U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-440/85066 (DRP); 50-441/85023 (DRP)

Docket No. 50-440; 50-441 License No. CPPR-148; CPPR-149

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

Facility Name: Perry Nuclear Power Plant, Units 1 and 2

Inspection at: Perry Site, Perry, OH

Inspection Conducted: September 9-13 and September 16-20, 1985

Inspector: C. H. Scheibelhut 10/2/85
Date

Approved by: R. C. Knop, Chief 10/4/85
Reactor Projects, Section 1A Date

Inspection Summary

Inspection on September 9-13 and September 16-20, 1985 (Reports No. 50-440/85066 (DRP); 50-441/85023 (DRP))

Areas Inspected: Routine safety inspection by a Regional Inspector of licensee actions on previous inspection findings, 10 CFR 50.55(e) items, and evaluation of licensee action with regard to IE Bulletins, IE Circulars, allegations, confirmatory items called for in the Safety Evaluation Report, and surveillance procedures. The inspection involved a total of 68 inspector-hours onsite by one NRC inspector and includes 0 inspector-hours during off-shifts.

Results: No violations, deviations, or safety significant issues were identified.

Details

1. Persons Contacted

Cleveland Electric Illuminating Company

*F. R. Stead, Manager, Nuclear Engineering Department (NED)
*C. M. Shuster, Manager, Nuclear Quality Assurance Department (NQAD)
*E. C. Willman, Senior Engineer, NED
*F. H. Sonderoth, Senior Engineer, NED
*B. S. Ferrell, Licensing Engineer, NED
*B. D. Walrath, General Supervising Engineer, NQAD
*R. D. Segen, Project Quality Engineer, NQAD
*R. O. Neuendorf, Associate Engineer, NQAD
*V. K. Higaki, Unit Supervisor, NQAD
*B. B. Liddell, Operations Engineer, Perry Plant Technical Department (PPTD)
*G. Chasko, Surveillance Engineer, PPTD
*P. A. Russ, Compliance Engineer, PPTD
*N. J. Lehman, Staff Analyst, PPTD

The inspector also interviewed other licensee and contractor personnel during the course of the inspection

2. Licensee Actions on Previously Identified Items (92702)

(Open) Violation (440/84015-02(DRP)): "Instrument air system cleanliness not verified during testing". In Inspection Report 50-440/85039 (DRP), 50-441/85017 (DRP), the inspector was satisfied with the corrective actions taken by the licensee, but had concerns for future system cleanliness following maintenance or design changes because the instrument air system itself is not safety related.

The licensee responded to these concerns by revising Plant Administrative Procedure (PAP)-0905, "Work Order Process" to include the instrument air system on the "Quality Augmented" system list. This inclusion assures that any work on the instrument air system has proper quality assurance coverage and documentation and also assures that the air quality will be tested after work is complete to determine if there is any degradation in quality.

The inspector reviewed the revised PAP and found it satisified his concerns.

However, the NRC Office of Nuclear Reactor Regulation (NRR) has sent the licensee questions concerning Amendment 15 to the FSAR that changed the maximum allowable size of particulate contamination in the air from three micrometers to 40 micrometers. Therefore, this item will remain open until NRR accepts the change in the next supplement to the Safety Evaluation Report.

No violations or deviations were identified.

3. Safety Evaluation Report Follow-up Inspection Items (92701)

The Office of Nuclear Reactor Regulation (NRR) has requested that Region III inspectors confirm that the licensee has acceptably implemented certain Three Mile Island (TMI) action items and confirmatory issues as described in the Safety Evaluation Report (SER) for Perry, Unit 1 (NUREG-0887 with Supplements 1 through 6). In Inspection Reports 50-440/85022, 50-441/85012, and 50-440/85033, these items were listed as Open Items and entered into the Region III tracking system for future inspection. The items reported below are the results of the inspection of certain of the items.

a. (Closed) SER Open Item (440/85022-26 (DRP)): "TMI Item II.K.1.5; Determine that engineered safety feature valves have position indication and other control (SER 13.0.2.3, also see IEB 79-05, 79-06, and 79-08)". The TMI item required review of all valve positions, positioning requirements, positive controls, and related test and maintenance procedures to ensure proper ESF functioning. SER Section 13.5.2.3 required verification that procedures were in place that cover the TMI item requirements. The IE Bulletins were specific to the subject and were issued following the TMI incident.

