connected the RHR system to the RPV. Previous problems with the RHR fill and vent procedure, failed RHR snubbers, and the operators' apparent acceptance of unexplained RPV level changes had been discussed as issues related to the resolution of the item. The inspectors had expected a comprehensive review of all issues. However, the licensee's evaluation of the issues was not adequate because the conclusions were not supported by the evaluation. The inspectors discussed the adequacy of the response with System Engineering management. The inspectors will review this URI when the licensee completes its evaluations.

3.4 Engineering Response to RCIC Turbine Governor Valve Sticking (IN 94-66)

Systems Engineering responded to a supplement to Information Notice 94-66 concerning RCIC turbine governor valve stem corrosion by developing a repetitive task to verify smooth movement of the RCIC turbine governor valve. This task supplemented surveillance testing which was conducted quarterly. This task was initially scheduled to be performed monthly and was first performed on September 25, 1995. Since there was some roughness in valve movement, the Responsible System Engineer (RSE) evaluated the condition of the valve and increased the task frequency to weekly. The valve did not move through its full range and the RCIC turbine was declared inoperable at 5 a.m. after the Unit Supervisor discussed the condition with the RSE. A test start of the RCIC turbine was then attempted and it tripped on overspeed, confirming that the governor valve was stuck. Replacement of the valve stem began a short time later. Engineering support of operations was excellent in this case.

3.5 Engineering Response to a Missing Bolt

The inspectors observed that a bolt was missing from a cooling water outlet flange for the turbocharger on the Division 1 Emergency Diesel Generator (EDG). The inspectors did not observe any leakage from the flange. The licensee promptly performed an engineering evaluation of the missing bolt and determined the EDG had been operable without the bolt. The bolt was promptly replaced.

4.0 PLANT SUPPORT

NRC Inspection Procedures 71750, 81700, 84750, and 92904 were used to perform an inspection of Plant Support Activities. No violations or deviations were identified.

4.1 Radiation Protection Performance

4.1.1 Incorrect Electronic Dosimeter Settings

On October 5, 1995, an inspector was briefed on Radiation Work Permit (RWP) #95-0120, in preparation for inspecting RCIC post maintenance testing. When the inspector logged his electronic dosimeter (MG) on the computer he observed that the dose alarm and dose rate alarm setpoints were reversed compared to the RWP briefing. Radiation protection personnel promptly changed the computer setpoints and notified personnel already at the job site that they had to return to the control point to reset their MGs. The inspectors were concerned about this situation for two reasons: there had been many earlier opportunities for licensee personnel to identify this error and having to recall personnel from the job site had the potential to delay restoration of safety equipment operability. The inspectors did not see a Potential Issue Form (PIF) on this situation the next day and on October 11 discussed the situation with the PIF coordinator and the QA manager. They both stated that a PIF should have been written and on October 12, PIF #95-2037 was written. The inspectors brought this to the attention of the Region III Radiation Protection Inspector for his consideration during a future inspection. This issue is further discussed in Section 5.1.

4.1.2 Licensee Failure to Control Leaks of Radioactive Fluid

On October 17, 1995, the inspectors observed fluid leaking from Reactor Feed Booster Pump B at two locations and reported this to the Radiation Protection Technician at the control point. On October 19, during a tour with the Plant Manager, the inspectors noted that nothing had been done to contain the previously reported leaks. The inspectors noted that the Plant Manager again reported the leaks to personnel at the control point. During a later inspection the inspectors observed that the leaks had been contained. This issue is further discussed in Section 5.1.

4.2 Emergency Preparedness

On September 27, 1995, during renovation of the warehouse office building, phone lines used for emergency plan radios were inadvertently cut. This resulted in a potential reduction in emergency response capability. The licensee promptly documented this and developed corrective actions with strong management support. However, on October 18 an emergency plan tone alert radio test was conducted and failed. A prompt licensee investigation revealed that radio cable had again been cut. This was an example of ineffective corrective action.

