U.S. NUCLEAR REGULATORY COMMISSION
REGION III

Report No. 50-440/88012(DRP)

Docket No. 50-440 License No. NPF-30

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

Facility Name: Perry Nuclear Power Plant, Unit 1

Inspection At: Perry Site, Perry, Ohio

Inspection Conducted: July 1 through August 23, 1988

Inspectors: K. A. Connaughton
G. F. O'Dwyer

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

Robert DeFayette

9/14/88
Date

Inspection Summary

Inspection on July 1 through August 23, 1988 (Report No. 50-440/88012(DRP))
Areas Inspected: Routine, unannounced inspection by resident inspectors of previous inspection items, Operational Safety Team Inspection (OSTI) findings, NRC Bulletins, 10 CFR Part 21 Reports, NRC Generic Letters, operational safety, nonroutine events, maintenance, surveillance, engineered safety (ESF) feature system walkdown, Licensee Event Reports, Allegations, onsite review committee activities, physical security, and radiological controls. A meeting between NRC and licensee management was conducted on July 26, 1988 at the NRC Region III office to discuss current plant status and recent events. Additionally, a meeting with local public officials was held on August 2, 1988, to discuss current plant status, Perry SALP 8 results, and the NRC organization, mission, and inspection programs.
Results: Of the 15 areas inspected, two violations were identified in one area (failure to implement established procedures for venting and filling the shutdown cooling system - Paragraph 3.a.; failure to implement established procedures for controlling the use of overtime for personnel performing safety related functions - Paragraph 3.b.); and one violation was identified in a second area (failure to follow procedure during surveillance testing of the standby liquid control system - Paragraph 7.b.). Additionally, one violation was identified in a third area; however, in accordance with the provisions of 10 CFR, Appendix C, Section V.A., a Notice of Violation was not issued (entry into Operational Condition 1 without having verified within the previous 12 hours that measured core flow was greater than or equal to established core flow - Paragraph 12.). Regarding the violation involving the standby liquid control system surveillance, the licensee is performing a detailed evaluation to determine root cause and contributing factors in order to reduce the likelihood of future similar human error-related events. With respect to Reportable Events, a marked reduction in frequency of occurrence was observed during the inspection period. The licensee operated at power throughout the inspection period.

DETAILS

1. Persons Contacted

a. Cleveland Electric Illuminating Company

+Alvin Kaplan, Vice President, Nuclear Group
C. M. Shuster, Director, Nuclear Engineering Department (NED)
+M. D. Lyster, General Manager, Perry Plant Operations Department (PPOD)
+R. A. Stratman, Manager, Operations Section, (PPOD)
M. Wesley, Acting Senior Operations Coordinator (PPOD)
V. K. Higaki, Manager, Outage Planning Section (PPOD)
F. R. Stead, Director, Nuclear Support Department (NSD)
W. R. Kanda, Manager, Instrumentation and Controls Section (PPTD)
S. F. Kensicki, Director, Perry Plant Technical Department (PPTD)
L. L. Vanderhorst, Radiation Protection Section (PPTD)
+R. A. Newkirk, Manager, Licensing and Compliance Section (NSD)
K. Pech, Manager, Technical Section (PPTD)
+E. Riley, Director, Nuclear Quality Assurance Department (NQAD)
T. A. Boss, Supervisor, Quality Audit Unit (NQAD)
D. J. Takas, Manager, Mechanical Maintenance Quality Section (NQAD)

b. U.S. Nuclear Regulatory Commission

#A. B. Davis, Regional Administrator, Region III
#K. E. Perkins Jr., Director, Project Directorate III-3, NRR
#+R. C. Knop, Chief, Projects Branch 3
+R. W. Cooper, II, Chief, Projects Section 3B
#T. G. Colburn, Project Manager, NRR
*K. A. Connaughton, Senior Resident Inspector
#*+G. F. O'Dwyer, Resident Inspector
#R. Lickus, Chief, State and Government Affairs, Region III
#R. J. Marabito, Public Affairs Officer, Region III

*Denotes those attending the exit meeting held on August 23, 1988.
+Denotes those attending the July 26, 1988 plant status meeting.
#Denotes those attending the meeting with local public officials on August 2, 1988.

2. Licensee Action on Previous Inspection Findings (92701)

a. (Closed) Open Item (440/85090-01(DRS)): Evaluation of unprotected cable tray supports against the effects of fire. This item was originally scheduled for closure prior to operation above 5% power. Based upon further inspector review documented in NRC Inspection Report No. 50-440/86024, this matter was rescheduled for resolution prior to first refueling. Open Item (440/86024-01(DRS)) was also issued to track this matter. Since these open items are redundant and the latter item correctly reflects the current schedule for resolution, this item is hereby administratively closed. The inspector notes that the licensee completed the requested analysis during this inspection period. The analysis will be forwarded to fire protection specialists in the NRC Region III Division of Reactor Safety for review.