Inspection Report 50-440/84015 closed IE Bulletins 79-05 and 79-06. Inspection Report 50-440/85010 closed IE Bulletin 79-08. When writing the following procedures the licensee incorporated the requirements of the item into the procedures:

Plant Administrative Procedure (PAP)-0205, "Operability of Plant Systems",
PAP-0905, "Work Order Process",
PAP-1105, "Surveillance Test Control",
PAP-1405, "Equipment Tagging", and Technical Administrative Procedure (TAP)-0503, "Preparation of Technical Specifications Surveillance Instructions".

The inspector reviewed the listed procedures and found that they incorporated the requirements of TMI Item II.K.1.5. This item is closed.

b. (Open) SER Open Item (440/85022-07 (DRP)): "TMI Items I.C.1 and I.C.9; Determine that the licensee has provided procedures for normal, transient, and accident conditions in accordance with SER 4.4.7.2, 6.3.3, and 13.5.2.2, and SSER3 13.5.2.2". As part of its normal inspection program, the IE Branch reviews operating procedures including those for normal, transient and accident conditions. These will be reviewed to determine if they are in accordance with the NRR Approved Procedure Generation Package as well as other considerations.

No violations or deviations were identified.

4. Licensing Review Follow-up Inspection Items (92701)

The Office of Nuclear Reactor Regulation, Division of Systems Integration, Instrumentation and Control Systems Branch (ICSB), has requested that Region III inspectors follow up on the applicant's activities in certain areas identified in a report transmitted on May 31, 1985, documenting a site review by ICSB personnel. Each of the items requiring inspection was assigned an open item tracking number. Additionally, the ICSB Trip Report reference section is noted in parentheses.

(Closed) Open Inspection Item (440/85033-04 (DRP)): "Verify that all nomenclature for the Low Pressure Core Spray and Residual Heat Removal System injection valve permissive instrumentation is correct (ICSB Trip Report, Section 3.f)". During a site review by ICSB personnel, they found a discrepancy regarding the nomenclature on the tag for the pressure permissive transmitter for the low pressure core spray injection valve. (An incorrect function was identified.) The ICSB personnel requested IE personnel to verify the accuracy of all nomenclature associated with the low pressure permissive interlocks for the LPCS and RHR injection valves.

The licensee wrote Nonconformance Report (NR)-OQC-2801 to document the error, determine if other errors existed, and provide a disposition for the error(s). A verification of RHR and LPCS pressure permissive nomenclature was performed by operational quality assurance and reported in Inspection Report R85-10112. One other case of incorrect nomenclature was found. Work Authorization (WA)-NTS-85-8085 was issued to remove and scrap the incorrect marker plates and install correct marker plates.

The inspector reviewed the NR and WA, and made an independent check of the nomenclature in question. No other discrepancies were found and the paperwork was in accordance with the licensee's quality assurance procedures. This item is closed.

No violations or deviations were identified.

5. Licensee Actions on 10 CFR 50.55(e) Items (92700)

a. (Closed) 10 CFR 50.55(e) Report (440/80004-EE, 441/80004-EE (DAR 027)): "Rosemount Model 510 DU Trip Calibration Units/1152 Pressure Transmitters". The General Electric Company (GE) notified the licensee of a 10 CFR 21 report concerning defects in certain Rosemount Model 510 DU trip/calibration units and Model 1152 pressure transmitters. Subsequently, the licensee submitted a 10 CFR 50.55(e) report on the same subject. GE wrote Field Disposition Instructions (FDIs) WNHW, WREC, and WNHX to document the condition and identify the deficient units. The licensee wrote Nonconformance Report (NR) OQC-1325 to document the condition and provide a disposition for the pressure transmitters and NR RECI-055 to do the same for the trip/calibration units. A total of 105 Model 510 DU trip/calibration units were removed from the plant, sent to the manufacturer for rebuilding and requalification, and reinstalled in the plant. A total of 46 Model 1152 pressure transmitters were removed from the plant and placed in the warehouse for future use as

non-safety-related spares. They were replaced in the plant with Model 1153 pressure transmitters that were fully qualified.