4.3 Housekeeping (Bulletin 95-02)

On October 17, 1995, the NRC issued Bulletin 95-02: "Unexpected Clogging of a Residual Heat Removal (RHR) Pump Strainer While Operating in Suppression Pool Cooling Mode." The bulletin had been transmitted electronically to the inspectors and they immediately provided the licensee with a copy. On October 18, 1995, the inspectors accompanied licensee personnel on a routine weekly containment inspection. Although the inspectors and licensee personnel identified many minor items, housekeeping was excellent and the suppression pool was very clean. Near the end of the inspection an RHR pump that had been running was stopped and the inspectors observed a system engineer conduct the normal post-operation inspection of the strainer. The inspection was thorough and the strainer was clean. Inspection of the strainer was facilitated by the cleanliness of the pool water.

5.0 SAFETY ASSESSMENT AND QUALITY VERIFICATION (SAQV)

NRC Inspection Procedures 40500, 92720, 92901, 91902, 91903, and 91904 were used to perform an inspection of Safety Assessment and Quality Verification activities. No violations or deviations were identified. However, deficiencies were again identified in the reporting of adverse conditions and in response to identified leaks.

5.1 Identifying and Responding to Anomalies in the Plant

In general, there continued to be aggressive identification of problems by a wide variety of individuals and organizations. However, over the past year the inspectors had identified failures on the part of the Perry Organization to identify and respond to equipment and work performance problems. In most cases, several individuals from different departments (including management, operations, maintenance, health physics, and engineering) had opportunities to identify and respond to the problems. The inspectors remained concerned that the failure to promptly identify and respond to these problems indicated that some individuals appeared to be narrowly focused on their assigned tasks and not observant of adverse conditions around them. The following are examples:

- In October 1994, Inspection Report No. 440/94013 described five small radioactive water and steam leaks found by the inspectors where, in all but one of the cases, workers in the area had either not noticed the leaks or had not reacted properly to them.

- In May 1995, Inspection Report No. 440/95004 described five negative examples where identification and communication of anomalies in the plant, either with equipment or work performance, were not promptly resolved.

- In July 1995, Inspection Report No. 440/95006 described poor storage of scaffold parts, poor conditions in the hot shop, and improper storage of transient combustibles.

- In July 1995, Inspection Report No. 440/95006 described leaks in the reactor feedwater booster pump (RFBP) area that were not posted or contained until after the second time the inspectors brought them to the attention of the licensee.

During this inspection period, the inspectors observed several adverse conditions that should have been documented with a PIF that were not initially documented (Sections 3.2, 3.5, and 4.1.1). The inspectors also observed that another RFBP (isolated for maintenance) had two uncontained leaks (Section 4.1.2). Neither leak had been identified and the inspectors reported them to health physics (HP). The inspectors later observed that the leaks had not been contained and they were again reported.

The individual significance of the above examples was minimal. However, Perry management had indicated that it was their expectation that each member of the Perry Organization had a duty to promptly bring such problems to the attention of the responsible party so that the problems can be promptly evaluated and resolved. The inspectors have seen ample evidence of this expectation at the manager level and above but have not seen this consistently at the individual worker level.

5.2 Corrective Actions

Progress had been made on reducing the large backlog of corrective actions, however some problems remained with timely and effective corrective actions. This was highlighted by the slow and inadequate evaluation of an unexplained reactor pressure vessel water level change (Section 3.3) and repetitive problems with the tone alert system (Section 4.2).

6.0 LICENSEE ACTION ON PREVIOUSLY IDENTIFIED ITEMS

NRC Inspection Procedures 92700, 92701, 92702, 92901, 92902, 92903 and 92904 were used to perform follow-up inspection of the items below.

6.1 Action on Licensee Event Reports (LER)

(Closed) LER 50-440/93-017-00: (Inadvertently closed as LER 50-440/93-012-00 in inspection report 50-440/95007, refer to that inspection report for the discussion of the LER) "Local Leak Rate Testing for Residual Heat Removal System Test Return Lines Not Performed in Accordance with 10 CFR Part 50, Appendix J, Requirements."

(Closed) LER 50-440/94-018-00: "Automatic Actuation of Annulus Exhaust Gas Treatment System (AEGTS) Standby Train." The cause of this event was a failure of an electronic signal selector card in the operating AEGTS train control loop. This resulted in a low flow signal and automatic start of the standby train as designed. The licensee replaced the failed card, tested the system satisfactorily, and returned it to normal operation. In addition, licensed plant operators received training on the event. The safety significance of this event was minimal.

