U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-341/84-36(DRS)

Docket No. 50-341 License No: CPPR-87

Licensee: Detroit Edison Company
2000 Second Avenue
Detroit, MI 48224

Facility Name: Enrico Fermi Nuclear Power Plant, Unit 2

Inspection At: Enrico Fermi 2 Site, Monroe, Michigan

Inspection Conducted: August 20 - September 28, 1984

Inspectors: S. G. DuPont 10/19/84
Date

D. E. Hills 10/19/84
Date

G. O'Dwyer 10/19/84
Date

T. E. Lang 10/22/84
Date

P. L. Eng 10/22/84
Date

Approved By: L. A. Reyes, Chief 10/22/84
Test Programs Section Date

Inspection Summary

Inspection on August 20 - September 28 1984 (Report No. 50-341/84-36(DRS)
Areas Inspected: Routine, unannounced inspection by regional inspectors for followup on licensee actions in regard to inspector previous findings, preoperational test result verification and review, preoperational test witnessing, operational and maintenance program implementation, meeting with licensee in regard to previous identified inspector concern with valve stroke timing, and review of startup test phase procedures. The inspection involved a total of

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277 inspector-hours onsite by three NRC inspectors, including 113 inspector-hours onsite during off-shifts. In addition, the inspection involved 131 inspector-hours in the Regional Office.

Results: Of the seven areas inspected, no items of noncompliance or deviations were identified in five areas. Within the remaining areas, four items of noncompliance were identified (inadequate engineering evaluation - Paragraph 3.a; failure to follow procedures - Paragraph 3.b; failure to maintain required log - Paragraph 6.b; failure to follow surveillance procedure - Paragraph 6.c).

DETAILS

1. Persons Contacted

+*R. S. Lenart, Superintendent, Nuclear Production
+*G. R. Overbeck, Assistant Superintendent, Startup
*E. P. Griffing, Assistant Manager, Nuclear Operations
+*T. S. Nickelson, Startup Engineer, Startup
+*E. Preston, Operations Engineer, Nuclear Production
+*H. O. Arora, Reactor Engineer, Nuclear Production
+*J. D. Leman, Maintenance Engineer, Nuclear Production
+*M. W. Shields, Lead Startup Test Phase Engineer
+*R. R. Eberhardt, Rad Chemical Engineer, Nuclear Production
+*J. H. Plona, Technical Engineer, Nuclear Production
*C. R. Gellently, Startup Engineering Supervisor, Field Engineering
*W. E. Miller, Operational Assurance Supervisor, Nuclear Engineering
*L. P. Bregni, Engineer, Licensing
*S. E. Conen, Engineer, Licensing
+M. Ripley, Startup Director, Startup
+F. Reimann, Assistant Startup Director, Startup
+M. E. Haver, Engineer, Startup Assurance
+J. Nyquist, Assistant Superintendent, Nuclear Production
+S. J. Latone, Assistant Superintendent
+L. G. Lessor, Advisor, Nuclear Production
+J. J. Wald, Quality Engineer, Nuclear Quality Assurance

The inspectors also interviewed others of the licensee's startup, nuclear production and technical staff.

±Denotes personnel attending the exit interview of September 25, 1984.

*Denotes personnel attending the exit interview of September 28, 1984.

2. Licensee Action on Previous Inspection Findings

(Closed) Open Item (341/84-04-06 (DRS)): Radwaste Building HVAC preoperational test procedure verification identified an FSAR discrepancy in that FSAR Section 9.4.3.5 stated that the Radwaste Building Ventilation Fan trips on a differential pressure of two inches of water across the building which is not per design. The inspector has reviewed FSAR Amendment 38-July 1984 with respect to this item and found that the subject revision corrected this discrepancy.

(Closed) Open Item (341/84-28-01 (DRS)): Lack of a requirement in the master startup instruction to review the startup procedures against the approved technical specifications prior to their use. The inspector reviewed Revision 8 to STUT.000.100, "Master Startup Test Phase Procedure," and verified that the licensee had included adequate instructions to require startup test procedures to be reviewed against the approved technical specifications prior to their use. The inspector has no further concerns in this area.