b. <u>(Closed) Open Item (440/86027-01(DRP))</u>: Perform a followup review of the licensee's administrative controls for trending problems and identification of generic concerns. During an Operational Safety Team Inspection (OSTI) conducted by the NRC Office of Nuclear Reactor Regulation, a broadscope review of the licensee's corrective action programs was conducted, including the licensee's trending programs. This review determined that the licensee's programs were generally adequate. The inspection team noted that a number of weaknesses had been identified by the licensee and that program improvements were underway. Specifically, in the area of trending, the OSTI noted that work order trending required the manual transfer of data from the computer-based Perry Plant Maintenance Information System (PPMIS) to the Reliability Information Tracking System (RITS). This process was determined to be labor intensive and thus may have contributed to untimely and/or incomplete trending of problems documented on work orders. The licensee was in the process of expanding the PPMIS to provide for trending directly from the PPMIS database. This would eliminate the need to manually transfer data between data management systems and permit more timely and thorough work order trends. During this inspection, the inspector was informed by licensee personnel that the expansion of PPMIS was still underway and was expected to be completed by April 1989. Based upon the OSTI review of this matter and the licensee's ongoing efforts to improve the work order trending process, this item is considered closed. Further inspector review of the licensee's work order trending program will be conducted during future routine inspections.

c. <u>(Closed) Open Item (440/87007-01(DRS))</u>: Additional evaluation of Safety Relief Valve Weeping to determine the long term operational significance. This item is duplicative of Open Item (440/87016(DRS)) and as such, serves no useful purpose. This item is hereby administratively closed.

3. <u>Operational Safety Team Inspection Findings (92701)</u>

a. <u>Background</u>

On March 14 through 25, 1988, an NRC Operational Safety Team Inspection (OSTI) was conducted for Perry Unit 1 in order to assess licensee performance in a number of areas and to determine whether or not the licensee had successfully made the transition from a construction/preoperational test status to a fully operational status. This inspection was documented in NRC Inspection Report No. 440/88200. Several issues raised during the OSTI were referred to the NRC Region III office for potential enforcement action.

Issues for which enforcement action was determined to be warranted are discussed in the following sub-paragraphs. For each issue, the applicable portion of the OSTI report is referenced and specific details supporting the associated enforcement action are provided.

b. Failure to Implement Instructions for Shutdown Cooling System Fill and Vent Operation

As detailed in Section 2.1.2 of the OSTI inspection report, two instances were observed where operating activities with the capacity to affect facility safety were conducted in the plant without benefit of procedural control. On March 22, 1988, plant operators attempted to vent and fill shutdown cooling system piping, but incorrectly attached a vent hose to a valve in a separate portion of shutdown cooling piping located near the correct vent valve. On March 23, 1988, an OSTI team member observed performance of the same evolution without benefit of procedural guidance during valve manipulation. Confusion among plant operators was again evident to the OSTI inspector. At the time of both instances, an approved temporary plant rounds instruction was available, but not used. Failure to operate plant equipment in accordance with written approved procedures is contrary to Section 6.2.1 of Plant Administrative Procedure (PAP)-0201, "Conduct of Operations," Revision 3, and Technical Specification 6.8.1 and is a violation (440/88012-01(DRP)).

c. Overtime Deviation

As detailed in Section 2.1.4 of the OSTI inspection report, a review of the operations, maintenance and I&C departments overtime and attendance records for the three months preceding the inspection identified eight instances in which the overtime guideline limits specified in Plant Administrative Procedure (PAP)-0110, "Shift Staffing and Overtime," Revision 2, had been exceeded, but had not been approved in advance as required.

Specifically, Section 6.5.4 of PAP-0110 stated the following:

"The following guidelines are applicable to personnel performing safety-related functions and key maintenance activities. These guidelines may be extended to include personnel not in these categories at the discretion of the section General Supervisor.

1. An individual should not work more than 16 hours straight (excluding shift turnover time).

2. An individual should not work more than 16 hours in any 24 hour period (excluding shift turnover time).

3. An individual should not work more than 24 hours in any 48 hour period (excluding shift turnover time).

4. An individual should not work more than 72 hours in any seven day period (excluding shift turnover time).

5. A break of at least eight hours should be allowed between work periods (including shift turnover time).

6. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift or crew."

The eight instances identified by the OSTI included:

- One power plant operator (PPO) worked approximately 30 hours in a 48 hour period without an approved Overtime Deviation Request.

- One PPO worked approximately 31.5 hours in a 48 hour period, excluding shift turnover time, and also worked approximately 79 hours in a seven day period, excluding shift turnover time, without an approved Overtime Deviation Request.

- One PPO worked approximately 76 hours in a seven day period, excluding shift turnover time, without an approved Overtime Deviation Request.

- One PPO worked approximately 31.5 hours in a 48 hour period, excluding shift turnover time, without an approved Overtime Deviation Request, and also worked approximately 83 hours in a seven day period, excluding shift turnover time, without an approved Overtime Deviation Request.

- One PPO worked approximately 28 hours in a 48 hour period, excluding shift turnover time, without an approved Overtime Deviation Request, and also worked approximately 79.5 hours in a seven day period, excluding shift turnover time, without an approved Overtime Deviation Request.

- One I&C technician worked approximately 25 hours in a 48 hour period, excluding shift turnover time, without an approved Overtime Deviation Request.