6.2 Review of Previously Opened Items (Violations, Unresolved Items, Inspection Followup Items)

(Closed) Violation (50-440/94004-02(DRP)): "Failure to Maintain Adequate Cleanliness." This violation resulted from inspector identification of a rag left inside the Division 1 Emergency Diesel Generator (EDG) following maintenance after the EDG had been declared ready for testing. Subsequently work on the other two EDG's had additional controls implemented to assure that no foreign material was left in the engines following maintenance. Licensee procedure PAP 0204 "Housekeeping/Cleanliness Control Program" was revised with more specific instructions to strengthen material exclusion controls for open equipment and inspections where direct visual observations are not possible. The Division 1 EDG was reopened and inspected in greater detail and no additional foreign material was identified. In addition, vice president nuclear, managers, and supervisors meetings were and are planned to be held on specific frequencies with all employees to reaffirm the issues of ownership, accountability and management expectations. The meetings were also a response to violation (50-440/94004-01(DRP)) which was closed in inspection report 50-440/95007. This response appears thorough and subsequent observations indicate that the corrective actions have been effective. This violation is closed.

(Closed) Unresolved item (50-440/94006-04(DRP)): Craftsmen worked on wrong residual heat removal test line after failing to perform triple verification. The inspectors reviewed this item and determined that it was a violation (see Section 2.4). The superintendent and foreman involved in this work were also subjects of individual enforcement actions. This item is closed.

(Closed) Unresolved Item (440/94009-02(DRP)): "Preventive maintenance (PM) tasks were deferred without engineering evaluations." The licensee initiated Condition Report 94-350 to address this licensee-identified issue and concluded that the condition did not result in inoperable equipment. Corrective actions to prevent recurrence were considered appropriate. In addition, the ongoing PM optimization efforts and maintenance rule implementation will affect the overall program implementation and will require further reviews in the future. This item is closed.

(Closed) Violation (50-440/94010-01(DRP)): "Potential Siphon Path from Reactor Vessel." This violation resulted from a personnel error on the part of an operator and a supervisor where they failed to follow a procedure. Although it was promptly recognized (alarm response) and corrected, it created a potential to drain the reactor vessel during a shutdown condition. The operator and supervisor were counseled and received additional coaching on the plant simulator. The event was discussed by operations management with all shift crews with respect to procedure compliance and the importance of self checking. Subsequent observations indicate that this appears to have corrected the matter. In addition, the event along with several others was not reported to management in a timely manner. This resulted in an unresolved inspection item (URI 50-440/94010-02(DRP)) which was reviewed and closed in Inspection Report 50-440/95007.

Open) Unresolved item (50-440/94011-02(DRP)): This item was opened in response to a 10 inch Reactor Pressure Vessel (RPV) level decrease when the Residual Heat Removal system was opened to the RPV. The licensee provided a response to this item that is discussed in Section 3.3. This item remains open.

(Closed) Violation (50-440/94013-01 (DRP)): "Premature Reactor Power Increase." This violation resulted from a lack of command and control while restoring automatic recirculation system flow control following a failed LPRM in an APRM string. It involved breakdowns in procedure adherence, self checking, supervision and training by operators and a supervisor. The error was promptly identified and corrected by an operator when the reactor power increased to about 101 percent. The individuals involved were disciplined and/or counseled. All licensed operators reviewed the event and reactor recirculation system classroom and simulator training was conducted with all crews. These actions appear to have given all licensed operators a greater awareness of this type of issue. This violation is closed.

7.0 Plant Hardware Modifications to Reactor Vessel Water Level Instrumentation (NRC Bulletin 93-03) Temporary Instruction (TI) 2515/128 Revision 1

The purpose of this TI was to verify and evaluate implementation of hardware modifications to the reactor vessel water level instrumentation by the licensees in response to NRC Bulletin (NRCB) 93-03, "Resolution of the Issues Related to Reactor Vessel Water Level Instrumentation in BWR's." It was also an evaluation of the licensees performance implementing the requirements of 10 CFR 50.59 with respect to these design modifications.