(Closed) Open Item (341/84-28-02(DRS)): Lack of provisions in the startup program for performing 10 CFR 50.59 reviews (unreviewed safety questions). The inspector reviewed Revision 8 to STUT.000.100 and verified that the licensee had included adequate instructions to require 10 CFR 50.59 reviews when needed during the startup program. The inspector has no further concerns in this area.

(Closed) Open Item (341/84-01-03 (DRS): Test the standby Gas Treatment Exhaust Fan flow in accordance with FSAR Section 6.2.3.2. The inspector witnessed the performance of preoperational test PRET T9200.001 Secondary Containment System and verified that the requirement of Section 6.2.3.2, verifying exhaust fan flow of 4000 cubic feet per minute with half filter flow was accomplished. The inspector has no further concerns in this area.

(Closed) Open Item (341/84-01-04(DRS)): Preoperational test PRET T4100.001 did not contain all of the provisions of the FSAR Section 9.4.2 in that the operation of various fan-coil units were not verified as a quantitative test during Design Basis Accident (DBA) conditions. The inspector witnessed the performance of the Essential Safety Features (ESF) preoperational test and verified that the fan-coil units were tested as a quantitative test during DBA conditions. The inspector has no further concerns in this area.

(Closed) Unresolved Item (341/84-01-05 (DRS)): Resolve FSAR requirements for testing of the third Main Steam Isolation Valves (MSIV). The inspector reviewed the approved FSAR change which described the third MSIV as a manual block valve with no safety-related functions and removed the requirements to be tested during safety-related preoperational testing. The inspector has no further concerns in this area.

(Closed) Unresolved Item (341/84-11-04(DRS)): Various cells of the 2A-1 safety-related battery (130/260VDC) appeared weak during preoperational testing. The inspector witnessed the performance of the duty load cycle test on the 2A-1 battery and verified that the cells performed as required. This item is closed, however, the inspector has other concerns with the safety-related batteries as documented in paragraph 3.d of this report.

No items of noncompliance or deviations were identified.

3. Preoperational Test Results Review

The inspector reviewed the results of the following test procedures against the prescribed acceptance criteria and reviewed the licensee's test evaluation for adequacy and found them satisfactory except as noted below:

C1100.001 Control Rod Drive (CRD) Manual Control System
C1108.001 Rod Worth Minimizer
C1150.001 CRD Hydraulic System
C3202.001 Feedwater Control System
E2100.001 Core Spray System
R3201.001 130/260 VDC System

a. During the review of PRET C1150.001, the inspector identified a problem with the adequacy of an engineering evaluation. In conjunction with the 10 minute pressure drop test with no CRD pumps running, the accumulator for HCU 38-15 failed to hold pressure above the alarm setpoint which is indicative of an accumulator charging header check valve leak. Prior to this test as a result of their evaluation, Project Engineering instructed Startup not to correct any accumulator check valve leakage problems due to a current technical specification issue. Specifically, BWR standard technical specifications require the accumulators to hold pressure for a specific length of time following CRD pump trip as part of operability. Fermi's proposed technical specifications only require measuring and recording the time up to 10 minutes while the pressure remains above the alarm setpoint. This is considered an item of noncompliance (341/84-36-01(DRS)) in that as a result of these instructions from Project Engineering, the accumulator check valve failure was not properly identified in the test result package or during the review process, the problem was not corrected, and maintenance was not scheduled to correct the problem even after turnover to Nuclear Production as required by 10 CFR 50, Appendix B, Criterion XVI.

b. In addition, during the preoperational test results review the inspector identified a problem with adherence to administrative procedures specifically in the use of Test Exception Disposition Reports (TEDR) and Test Change Notices (TCN) as delineated in the following examples:

   (1) In PRET C1150.001, TEDR #10 did not receive the Nuclear Shift Supervisor's signature indicating his review and concurrence when initiated contrary to Startup Instruction 8.4.2.04 steps 4.3 and 5.1.1.