- One I&C technician worked approximately 73 hours in a seven day period, excluding shift turnover time, without an approved Overtime Deviation Request.

The OSTI also identified one instance when blanket authorization to exceed the overtime guideline limit was approved. This instance had previously been identified by the licensee's Quality Assurance Department.

The foregoing examples of failure to authorize in advance deviations from the overtime guidelines of PAP-0110 is contrary to Technical Specification 6.2.2.e. and is a violation (440/88012-02(DRP)).

d. Plant Logs and Records

During the OSTI, a number of instances where items recommended for inclusion in the Unit Log by Operations Administrative Procedure (OAP)-1702, "Operations Section Rounds, Logs, and Records," Revision 5, were not included in the Unit Log. While these omissions did not constitute procedural violations, the inspector continues to be concerned that the Unit Log does not always include all items encompassed by the OAP-1702 recommendations. The inspector has expressed this concern to licensee management and will continue to review the adequacy of operating logs and licensee actions to improve their content and usefulness. This matter will be tracked as an open item (440/88012-03(DRP)).

A related concern identified during the OSTI was the use of a loose leaf binder for the Unit Log instead of a bound log. Prior to and during this inspection, the inspector verified during routine operating log reviews that the licensee re-established the Unit Log as a bound document. The inspector has no further concerns regarding this matter.

4. NRC Bulletin Followup/Temporary Instruction 2500/26 (92701,25026)

a. (Closed) NRC Bulletin (440/87001-BB): "Thinning of Pipe Walls at Nuclear Power Plants," Sims No. 139. By letter dated September 8, 1987, the licensee provided the documented response required by this NRC Bulletin. NRC staff in the NRC Office of Nuclear Reactor Regulation reviewed the licensee's response and informed the licensee by letter dated March 29, 1988, that no further action with respect to this NRC Bulletin was required. Since submittal of the requested information was the only action required, no additional followup inspection activity associated with this NRC Bulletin is planned.

b. (Closed) NRC Bulletin (440/87002-BB): "Fastener Testing to Determine Conformance With Applicable Material Specifications." Inspection requirements associated with this NRC Bulletin are specified in NRC Inspection Manual Temporary Instruction (TI) 2500/26. During a previous inspection documented in NRC Inspection Report No. 50-440/87023, the inspector participated in the selection of fasteners to be tested in accordance with the requirements of this NRC Bulletin. During this inspection, the inspector reviewed the licensee's response submittal dated January 19, 1988, against inspector notes generated during the sample selection and identification process and determined that traceability of the samples had been maintained and that testing performed on each sample was consistent with applicable specifications, grades, and classes. The inspector further verified that samples procured to ASME Section III requirements were appropriately identified and subjected to impact testing.

The inspector reviewed licensee procedures for receipt inspection of fasteners and determined that current fastener receipt inspections included all of the attributes specified in the licensee's response submittal. Prior to September 1987, receipt inspection procedures for bolts generally included the standard inspection attributes contained in Nuclear Quality Assurance Department Procedure (NQADP)-1003, "Inspection of Procured Items." The standard attributes contained in NQADP-1003 are those specified in the licensee's response submittal as currently applicable to nuts. Enhancement of the licensee's receipt inspection requirements for bolts was prompted by the issue raised in the subject NRC Bulletin (i.e., fasteners which did not possess chemical and/or physical properties specified in certification documentation).

The inspector reviewed current revisions of the following licensee procedures applicable to the issuance and control of safety-related and nonsafety-related fasteners:

PAP-0905, "Work Order Process"
PAP-0402, "Material Request Processing"
SMI-0001, "Stock Code Creation"
SMI-0005, "Material Issue"

The inspector determined that the requirements contained in the forgoing procedures were consistent with the description of the material issuance and control activities contained in the licensee's response submittal.

No violations or deviations were identified.

5. 10 CFR Part 21 Report Followup (92701)

a. (Closed) 10 CFR 21 Report (440/86001-PP) (DAR-262): Defective high voltage power supply circuitry in Kaman post-accident radiation monitors. Licensee investigation into the reported problem determined that upon energizing the radiation monitors, high voltage power supply output voltage spiked to approximately 1300 volts DC. Mineral insulated detector cabling experienced momentary insulation failure resulting in detector high voltage oscillation. Continued voltage oscillations could have led to complete failure of the detector cable. The defective high voltage power supplies were sent to the manufacturer and modified to limit output voltage spike upon energization to less than 1000 volts DC. Subsequent preoperational testing was performed to verify that the problem had been satisfactorily resolved. Previous inspector review of this matter, including verification of preoperational test completion, was conducted during followup inspection of Open Item (440/85022-21(DRP)) documented in NRC Inspection Report No. 440/86018.

b. <u>(Closed) 10 CFR 21 Report (440/88001-PP)</u>: Eaton/Cutler-Hammer motor control center (MCC) anomalies involving conical spring washers and stab/bus interface mechanism (a stab is a conductor which connects a circuit breaker to an associated bus bar). The specific conditions identified in this report were: cracking of the conical spring washers due to hydrogen embrittlement and/or corrosion, and evidence of high temperature conditions (discoloration and corrosion) at the stab/bus interface. The inspector verified by document review that the licensee had been informed of the reported anomalies by letter dated July 6, 1988, from Eaton/Cutler-Hammer. The licensee performed an investigation to determine wether or not the anomalous conditions had been detected during preventive and corrective maintenance activities associated with the 30 potentially affected class 1E MCCs utilized for Perry, Unit 1.

