U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-440/85081(DRS)

Docket No. 50-440 License No. CPPR-148

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, OH

Inspection Conducted: November 16, 1985 through January 9, 1986

Inspectors: MA Ring for G. F. O'Dwyer 1/27/86 Date

D. E. Hills
D. E. Hills 1/24/86 Date

Approved By: MA Ring M. A. Ring, Chief 1/27/86 Date
Test Programs Section

Inspection Summary

Inspection on November 16, 1985 through January 9, 1986 (Report No. 50-440/85081(DRS))

Areas Inspected: Routine unannounced inspection of previous inspection findings, preoperational test results verifications and preoperational test results reviews. The inspection involved a total of 136 inspector-hours onsite by two inspectors including 24 inspector-hours during off-shifts. In addition, there were 227 inspector-hours spent offsite. Inspection modules consisted of 70322B, 70325B, 70326B, 70536B, 70542B, 70544B, 70559B, 70329B, 92701B and 92702B.

Results: Of the three areas inspected, no violations or deviations were identified in two areas. Within the remaining area, one violation was identified (failure to adequately evaluate test results - Paragraphs 4.b and 4.c).

DETAILS

1. Persons Contacted

   *M. D. Lyster, Manager, Perry Plant Operations Department
   *C. M. Shuster, Manager, Nuclear Quality Assurance Department
   *B. D. Walrath, General Supervising Engineer, Operations Quality Assurance
   *B. S. Ferell, Licensing Engineer, Nuclear Engineering Department
   *G. R. Leidich, General Supervising Engineer, Nuclear Test Section
   *G. H. Gerber, Element Supervisor Administration, Nuclear Test Section
   *B. B. Liddell, Operations Engineer, Perry Plant Technical Department
   *J. M. Warren, Mechanical Engineer, Nuclear Construction Engineering Section
   *D. D. Jones, Lead G.E. Site System Engineer, Site Engineering Response Team
   *J. Eppich, Senior Engineer, Nuclear Construction Engineering Section
   *T. Heatherly, Operations Engineer, Perry Plant Technical Department

   *Denotes persons attending the exit meeting of January 9, 1986.

   The inspector also interviewed other licensee employees including members of the quality assurance, technical, operating and testing staff.

2. Licensee Action on Previous Inspection Items

   a. (Closed) Unresolved Item (440/85002-02(DRS)): Consideration of instrument inaccuracy in determination of preoperational test procedure acceptance criteria. The inspector has reviewed licensee activities as described in Inspection Report 440/85063(DRS) which were determined to be adequate. The inspector has no further concerns in this area.

   b. (Open) Unresolved Item (440/85069-01(DRS)): Licensee to revise PAP-1104 "Startup Test Program" to require additional review of Startup Test Instructions (STI) by Operations Quality Section. This line-by-line review is to ensure that the procedures are correct as written. The inspector has reviewed Revision 2 of this administrative procedure and verified that the indicated change has been incorporated. The inspector intends to review selected STIs after new revisions have been issued per the new requirements. This is to determine the effectiveness of this change to preclude problems such as those identified in Inspection Report 440/85069(DRS).

   c. (Closed) Open Item (440/85063-02(DRS)): Licensee to revise STI J11-0003 "Fuel Loading" to incorporate various changes as delineated in Inspection Report 440/85063. The inspector has reviewed Revision 1 to this procedure and verified that the indicated changes have been incorporated. The inspector has no further concerns in this area.

d. (Closed) Unresolved Item (440/85042-03(DRS)): Preoperational test procedure acceptance criteria tolerances not consistent with the proposed plant specific setpoint methodology. The licensee has committed to provide to the Office of Nuclear Reactor Regulation (NRR), prior to startup following the first refueling outage, a detailed technical assessment of the methods used to establish the Perry protection system setpoints and allowable values based on the generic findings of the Instrument Setpoint Methodology Group. The licensee therefore intends to delay the establishment and submittal of technical data supporting the plant specific "Leave-As-Is" zone and will now perform surveillance requirements as depicted in the Perry Technical Specifications. Because setpoint methodology is being handled as a generic issue by NRR, the inspector has no further concerns in this area.