The inspector reviewed the FDIs and NRs along with their accompanying documentation and concludes that no potentially deficient Rosemount Model 510 DU trip/calibration units or Rosemount Model 1152 pressure transmitters are installed in safety-related systems in the Perry Plant. This item is closed for both units.

b. (Closed) 10 CFR 50.55(e) Report (440/84014-EE (DAR 172)): "HPCS Power Supply -- FSAR states that regulator switch S-26 in a manual position will prevent the diesel generator from starting. There is no indication of this logic on the drawings. Mispositioned regulator switch - not annunciated - do not concur with licensee position". In Inspection Report 50-440/85030, 50-441/85014, the inspector disagreed with the licensee's conclusion that mispositioning of the voltage regulator switch on the diesel generator need not be annunciated in the control room because the normal position of the switch is in the automatic position. The inspector pointed out that all other switches on the same panel that cause loss of the HPCS power supply when in the incorrect position are annunciated in the control room.

As a result, the licensee wrote Engineering Change Notice (ECN) 28351-86-2158 to delete the voltage regulator switch (S-26) from the Division 3 diesel generator and to hardwire controls for automatic operation. This resolves the problem because the switch no longer exists.

The inspector reviewed the ECN and accompanying documentation and determined that the switch had been disabled and no longer has any function on the unit 1 Division III power supply. The inspector concurs that the problem has been resolved. This item is closed for unit 1.

c. (Closed) 10 CFR 50.55(e) Report (440/85001-EE (DAR 220)): "Two locations were found where condensate can collect in the RCIC steam supply line. The condensate could disable the RCIC system". Following the decision to have the RCIC steam supply line inboard isolation valve (E51-F063) in a normally closed position rather than a normally open position during operation, the licensee discovered that condensate could collect upstream of valve E51-F063 in the RCIC steam supply line and in the lower portion of the steam line going to the RHR heat exchanger. This line is used during the steam condensing mode of operation.

The licensee analyzed the problem and issued Engineering Change Notice (ECN) 27859-86-1842 to provide drainage of condensate trapped upstream of valve E51-F063 in the RCIC steam supply line. To accommodate the condensate trapped in the steam line to the RHR heat exchanger, System Operating Instruction (SOI)-E12 was revised to require a slow controlled opening of the pertinent isolation and supply valves to prevent possible water hammer.

The inspector reviewed the ECN and accompanying documentation and determined that the drain piping changes had been made. The piping changes should prevent collection of condensate in the RCIC steam supply line. The inspector also reviewed the revised SOI and concludes that the changes should prevent the possibility of water hammer when placing the plant in the steam condensing mode. This item is closed for unit 1.

d. (Closed) 10 CFR 50.55(e) Item (440/85015-EE (DAR 241)): "Potential excess offsite releases due to a single isolation valve between post-accident sample system and primary containment (via RHR suppression pool return line)". The licensee discovered that containment isolation for the 1/2 in. Post Accident Sampling System (PASS) sample dump line did not meet the requirements of General Design Criterion 56 in that only one isolation valve was present. Criterion 56 requires two valves unless suitable justification is given. The licensee's evaluation of the problem indicated that a second valve was necessary.

The licensee wrote Engineering Design Deficiency Report (EDDR) 207 to document the deficiency, determine reportability, determine the cause of the deficiency and effect on other systems, and report the actions taken to resolve the deficiency and preclude recurrence. It was determined that the cause was an engineering oversight and other systems were not affected. ECN 28343-86-2151 was written to add a second containment isolation valve to resolve the deficiency. It was also determined that no further actions were necessary to preclude recurrence since it was an isolated case.

The inspector reviewed the EDDR and supporting documentation and the ECN and supporting documentation and determined that the second isolation valve was installed. As a result of the inspection reviews, the inspector concludes that the deficiency has been resolved. This item is closed.

No violations or deviations were identified.