Inspections of 21 of the MCCs conducted between June 1986 and May 1988 did not identify any discoloration indicative of excessive heat at the stab/bus interfaces, loose hardware on joints, or corrosion. The inspections, which were conducted as part of periodic, preventive maintenance procedures, included QA witness points for inspection of 10% of the electrical bus bolted connections. Additionally, the preventive maintenance procedures required that, upon identification of evidence of corrosion, QA was required to be notified prior to initiating actions to remove the corrosion. If a Size 3 or 4 starter was removed, rust inhibitor/lubricant was required to be applied to the bus stabs prior to reinsertion, thereby reducing the likelihood of excessive wear and corrosion at the bus/stab interface. Identified problems which were not corrected in the course of preventive maintenance activities would have required issuance of a work order. Licensee review of maintenance and quality assurance records associated with the preventive maintenance activities and all corrective maintenance work orders associated with the affected MCCs did not identify any evidence of the reported conditions. The licensee therefore concluded that no actions in addition to scheduled preventive maintenance were warranted.

The inspector reviewed the current revision of General Electrical Instruction (GEI)-0006, "General Maintenance of Motor Control Centers" which was used to direct the preventive maintenance activities referenced above. The inspector verified that the procedure provided for inspection of the MCCs consistent with the requirements cited in the licensee's evaluation. Based upon the lack of evidence of the anomalies and the preventive maintenance requirements applied to the MCCs at Perry, the inspector has no further concerns regarding this matter.

No violations or deviations were identified.

6. Generic Letter Followup (92701)

a. (Closed) Generic Letter (440/85006-HH): Sims No. NPA-A-20 "Quality Assurance Guidance for ATWS Equipment That is Not Safety-Related." The subject Generic Letter was issued to the licensee for consideration in the development and application of quality assurance requirements for nonsafety-related equipment encompassed by 10 CFR 50.62. No specific action was required of the licensee. Inspector followup of this Generic Letter therefor consisted of verification that the licensee received it and distributed it for review in accordance with administrative procedures. The inspector noted that the matters discussed in this Generic Letter would be the subject of a future inspection to be conducted in accordance with NRC Inspection Manual Temporary Instruction (TI) 2500/20.

b. (Closed) Generic Letter (440/86002-HH): "Technical Resolution of Generic Issue B-19 - Thermal Hydraulic Stability," (Sims No. B-19). This Generic Letter was issued to inform licensees that the NRC staff had concluded that the adoption of operating limitations which provide for the detection and suppression of flux oscillations in operating regions of instability, consistent with the recommendations of General Electric SIL-380, were acceptable to demonstrate compliance with General Design Criteria 10 and 12 for cores loaded with approved fuel designs. The inspector verified by review of licensee file information that the licensee had received and reviewed the Generic Letter in accordance with licensee administrative procedures. The licensee adopted the recommendations of General Electric SIL-380 and, in the Perry Safety Evaluation Report, Supplement 7, the NRC staff concluded that the Perry Technical Specifications were adequate to address this issue. Operating procedures were revised to implement Technical Specification operational restrictions and to provide the recommended guidance to plant operators for dealing with circumstances which place the reactor in a potentially unstable condition. The inspector concluded that licensee actions relative to the subject Generic Letter were appropriate. The inspector noted, however, that as a result of an event involving flux oscillations at LaSalle, Unit 2 on March 9, 1988, the NRC staff issued NRC Bulletin 88-07 which required additional licensee actions to address this issue. Future followup inspection will be conducted for NRC Bulletin 88-07.

No violations or deviations were identified.

7. Operational Safety Verification (71707)

The inspectors observed control room operations, reviewed applicable logs, and conducted discussions with control room operators during this inspection period. The inspectors verified the operability of selected emergency systems, reviewed tag-out 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 observed plant housekeeping/ cleanliness conditions and verified implementation of radiation protection controls.

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

During this inspection period, the licensee completed tube leakage repairs on the 6A feedwater heater and returned it to service. Based upon operating experience at other operating plants and the existing piping configuration for venting of the 6A heater, the licensee was concerned that during heater restoration, entrapped air may have been swept through the reactor vessel with a high potential for causing a main steam line isolation and reactor scram with the main steam line radiation monitors in service. The licensee determined that a reduction in power to less than 20% of rated may have permitted system restoration and prevented a main steam line high radiation trip due to the activation of the air to nitrogen-16. However, based upon the nature of underlying assumptions, there was no guarantee that prevention of a reactor scram would have been accomplished via power reduction.