e. (Closed) Violation (440/85053-01(DRS)): Preoperational test procedure TP 1P57-P001 "Safety-Related Instrument Air" was inappropriate in that it did not adequately control the sequence of testing. This test had not yet been conducted when this problem was identified. The inspector verified that the procedure has now been changed to ensure adequate prerequisites are established for each test section independent of the sequence in which the test is performed. Since this test had been reviewed and approved by the Management Procedure Review Team (MPRT), the other test procedures reviewed by the same reviewer were re-evaluated. No discrepancies were identified in this re-evaluation. Due to the results of this re-evaluation and to other post-MPRT procedures that have been reviewed by the NRC with no identified discrepancies, the inspector considers this an isolated case. As to the generic issue concerning the adequacy of preoperational test procedures, the inspector has evaluated the activities that resulted from Special Project Plan 1102 "Test Procedure Assurance Review." Based on the extensive corrective actions involved in these activities and more recent NRC conducted procedure reviews, the inspector considers these actions to have been effective in assuring the adequacy of preoperational test procedures. Therefore, the inspector has no further concern in this area.

f. (Open) Open Item (440/85053-07(DRS)): Review administrative controls for preoperational testing conducted after fuel load. The inspector has reviewed the proposed program as depicted in draft 5 of Revision 0 of PAP-0113 "Nuclear Test Section Organization and Responsibilities." This procedure essentially endorses the Test Program Manual (TPM). Therefore, the inspector also reviewed proposed changes to the TPM which are to be instituted to ensure conformance with operating requirements. During the review of these documents the inspector noted that the licensee intends to perform reviews pursuant to 10 CFR 50.59 for preoperational test procedures during the release for test process. However, the licensee did not intend to perform these reviews for changes to preoperational test procedures after release for test. After discussions with the inspector, the licensee has decided to incorporate requirements into the program for 10 CFR 50.59 reviews for major changes to preoperational test

procedures. This would be consistent with the similar requirements pertaining to procedures contained in the Perry Nuclear Power Plant (PNPP) Operations Manual. The inspector intends to review this program area in more detail once the implementing procedures are actually approved by the licensee.

3. Preoperational Test Results Verification

The inspector verified that the following preoperational test results were documented, reviewed, and approved by the licensee in accordance with the requirements of Regulatory Guide 1.68, the Test Programs Manual (TPM), the Final Safety Analysis Report (FSAR), the Safety Evaluation Report (SER), and the Quality Assurance (QA) Program and found them satisfactory.

TP 1B13-P001, "Reactor Vessel Flow Induced Vibration Test without Fuel," Revision 1
TP 1B21A-P001, "Nuclear Boiler Process Instrumentation," Revision 1
TP 1B33-P001, "Reactor Recirculation and Control System," Revision 1
TP 1C11B-P001, "Control Rod Drive Hydraulic," Revision 0
TP 1C51A-P001, "Neutron Monitoring System: Startup Range Monitors," Revision 1
TP 1C51A-P002, "Neutron Monitoring System: Intermediate Range Monitors," Revision 1
TP 1E31-P001, "Leak Detection System Test," Revision 1
TP 1E67-P001, "Control Room Leakage Test," Revision 2
TP 0F14-P001, "In-Vessel Servicing Equipment," Revision 1
TP 1F15-P001, "Refueling Equipment," Revision 1
TP 1G33-P001, "Reactor Water Cleanup," Revision 0
TP 1G36-P001, "RWCU Filter/Demineralizer Test," Revision 1
TP 1M13-P001, "Drywell Cooling Test," Revision 2
TP 1P53-P001, "Penetration Pressurization," Revision 1
TP 0P72-P002, "Plant Foundation Underdrain," Revision 1
TP 1R71-P001, "Essential and Emergency Lighting," Revision 1
TP 1C22-P001, "RRCS," Revision 1
TP 1C85-A001, "Steam Bypass and Pressure Regulation," Revision 0
TP 1M14-A001, "Containment Vessel and Drywell Purge," Revision 2
TP 1N27-A001, "Feedwater System," Revision 0
TP 0P43-A001, "Nuclear Closed Cooling," Revision 3
TP 1P52-A002, "Loss of Instrument Air," Revision 1
TP 1R10-A001, "Normal AC Power," Revision 0
SP 1E68-002, "System Thermal Expansion Test," Revision 0

No violations or deviations were identified.