6. Evaluation of Licensee Action with Regard to IE Bulletins (92703)

For the IE Bulletins listed below, the inspector verified that the Bulletin was received by licensee management and reviewed for its applicability to the facility. If the Bulletin was applicable the inspector verified that the written response was within the time period stated in the Bulletin, that the written response included the information required to be recorded, that the written response included adequate corrective action commitments based on information presented in the Bulletin and the licensee's response, that the licensee's management forwarded copies of the written response to the appropriate on-site management representatives, that information discussed in the licensee's written response was accurate, and that corrective action taken by the licensee was as described in the written response.

a. (Closed) IE Bulletin 79-12 (440/79012-BB, 441/79012-BB): "Short Period Scrams at BWR Facilities". Several instances of short period

scrams had been experienced at operating boiling water reactors under a wide variety of circumstances. However, they did have several things in common. In no case was an accurate estimate of the critical position made prior to the approach to critical. In each case a rod was being pulled in a high worth region and in each case the operator was pulling a rod on a continuous withdrawal basis. As a result, the Bulletin listed a number of actions to be taken by licensees of operating boiling water reactors. The Bulletin was also sent to BWR facilities with a construction permit for information.

When the licensee wrote Fuel and Core Analysis Instructions FTI-B02, "Control Rod", and FTI-B01, "Preparation of Control Rod Sequences", the guidance contained in the Bulletin was incorporated.

The inspector reviewed FTI-B02, FTI-B01, and Bulletin 79-12 and found that the guidance contained in the Bulletin was incorporated in the instructions. This item is closed.

b. (Closed) IE Bulletin 80-25 (440/80025-BB, 441/80025-BB): "Operating problems with Target Rock safety-relief valves at BWRs". A number of events occurred over a short period of time at an operating reactor involving two types of malfunction of Target Rock safety-relief valves. One type of malfunction involved the solenoid actuators. Failure of the pneumatic pressure regulator caused abnormally high pneumatic supply pressure that forced the solenoid actuators open and, therefore, the safety-relief valves. In another instance, the excessive use of "Loc-tite" (i.e., a trademark adhesive for nuts and bolts) caused the solenoid plunger to adhere to the bonnet thus preventing pneumatic pressure from entering the actuator. The other malfunction involved the valve itself. Investigation following the failure of a valve to open on demand did not reveal the cause. However, the investigation was limited to the top-works of the valve and did not include the main stage internals. Following another malfunction, the cause was found in the main stage internals. The Bulletin required certain actions to be taken by utilities with operating reactors and provided the information to utilities with construction permits.

The licensee responded to the information as follows:

- The design of the pneumatic supply system contains relief valves to prevent overpressurization. The valves are located in the proximity of the safety-relief valves and underpressure and overpressure of the pneumatic supply system are annunicated in the control room.

- The Generic Mechanical Instruction (GMI)-0015, "Repair of Safety Relief Valves" contains cautions on the correct use of "Loc-tite". It also requires complete disassembly of a relief valve following a malfunction unless the cause is clearly evident.

The inspector determined that the pneumatic supply system contains relief valves in close proximity to the safety-relief valves and that underpressure and overpressure conditions are annunciated in the control room. The inspector reviewed GMI-0015 and found that it contains adequate cautions on the use of "Loc-tite" and that complete disassembly and inspection is required following a malfunction unless the cause is clear. This item is closed.

No violations or deviations were identified.

7. Evaluation of Licensee Action with Regard to IE Circulars (92717)

For the IE Circular listed below, the inspector verified that the Circular was received by the licensee management, that a review for applicability was performed, and if the Circular was applicable to the facility, appropriate corrective actions were taken or scheduled to be taken.

(Closed) IE Circular 81-12 (440/81012-CC, 441/81012-CC): "Inadequate periodic test procedure of PWR protection system". While investigating another problem, the operator of a Pressurized Water Reactor (PWR) found that the reactor Trip Circuit Breakers (TCB) did not trip immediately (because of binding) on undervoltage. The shunt trip function of the TCBs was functioning normally. Investigation showed that the cause was an out-of-adjustment condition in the linkage mechanism of the undervoltage trip device. Also, maintenance of the undervoltage trip device was not normally scheduled. The investigation further revealed that the periodic test procedure for the reactor protection system did not independently test the undervoltage trip function.

The licensee determined that this Circular was not applicable to the Perry Plant. The type of breaker in question is not used at the Perry Plant in any safety-related system. The reactor protection system uses relays and other breakers do not have dual function (i.e., shunt and undervoltage) mechanisms.

The inspector determined that dual function mechanism breakers are not used at Perry in safety systems. This item is closed.

No violations or deviations were identified.