Based upon this consideration and the fact that a significant power reduction was not advisable under the given electrical transmission system conditions, the licensee proposed bypassing the main steam line high radiation isolation and scram functions during feedwater heater restoration. Bypassing these scram and trip functions were permissible under Perry Unit 1 Technical Specification for periods of up to one hour for maintenance and surveillance testing. Due to the NRC's sensitivity towards voluntary entry into Technical Specification limiting conditions for operation, the licensee presented this proposal to representatives of the NRC Region III Office and the Office of Nuclear Reactor Regulation during discussions on August 2 and 5, 1988. NRC staff concurrence with the licensee's proposed course of action was obtained on August 5, 1988, provided that the licensee station a licensed operator to monitor main steam line radiation indications throughout the heater restoration activity so that, if necessary, main steam line isolation and reactor scram signals could be inserted manually.

On August 7, 1988, the licensee restored the 6A feedwater heater to service. No change in main steam line radiation levels were observed throughout the evolution. The Senior Resident Inspector was in the Control Room throughout the evolution.

No violations or deviations were identified.

8. <u>Followup of Nonroutine Events at Operating Power Reactors (93702)</u>

a. During the inspection, unseasonably warm weather resulted in higher than normal Lake Erie water temperatures. On July 27, 1988, with Lake Erie water temperature at 79 degrees F, operating personnel initiated Facility Change Request (FCR) No. 10090 requesting that the licensee's engineering organization evaluate Lake Erie and Emergency Service Water (ESW) temperature limits contained in the Perry U.S.A.R. At the time, the Perry U.S.A.R. assumed the maximum values of Lake Erie and RHR heat exchanger ESW inlet temperatures to be 80 degrees F. While Lake Erie water temperature had not exceeded 80 degrees F, ESW "A" pump discharge temperature indicators had been observed reading as high as 82 degrees F. The response to FCR 10090, issued on July 30, 1988, stated that the current design basis temperature for ESW at the inlet of the RHR heat exchangers was 80 degrees F, but that calculations had been performed to document that ESW temperatures of 82.2 degrees F would provide the design basis RHR heat exchanger heat transfer. Additionally, the FCR response stated that with the ESW system flow redistributed to provide additional flow to the RHR heat exchangers, ESW temperatures of as high as 84.4 degrees F would be acceptable.

On July 30 and August 3, 1988, following initiation of suppression pool cooling utilizing the "A" ESW loop, ESW pump discharge temperature was observed to be between 84.4 and 87 degrees F. In each instance, temperature instrumentation on the associated RHR heat exchanger outlet gave conflicting readings of between 80 and 82 degrees F. Licensee personnel believed that ESW temperature decreased between the ESW pump discharge and the RHR heat exchanger due to heat losses to the ground surrounding the buried ESW piping. As the ESW loop continued to operate, ESW pump discharge temperature dropped to 82 degrees F within the following three hours. Since the temperatures specified in the response to FCR 10090 appeared to have been temporarily exceeded at the ESW pump discharge, Condition Report No. 88-187 was issued on August 5, 1988, to require additional evaluation. Also on August 5, 1988, an ESW pump was placed in continuous service. On August 13, 1988, the ESW system was rebalanced to raise the maximum permissible RHR heat exchanger ESW inlet temperature to 84.4 degrees F. On August 14, 1988, FCR No. 10301 was issued which demonstrated that RHR heat exchanger ESW inlet temperature could be as high as 85 degrees F and still provide the design basis heat transfer.

While the licensee has taken action to ensure that the elevated ESW temperatures were not a significant safety concern, it was not clear to the inspector that technical justification revising the design basis Lake Erie and ESW maximum temperature limits existed prior to the occurrences of elevated temperatures on July 28, 30 and August 3, 1988. This matter is an Unresolved Item (440/88012-04(DRP)).

b. On August 8, 1988, the licensee determined that there was a valid alarm on Loose Part Monitoring System (LPMS) Channel 7, monitoring the "A" recirculation pump discharge riser. An evaluation of the LPMS indications and their potential safety significance was conducted by the licensee in consultation with General Electric over the following 72 hours. A determination was made that continued operation was justified for the following seven days pending acquisition and evaluation of further LPMS data. LPMS experts from General Electric, Gilbert Associates Inc. (the A/E), and Rockwell (the LPMS vendor) gathered and analyzed additional data and on August 18, 1988 presented the following conclusions to the licensee: (1) the LPMS data was not indicative of a loose part impacting or moving within the reactor system; (2) the amplitude of the signal component responsible for the LPMS alarm is recirculation flow dependent and may be due to LPMS sensor resonance in the 10 to 11 kHz range; (3) recirculation system performance was normal over the timeframe in which the LPMS alarms were received; (4) there is no safety significance to continued operation with the LPMS alarms, and (5) segmented surveillance of the LPMS signals should continue to detect changes in the signal frequency content indicative of a change in conditions. Additional data will be collected from other BWRs to determine if the dominant signal component is related to recirculation pump resonant frequencies. An attempt will be made to determine the mounted resonant frequencies of the LPMS acoustic sensors. A final formal report will be issued by October 1988.