4. Preoperational Test Results Reviews

The inspector reviewed the results of the following tests against the FSAR, the SER, Regulatory Guide 1.68, the QA Manual, and the Test Program Manual, and determined that test changes and test exceptions were processed in accordance with administrative controls, test deficiencies

were identified, processed, and corrected as required, results were evaluated and met the acceptance criteria, and the results were reviewed and approved as required except as noted below:

a. TP 1R43-P001, "Division 1 Standby Diesel Generator," Revision 1.
TP 1R43-P002, "Division 2 Standby Diesel Generator," Revision 1.

Regulatory position C.2.a(9) of Regulatory Guide 1.108, "Periodic Testing of Diesel Generator Units As Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, requires testing to demonstrate the required reliability by means of any 69 consecutive valid tests (per plant) with no failures, with a minimum of 23 or 69/n tests, whichever is larger, per diesel generator unit (where n is equal to the number of diesel generator units of the same design and size.) In the performance of this testing requirement, the wait period between consecutive starts was in most cases only a few minutes. This is insufficient time for the diesel generators to cooldown to minimum standby temperature conditions. Regulatory Guide 1.108 also prescribes that testing of the diesel generator should simulate, where practical, environments (temperature, humidity, etc.) that would be expected if actual demand where to be placed on the system. Therefore, it is not clear whether the intent of this regulatory guide has been met in that testing was not conducted from the most conservative start conditions that might be expected (e.g., minimum standby temperature conditions.) This is considered an unresolved item pending clarification of regulatory requirements, evaluation of measured data to determine the effect of actual start conditions on start ability, and determination of any deteriorating effects on the diesel generators as might be caused by repetition of the consecutive starts (440/85081-01(DRS)).

During this review the inspector questioned the adequacy of calculations used to determine the minimum fuel oil storage tank capacity required by Technical Specification 3.8.1.1. Acceptable methods to perform total required storage capacity calculations were depicted in American National Standards Institute (ANSI) N195-1976, "Fuel Oil Systems For Standby Diesel Generator," as endorsed by Regulatory Guide 1.137, "Fuel Oil Systems For Standby Diesel Generators," Revision 1. The licensee has committed to this regulatory guide in the Perry FSAR. Storage capacity is required to be sufficient to operate the diesel generators following the limiting design basis accident for seven days. One method of determining storage tank capacity depicted in the ANSI standard is to base the calculations upon the diesel generators operating at the minimum required capacity for the plant condition which is most limiting for the calculation of such capacity. This method takes into account the time dependence of diesel generator loads over the seven day storage requirement. If this method is used a minimum margin of 10% is to be added to the calculated storage requirement for conservatism. A conservative alternate method is also allowed which calculates the storage capacity by assuming that the diesel operates continuously for seven days at its rated capacity. In the latter method a 10%