The licensee chose the technique of continuous reactor vessel level reference leg backfill from the control rod drive (CRD) C11 system in order to assure that the reactor vessel level did not experience "notching" upon a rapid depressurization.

The inspectors reviewed the applicable documents, including the 10 CFR 50.59 modification packages, related procedures and corrective action documents. Various plant staff members were interviewed and the installations were walked down to observe them in operation. The following is a summary of the specific points of the TI related to the modification.

The preinstallation and post installation test results were reviewed and verified that the back flow was nominally 0.48 to 1.2 gallons per hour. This met the acceptance criteria of about 0.5 gallons per hour.

The 10 CFR 50.59 evaluations were reviewed and verified that the consequences of closure of a manual isolation valve were addressed. The evaluation of the worst case situations showed that the inadvertent closure of a manual isolation valve could result in Automatic Depressurization System (ADS) initiation, safety relief valve (SRV) lift, a half scram or an Emergency Core Cooling System (ECCS) initiation. In order to prevent these occurrences, the licensee has locked the associated reference leg manual isolation valves open and corrected drawings to reflect the change.

For administrative controls, an early compensatory measure was to install two computer alarms (wide range and narrow range) to annunciate if any level channel deviated by prescribed amounts from the average of the other channels. The licensee chose to retain these alarms after subsequent hardware modifications negated their requirement. Subsequently, the licensee changed procedures, such as control room alarm annunciator procedure, ARI-H13-P601-22 "CRD Pump Auto Trip," and PAP 0201, "Conduct of Operations," so that both appropriately addressed the CRD back fill to the reactor pressure vessel reference legs. Although the inadvertent closing of the manual isolation valves could have caused the events listed in the previous paragraph, the controls of computer alarms, locking the valves and changing drawings appeared to be acceptable preventive actions.

The potential for single or common mode failure mechanisms of the back flow system that could result in excess or reduced flow were addressed by redundancy in the components. For each reference leg back flow panel, duplicate pressure regulators, filters and throttle valves were used. No bypass valves were used.

Safety-related versus nonsafety-related interfaces were considered, with the boundary at the inlet to the upstream side of duplicate check valves in the backfill lines. In addition, duplicate surge suppressors were placed in the backfill control panel outlet line to the reference legs.

Containment isolation was provided by the duplicate check valves in series in the safety-related portion of the backfill system. These check valves were in the licensees ISI and 10 CFR Part 50, Appendix J, test programs. The test procedures were reviewed to verify that they met the acceptance criteria and found to be acceptable.

The licensee had been monitoring data on the system to evaluate the need for specific maintenance or surveillance activities such as filter changes or additional testing of equipment. In addition, the panels were part of the operator walkdowns with data recorded on rounds logs.

Procedures IMI-E2-55 (56, 57 and 58), "Reference Leg Purge Panel 1H51-P1432 A (B, C and D) Operation," used for the backfill system operation, startup, shutdown, and changing flow rates were reviewed and found to be acceptable.

Seismic protection was provided to the purge control panels by mounting them on the associated instrument racks.

Self assessment or self checking was good for this project. Routine QA/QC monitoring was performed on all phases of the modification as well as self checking by personnel involved. This resulted in several engineering hold orders that were properly dispositioned prior to continued testing or operation.

The hardware modification and associated activities meet the criteria of the TI and this TI is closed.

8.0 Persons Contacted and Management Meetings (Exit)

The inspectors contacted various licensee operations, maintenance, engineering, and plant support personnel throughout the inspection period. Senior personnel are listed below.

After the conclusion of the inspection on October 27, 1995, the inspectors met with licensee representatives (denoted by *) and summarized the scope and findings of the inspection activities. The licensee did not identify any of the documents or processes reviewed by the inspectors as proprietary.

D. C. Shelton, Senior Vice President
*R. D. Brandt, General Manager Operations
*N. L. Bonner, Engineering Director
*R. W. Schrauder, Nuclear Services Director
*K. R. Pech, Nuclear Assurance Director