Inspector review of additional actions to be taken by the licensee to address the loose part indications is an Open Item (440/88012-05(DRP)).

c. On August 12, 1988, while performing surveillance test instruction (SVI) C41-T2001, "Standby Liquid Control System Pump and Valve Operability Test" both trains of the standby liquid control system (SLCS) were simultaneously rendered inoperable. As written, the test instruction first rendered the "A" train inoperable for valve testing then restored it to operability. The "B" train was then rendered inoperable for valve testing and restored to operability. Pump testing was then performed in a similar manner. The "A" train was rendered inoperable for pump testing and restored; the "B" train was rendered inoperable for pump testing and then restored. At the end of the testing, the procedure requires the verification and independent verification of restoration of both trains to operable status.

During the "B" train valve testing, standby liquid control pump manual suction isolation Valve 1C41-F002B was closed. The valve was not reopened as required by the instruction prior to initiating the "A" train pump test portion of the procedure. As a result, during testing of the "A" pump ("A" train rendered inoperable), both SLCS trains were rendered inoperable for a period of approximately 1 hour and 45 minutes. The mispositioned valve was not discovered until the "B" train pump test was attempted and operators noted that pump discharge pressure remained low following pump start. Following this observation, approximately 20 seconds after pump start, the pump was secured in order to prevent pump damage. Subsequently, the pump was tested with satisfactory results. While the cause of the valve mispositioning and SLCS inoperability appeared to be personnel error, a contributing factor was that independent verification of valve lineup was not required each time a train was to be restored to operable status. Failure to perform the surveillance test in accordance with SVI C41-T2001 was contrary to Plant Administrative Procedure (PAP)-0201, "Conduct of Operations," Revision 3, Section 6.2.2.d. and Technical Specification 6.8.1 and is a violation (440/88012-06(DRP)).

9. Monthly Maintenance Observation (62703)

Station maintenance 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.

The following maintenance activities were observed/reviewed:

- From July 11 to 22, 1988, the installation of interlocks on the Combustible Gas Purging Compressor "A" in accordance with Work Order (WO)-2949, Revision 0.
- From August 8 to 17, 1988, troubleshooting/repair of the Drywell Atmosphere Radiation Monitor, 1D17-K670 as documented by WO 88-5863, Revision 0. The root cause of the problem was not clearly documented on WO 88-5863. Additionally, the inspector could not determine if WO 5863, which authorized troubleshooting and repair activities permitted the installation of a new differential pressure switch with a range different from that originally employed to provide sample flow indication and alarm functions. This matter is an Open Item (440/88012-06(DRP)).

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

10. Monthly Surveillance Observation (61726)

On August 10, 1988, the inspectors observed Technical Specifications required testing that was conducted in accordance with Surveillance Instruction (SVI)-B21-T0034-B, Revision 2, "Reactor Vessel Level 3 and Level 8 Reactor Protection System and Residual Heat Removal Isolation Channel B Functional for 1B21-N680B," and verified that testing was performed in accordance with procedures, that test instrumentation was calibrated, that limiting conditions for operation were met, that removal and restoration of the affected components were accomplished, that test results conformed with Technical Specifications and procedure requirements and were reviewed by personnel other than the individual directing the test, and that any deficiencies identified during the testing were properly reviewed and resolved by appropriate management personnel.

No violations or deviations were identified.

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

From August 10 to 17, 1988 of this inspection period, the inspector performed a detailed walkdown of the accessible portions of the High Pressure Core Spray (HPCS) system, the Division 3 HPCS diesel and the diesel support systems. The system walkdown was conducted using the following Valve Lineup Instructions (VLI):

VLI-E22A, Revision 3, "High Pressure Core Spray (Unit 1)"

VLI-R44/E22B, Revision 2, "Division 3 Diesel Generator (DG) Starting Air System (Unit 1)"

VLI-R45/E22B, Revision 2, "Division 3 Diesel Generator Fuel Oil System (Unit 1)"

VLI-R46/E22B, Revision 3, "Division 3 Diesel Generator Jacket Water System (Unit 1)"

VLI-R47/E22B, Revision 2, "Division 3 Diesel Generator Lube Oil System (Unit 1)"

Prior to conducting the walkdown, the inspector verified the VLIs against the controlled Piping and Instrumentation Diagrams (P&IDs) for the HPCS System, the HPCS diesel and the diesel support systems.

During the walkdown, the HPCS system was identified by the licensee as an operable ECCS system in accordance with Technical Specifications. During the walkdown, the inspector directly observed equipment conditions to verify that housekeeping was adequate; no prohibited ignition sources or flammable materials were in the vicinity; valves and dampers in the system were installed correctly and did not exhibit gross packing leakage, bent stems, missing handwheels, or improper labeling; major system components were properly labeled, lubricated, and cooled and exhibited no leakage. The inspector verified that instrumentation was properly installed and functioning and that process parameter values were consistent with normal expected values; valves and dampers were in their proper positions and local and remote indications were functional; essential support systems were operational; and the electrical and control board lineups were proper.

No violations or deviations were identified.