margin for conservatism is not required. Both methods are required to include an explicit allowance for fuel consumption required by periodic testing. However, as stated previously, this ANSI standard appears to apply only to determination of total storage tank capacity and not to the technical specification minimum capacity. Therefore, it is not clear whether assumptions and conservatisms prescribed by the ANSI standard are also required to be considered when calculating the technical specification value. The method chosen for the technical specification calculations performed in Design Verification Record (DVR) Assignment 1928 actually does not correspond to either method described in the ANSI standard (e.g., calculations based on the varying seven day accident load or upon rated capacity for the seven days). Instead, the calculations are based upon assuming the maximum accident load for the entire seven days and includes a 10% margin added for conservatism. The inspector considers this an acceptable method since it is in effect even more conservative than the first method described in the ANSI standard. This method resulted in a minimum storage tank capacity of 69,430 gallons which has been incorporated into technical specifications. (Total storage tank capacity is 90,000 gallons.) However, the licensee used fuel consumption rate data from the vendor's established curves in the calculations, although actual consumption rates determined from preoperational testing were available at that time. Actual onsite test data shows that the diesel fuel oil consumption rate is approximately 5% higher than that depicted on the vendor's curve. Personnel responsible for the technical specification calculations had previously been made aware of this variation through Field Question 46721. If the actual consumption rate determined from test data is substituted for the vendor's consumption rate used in DVR Assignment 1928, the minimum storage tank capacity can be calculated to be 72,570 gallons. Therefore, the technical specification value of 69,430 gallons appears to be too low to meet the seven day requirements. Upon identification of this problem by the inspector, the licensee performed additional calculations documented in Field Change Request (FCR) 1197 to ensure adequacy of the technical specification value. The inspector has reviewed these new calculations and has also questioned their adequacy. These calculations used actual preoperational test data to determine the fuel oil consumption rate at the maximum design accident load. This rate was then used to determine the storage capacity to meet the seven day requirement. The value obtained was 66,409 gallons which is well within the technical specification value. Thus, the FCR determined the technical specification value to be adequate. However, the inspector has noted that these calculations do not include the 10% margin for conservatism as was included in the original calculational method of DVR Assignment 1928. Although using a maximum design accident load is a more conservative approach than using the varying accident loads, it is not clear whether this accounts for the total 10% margin for conservatism required by the ANSI standard in determining total storage capacity. The licensee has not performed any calculations to ensure this method provides

this margin. This is considered an unresolved item until sufficient assurances are provided that the technical specification minimum capacity is sufficient to correspond to the ANSI standard conservatisms or it has been determined that these conservatisms are not required. (440/85081-02(DRS)).

b. TP 1G43-P001, "Suppression Pool Makeup (SPMU) System," Revision 2.

During the review, the inspector noted that upper pool to suppression pool dump time failed to meet the requirements of acceptance criteria 7.1 as documented in test exception E-01. This acceptance criteria, consistent with design criteria depicted in FSAR Section 6.2.7, requires that when dumping through either the "A" or "B" SPMU piping, that the level decrease in the upper pool equivalent to 32,830 cubic feet be achievable within 8.67 minutes. Test data showed that the actual drop in upper pool level within this time period corresponded to approximately 32,367 cubic feet (98.6%). Justification for the test exception resolution was documented through Field Question (FQ) 40640. Upon review of the FQ, the inspector has determined that the documented response was inadequate to provide resolution. The FQ indicated that the dominant design parameter was the ability to maintain a minimum of two feet coverage above the top of the top vent in the suppression pool in order to ensure adequate pressure suppression capabilities. Suppression pool levels measured during the test indicated levels of 6.6 inches and one inch below this criteria (two feet coverage) for the "A" and "B" dumps, respectively. The FQ further indicated that strip chart recordings of suppression pool level obtained during the test showed that the suppression pool test data was erroneous. Based upon this justification, it was decided to disregard suppression pool level data for dump "A" which corresponds to the worse of the two cases. Data for dump "B" however was retained and, in fact, the licensee changed the FSAR by Amendment 22 to indicate that a minimum vent submergence of one foot eleven inches was sufficient based upon General Electric test facility data. The inspector has reviewed the indicated strip chart recordings and has determined them to be inadequate to provide justification for disregarding the data. The strip chart resolution is not sufficient to draw any definitive conclusions and in fact appears to substantiate rather than contradict the actual test data. In addition, scales for level and the time base were not designated on the strip charts, thereby, making it impossible to correlate the starting and stopping elevations on the strip chart with those recorded in the test procedure and chronological test log. Possible scales suggested by the licensee also did not provide correlation. Finally, no rationale was given in the FQ for treating the suppression pool data as erroneous for the worse case while data for the better case was treated as reliable. In order to deal with the case where suppression pool data was disregarded, additional calculations were performed in the FQ. These calculations were also performed for the better case, although it was already indicated as resolved in the FQ. These calculations used actual test data for upper pool level to determine the corresponding change in suppression pool level, assuming maximum