8. Licensee Actions on Allegations (99014)

(Closed) Allegation (RIII-85-A-0138): "Allegation concerning diesel generator control circuit design deficiencies". While doing preoperational testing of the Division 3 diesel generator, a control circuit anomaly was found that was common to all three divisions. The following sequence of events is necessary for the anomaly to appear:

- Open the preferred source breaker by throwing the switch in the control room (a green "flag" shows).

- Transfer the diesel-generator control to the diesel room by throwing the local-remote switch.

- Perform the required diesel-generator test, maintenance, repair, etc.

- Close the preferred source breaker by using the switch in the diesel room (a red "flag" shows).

- Return diesel-generator control to the control room using the local-remote switch.

If this sequence is followed and an undervoltage condition is experienced by the 1E bus, the diesel would start, but the diesel-generator bus breaker would not close. The anomaly is caused by the fact that one preferred source breaker switch has a green "flag" while the other preferred source breaker switch has a red "flag".

When this condition was experienced, Nonconformance Report (NR)-OQC-2273 was written to document the condition and provide disposition. The disposition initially proposed was to change the System Operating Instructions (SOIs) for the affected systems to include steps to align both preferred source breaker control switches to agree with the indicated position (both "flags" to show the same color). The person who discovered the anomaly did not agree with the disposition for providing administrative controls to solve the problem. When he was not successful in changing the disposition to provide a circuit change to solve the problem, he informed the Senior Resident Inspector of his concerns.

In subsequent meetings with the licensee, the inspector pointed out that the guidance contained in Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems", may not have been implemented if the system were left unchanged. As a result, the licensee issued Engineering Change Notice (ECN) 28808-86-2465 to change the control circuit wiring to solve the problem. The inspector reviewed the ECN and associated documents and found that the anomaly no longer exists in the Division I, II or III diesel-generator control circuits. Since the wiring changes have been made only in the unit 1 control circuitry, the allegation should be considered closed for unit 1 only.

No violations or deviations were identified.

9. Procedure Review (42450)

The inspector reviewed the following procedures to verify that: (1) each procedure was technically adequate for the task to be performed, (2) each procedure was written in such a manner as to be easily understood by the user, and (3) each procedure was consistent in content and format with all applicable regulatory requirements.

Technical Specification Surveillance Instructions

SVI-E31-T0079A Main Steam Line Tunnel Ambient Temperature High Channel A Calibration for 1E31-N031A

SVI-E31-T70078A Main Steam Line Tunnel Ambient Temperature High Channel A Functional for 1E31-N604A

SVI-E31-T0074A Main Steam Line Isolation Flow-High Channel A Functional for 1E31-N686A

SVI-D17-T8036 Emergency Service Water Loop A Radiation Monitor Functional for 1D17-K604

SVI-E51-T0339 Reactor Core Isolation Cooling System Flow (Rem Shtdn Mon) Channel Calibration for 1E51-N003

SVI-E51-T1292A RCIC Actuation-Condensate Storage Tank Level-Low Channel A Functional for 1E51-N635A

The review showed that the above procedures were technically adequate and easily understood. However, there were inconsistencies in the format. Plant Administrative Procedure (PAP)-1105, "Surveillance Test Control", establishes the policy and administrative controls governing surveillance testing at Perry. As part of the procedure, it lists the quality records that are generated as a result of performing surveillances. Technical Administrative Procedure (TAP) 0503, "Preparation of Technical Specification Surveillance Instructions", contains detailed instructions for preparing the instructions and sets the format for the instructions. A majority of the SVIs were written following Rev. 3 of TAP-0503. This revision did not indicate that quality records were generated as part of performing the instruction. Revision 4 of TAP-0503 corrected this shortcoming and changed the format to indicate what quality records are generated. The SVIs written following Rev. 4 of TAP-0503 therefore indicate what quality records are generated. This inconsistency is regarded as an open item (440/85066-01) pending resolution by the licensee and further review by an inspector.

No violations or deviations were identified.

10. Open Items

Open items are matters which have been discussed with the licensee, which will be reviewed further by the inspector, and which will involve some action on the part of the NRC or licensee or both. An open item disclosed during the inspection is discussed in paragraph 9.

11. Exit Interview

The inspector met with licensee representatives (denoted in paragraph 1) at the conclusion of the inspection on September 20, 1985. The inspector summarized the scope and findings of the inspection. The licensee acknowledged the inspector's findings. The licensee did not indicate that any of the information disclosed during the inspection could be considered proprietary in nature.