12. Licensee Event Reports Followup (92700)

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

LER 87067-1L	Improper Maintenance Results in the B Main Steam Line Penetration Exceeding Technical Specification Leakage Limit
LER 87070-LL	Loss of Reactor Protection System Bus Due to an Over Voltage Trip of the Electrical Protection Assembly Results in a Division I Balance of Plant Isolation
LER 87070-1L	Loss of Reactor Protection System Bus Due to an Over Voltage Trip of the Electrical Protection Assembly Results in a Division I Balance of Plant Isolation
LER 87074-LL	Flow Indication Inaccuracy Results in Indicated High Differential Flow and Reactor Water Cleanup Isolation
LER 88006-LL	Loss of Reactor Protection System Bus Due to an Over Voltage Trip of the Motor-Generator Output Circuit Breaker Results in a Division II Balance of Plant Isolation
LER 88006-1L	Loss of Reactor Protection System Bus Due to an Over Voltage Trip of the Motor-Generator Output Circuit Breaker Results in a Division II Balance of Plant Isolation
LER 88009-LL	Misadjustment of Average Power Range Monitor Readings Due to an Error in Heat Balance Calculation Results in Technical Specification Violation
LER 88011-LL	A Momentary Decrease of the Diesel Generator Control Tachometer Resulted in an Unexpected Start of the Diesel Generator Building Ventilation System
LER 88012-LL	Improper D.C. Bus Transfer Due to Operating Error Results in a Complete Loss of Feedwater and a Reactor Scram on Low Reactor Water Level

LER 88017-LL Blown Fuse in RPS During Surveillance Causes Reactor Scram of Group 3 Control Rods Resulting in Reactor Vessel Low Level 3 and Full Scram

LER 88019-LL Failure of Chiller Linkage and Fan Power Supply Causes Loss of Both Trains of Control Room Ventilation and Entry Into Technical Specification 3.0.3.

LER 88020-LL Operator Error Causes Inadvertent Transfer of Recirculation Flow Control System to Flux Auto, Resulting in Reactor Scram on High APRM Levels

LER 88021-LL Deenergization of Reactor Protection System Bus A Due to Operator Error Results in Balance of Plant Isolation

LER 88022-LL Failure to Complete Surveillance Requirement Prior to Operational Condition Change Results in Technical Specification Violation

LER 88023-LL Reactor Scram and Containment Isolation Caused by Loss of Electrical Distribution Busses Due to Inadvertent Breaker Operation

LER 88024-LL Reinsertion of Automatic Flux Control Card Causes Spurious Spike in Recirculation Flow Resulting in Upscale Trip of APRMs and Reactor Scram

LER 88025-LL Overtravel of the Reactor Protection System Power Transfer Switch Results in a Loss of Power to Both Busses and a Full RPS Actuation

LER 88026-LL Reactor Scram Due to Unexpected Main Turbine Trip Caused by Mechanical Failure of Turbine Trip Latch Assembly

Regarding LER 87067-1L, the event described in this LER was the subject of violation (440/87004-02(DRP)). Licensee corrective actions were determined to be satisfactory and the violation was closed in NRC Inspection Report No. 50-440/88009.

Regarding LERs 87070-LL, 87070-1L, 88006-LL, and 88006-1L, licensee investigation determined that the RPS buses were subject to small, random voltage disturbances and that the over voltage trip settings were overly restrictive. Corrective actions included replacement of the RPS MG set voltage regulators and output voltage rheostats, raising the overvoltage trip settings, and establishing repetitive tasks (preventive maintenance requirements) to periodically clean, adjust, and secure the output voltage rheostats.

Regarding LER 88009-LL, the event described in this LER was the subject of violation (440/88004-02(DRP). The violation will track inspector followup of licensee corrective action implementation.

Regarding LERs 88012-LL, 88017-LL, 88019-LL, 88020-LL, 88023-LL, 88024-LL, and 88026-LL, inspector followup activities and findings relative to each event were documented in NRC Inspection Report No. 440/88009.

Regarding LER 88022, operating personnel failed to perform a verification that measured core flow was greater than established core flow which is used in the Average Power Range Monitor flow-biased scram circuitry prior to entry into Operational Condition 1 on June 7, 1988. Operators assumed that the surveillance was not required to be performed until the next regular surveillance interval. Licensee investigation into this event disclosed two previous similar occurrences attributable to the same misunderstanding on the part of plant operators. In all instances, the surveillances were performed within the 12 hours following the entry into Operational Condition 1. The inspector verified that Integrated Operating Instruction (IOI)-1, "Cold Startup" and IOI-2, "Hot Startup" had been revised to explicitly require performance of this surveillance requirement just prior to entry into Operational Condition 1. These revisions were accomplished via Temporary Change Notice (TCN) No. 4 to IOI-1 and TCN No. 4 to IOI-2. Failure to verify that measured core flow is greater than established core flow within 12 hours of entry into Operational Condition 1 is contrary to Technical Specification 3.0.4 and is a violation (440/88012-08(DRP)); however, since this violation meets the tests of 10 CFR 50, Part 2, Appendix C, Section V.A, a written Notice of Violation was not issued.

13. Allegation Followup (99014)

Discussed below is an allegation brought to the attention of NRC Region III. This allegation was evaluated when received to determine need for immediate onsite followup; such need was not indicated. Further reviews were performed during this inspection.

On January 28, 1988, the Nuclear Regulatory Commission received a telephone call from an individual who wished to remain anonymous. The subject of the call and followup discussions are presented below (AMS No. RIII-88-A-0019).