design runout flow for the Emergency Core Cooling System (ECCS) pumps. The results showed that level slightly surpassed the minimum two feet submergence for both the "A" and "B" dumps. However, upon review of these calculations, the inspector noted that they do not account for nonconservative error interjected by this method. Specifically, values used for upper pool and suppression pool dimensions in these calculations were the nominal design values and therefore did not account for the allowed construction tolerances. These considerations are regarded as a violation of 10 CFR 50 Criterion XI in conjunction with the example given in Paragraph 4.c in that test results as pertaining to test exceptions were inadequately evaluated and dispositioned (440/85081-03a(DRS)). Of particular concern is verification of the SPMU system to meet design specifications. As a result of the inspector concerns, the licensee has indicated that additional calculations will be performed and provided for the inspector's review. The calculations should incorporate either construction tolerances with the nominal design values or replace these values with the as-built dimensions. In addition, the licensee has indicated that they may have to rely upon actual ECCS pump runout flows from testing instead of the higher design flows to ensure the one foot eleven inches vent submergence as currently specified in the FSAR. Preliminary calculations indicate that the ECCS pump flows are a more dominant factor in the calculations than construction tolerances. The inspector will review these calculations once they are available to ensure adequate system capabilities. In addition, changes to the FSAR may have to be initiated to incorporate modifications of design requirements and assumptions from those currently contained in the FSAR.

c. TP 1C71-P-002, "Reactor Protection System (RPS) Motor Generator (MG) Sets," Revision 0.

During this review the inspector noted that acceptance criteria indicated in step 6.1.2.13 required the RPS MG set underfrequency trip to be greater than or equal to 54 Hertz. The underfrequency trip occurred once at 53.8 Hertz and twice at 54.0 Hertz in three successive test performances. Test exception E-03 and Field Question (FQ) 45000 were generated to provide resolution. The FQ and the test exception determined the accuracy of the measuring and test equipment (M&TE) as ±0.57 Hertz. The FQ and the test exception both recommended that 53.8 Hertz be accepted because "the observed error (-.2 Hz) is within the tolerance of the M&TE used to make the measurement." The inspector considers this to be inadequate justification to resolve the test exception. The measured value of the frequency setpoint was 53.8 Hertz and the instrument accuracy was ±0.57 Hertz and therefore the actual value of the frequency could have been anywhere between 53.23 Hertz and 54.37 Hertz.

The justification did not consider that based upon M&TE accuracy the majority of the possible values were below the acceptance criteria. In addition, the licensee did not review post-test calibration data of the M&TE in order to substantiate the resolution. This is

considered a violation of 10 CFR 50 Criterion XI in conjunction with the example given in Paragraph 4.b in that test results as pertaining to test exceptions were inadequately evaluated and dispositioned (440/85081-03b(DRS)). As a result of the inspector's concerns, the licensee generated Field Change Request 1166 which the inspector has reviewed. The inspector considers this sufficient justification to resolve the test exception.

d. TP 1E12-P001, "Residual Heat Removal System," Revision 1

e. TP 1B21B-P001, "Nuclear Boiler: Automatic Depressurization System," Revision 1

f. TP 1E53-P001, "Containment Isolation System," Revision 2

g. TP 1N27B-P001, "Feedwater Leakage Control System," Revision 1

h. TP 1M51-P001, "Combustible Gas Control System," Revision 1

i. TP 1C71-P001, "Reactor Protection System," Revision 2

j. TP 1R76-P001, "ECCS Initiation/Loss of Offsite Power," Revision 1

k. TP 1B33-P002, "Reactor Recirculation System Flow Control," Revision 1

l. TP 1B21C-P001, "Nuclear Steam Supply Shutoff System," Revision 0

One violation and two unresolved items were identified. No other violations or deviations were identified in this program area.

5. Unresolved Items

Unresolved items are matters about which information is required in order to ascertain whether they are acceptable items, violations, or deviations. The unresolved items disclosed during the inspection are discussed in Paragraph 4.a.

6. Exit Interview

The inspector met with licensee representatives denoted in Paragraph 1 on January 9, 1986. The inspector summarized the scope and findings of the inspection and discussed the likely content of this inspection report. The licensee did not indicate that any of the information disclosed during the inspection could be considered proprietary in nature. The licensee acknowledged the statements by the inspector with respect to the violation in Paragraphs 4.b and 4.c.