Allegation:

The yoke cracked on Valve No. 1E22-F011 and a Nonconformance Report (NR) was written. The NR went to PPTD for disposition to get the valve repaired. However, in violation of Procedure No. IWA 7000, there was no attempt by the utility to analyze the cause for the failure of the valve.

Discussion:

Inspector review of this allegation determined that the alleger's reference to Procedure No. IWA 7000 was, in fact, a reference to the ASME Code, Article IWA-7000, "Replacements," which requires, in part, that prior to authorizing the installation of a replacement, the owner shall conduct an evaluation of the suitability of the replacement. If a replacement is required because of failure of a part or component, the evaluation shall consider cause(s) of failure of the existing part or component to ensure that the selected replacement is suitable. The inspector reviewed Nonconformance Report (NCR) No. MMQS-3100 and revisions thereto. The subject NR was dispositioned "Rework" with documented justification. Following disassembly and inspection of the damaged valve, Revision 3 to the subject NR was issued which added a documented root cause evaluation, a design change to prevent recurrence, and an evaluation of the adequacy of the original design. Specifically, it was determined that the cracked yoke on Valve 1E22-F011 was caused by failure of the stem clamp to remain in place on the stem key as a result of a loose set screw. A Design Change Package (DCP 880040) which replaced the existing keys with "L" shaped keys was implemented to prevent movement of the stem clamp along the stem axis in the event that the set screw became loose. A documented evaluation of the original design for Valve 1E22-F011 and similar valves concluded that the design was adequate, provided preventive maintenance included periodic inspection of the set screws and that the set screws are properly restaked following maintenance. Preventive maintenance instructions were revised to inspect the valve set screw once every six months. In addition to the subject NR, the inspector reviewed documentation associated with DCP No. 880040 including 10 CFR 50.59 evaluations and Work Order Nos. 88-324 and 88-1201 which authorized and controlled the rework activities associated with Valve 1E22-F011. The inspector determined that the evaluation required by the ASME Code, Section XI, Article IWA-7000 was performed well in advance of valve rework completion which was accomplished on April 28, 1988. This allegation is, therefore, unsubstantiated.

No violations or deviations were identified.

14. Onsite Review Committee (40700)

The inspectors reviewed the minutes of the Plant Operations Review Committee (PORC) Meetings No. 88-046 through 88-077, conducted prior to and during the inspection period to verify conformance with PNPP procedures and regulatory requirements. These observations and examinations included PORC membership, quorum at PORC meetings, and PORC activities.

No violations or deviations were identified.

15. Physical Security Procedures For The Resident Inspector (71881)

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

No violations or deviations were identified.

16. <u>Radiological Protection Procedures For The Resident Inspector (71709)</u>

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

During the course of routine inspection activities conducted during this inspection period, the inspectors accessed plant areas requiring a radiation work permit (RWP). The inspectors reviewed the radiation work permits and verified that, in accordance with licensee procedures, the RWPs contained a description of activities authorized, radiation levels, contamination levels, protective clothing requirements, dosimetry requirements, health physics coverage requirements, expiration dates, and required review and approval signatures. The RWPs were determined to be current and readily available for employee review. Work activities observed by the inspectors were conducted in accordance with RWP requirements.

Inspector observation of personnel within the radiologically controlled area determined that personnel monitoring equipment was properly utilized and that dosimeter readings were recorded as required upon leaving the radiologically controlled area. Personnel exiting the radiologically controlled area were observed to properly utilize personal contamination monitors. Posting of radiation areas, contaminated areas, and labeling of containers holding radioactive material were determined to be in conformance with NRC regulations and licensee procedures.

No violations or deviations were identified.

17. <u>Violations For Which A "Notice of Violation" Will Not Be Issued</u>

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 licensees' 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.A. 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 for which a Notice of Violation will not be issued are discussed in Paragraph 12.

18. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, violations, or deviations. An unresolved item is identified in Paragraph 8.a.

19. Open Inspection Items

Open inspection items are matters which have been discussed with the licensee, which will be reviewed further by the inspector, and which involve some action on the part of the NRC or licensee or both. Open inspection items disclosed during the inspection are discussed in Paragraphs 3.c., 8.b., and 9.

20. Plant Status Meetings (30702)

On July 26, 1988, NRC management met with CEI management at the NRC Region III office to discuss the current status of the plant, recent events and licensee initiatives to improve the quality of plant operating and maintenance activities. These meetings are being held on a periodic (initially monthly) basis

21. Information Meeting With Local Public Officials (94600)

On August 2, 1988, at 6:00 p.m. through 8:00 p.m., the NRC personnel listed in Paragraph 1.b. met with local officials in the Nurse's Auditorium at Lakeland Community College. NRC personnel discussed mission of the NRC, the resident inspector program, current plant status, location of the Local Public Document Room (LPDR), and related information. Following the NRC presentation, questions from local public officials were addressed.

22. Exit Interviews (30703)

The inspectors met with the licensee representatives denoted in Paragraph 1 throughout the inspection period and on August 23, 1988. 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.