   (2) In PRET C1150.001, TEDR #18 did not indicate where subsequent retesting was accomplished and TEDR #38 did not indicate where retesting for HCUs 26-43 and 26-51 were accomplished. Thus, these TEDRs did not provide the cross reference bridges as required by Startup Instruction 8.4.2.05 step 4.2.3.

   (3) In PRET C3202.001, an incorrect negative reading at the process computer was received (step 6.9.2.9) which required terminations to be swapped at the computer to give the correct positive reading. However, this correction was not made through the use of a written TEDR which is contrary to Startup Instruction 8.4.3.04.

   (4) In PRET C3202.001, TEDRs #3 and #5 were not initiated in a timely manner as prescribed by Startup Instruction 8.4.2.04 step 4.8. Specifically more than a week had passed between identification and actual initiation of the respective TEDR. Furthermore, TEDR #3 was not written until after Supplemental Test #1 had been conducted as part of its resolution.

(5) In PRET C1100.001, TEDR #1, and in PRET C1150.001, TEDR #1, were not properly dispositioned. After initiation, these TEDRs were determined to be no longer applicable and were just marked "VOID" instead of being dispositioned per Startup Instruction 8.4.2.04.

(6) In PRET E2100.001, dispositioning of TEDR #20 was inadequate. This TEDR involved failure to meet the minimum bypass flow for loop "B" as prescribed in acceptance criteria 8.7. This was accepted "as is" by Startup Engineering Assistance per Startup Field Request (SFR) 2558. However, the subject flow value was prescribed by General Electric (GE) and therefore should have been evaluated by G.E. in order to deviate from it. However, G.E.'s evaluation and concurrence with this value was not properly documented. Following the inspector's indication of this discrepancy, the licensee has since received documentation indicating G.E. concurrence.

(7) In PRET C3202.001, dispositioning of TEDR #15 was also inadequate. This TEDR involved Startup Control Valve N21-F403 failing to meet response requirements in regard to positioner delay time, rise time, and average stroking rate. The description of the resolution to Startup Field Report SFR 1564 used to justify acceptance "as is" was incomplete in that it addressed only the problem with rise time and thus did not address the entire problem. Following inspector's identification of the discrepancy, the licensee has since developed written justification of the acceptance of delay time and average stroking rate in addition to rise time.

(8) In E2100.001, the testing method was changed by adding notes to the test steps without the use of an approved TCN as required by Startup Instruction 4.5.1.01. Changes to test steps 6.9.2.7b and 6.9.2.7c allowed actual breaker trips to be utilized instead of the simulation of lifted leads as prescribed in the test step and the change to test step 6.9.2.7f allowed the K12 relays to be manually actuated instead of using the test switch.

This is considered an item of noncompliance (341/84-36-02(DRS)) in that these numerous examples indicate a persistent failure to properly follow administrative procedures. However, in consideration that the time frame of the Test Review Committee (TRC) approval of the subject Test Result Packages (TRP) is prior to implementation of the licensee's new review process, future inspections will be conducted to evaluate adequacy in this regard for TRPs approved subsequent to this implementation.

c. During the review of C1108.001, the inspector identified the following open items:

(1) The inspector discovered that the Rodworth Minimizer Low Power Setpoint (LPSP) is currently set at 20% power as sensed by steamflow and feedwater flow which is directly on the technical specification limit. This does not allow for sufficient margin

between the technical specification limit for the process variables and the nominal trip setpoint to allow for the inaccuracy of the instruments, uncertainties in the calibration, and instrument drift that could occur during the interval between calibrations as prescribed by Reg. Guide 1.105 Instrument Setpoints. After the inspector's indication of the discrepancy, the licensee has indicated that they will evaluate a new setpoint. This is to remain an open item (341/84-36-03 (DRS)) until the new setpoint has been established and instrumentation is recalibrated to reflect the new setpoint.

(2) During testing it was found that control rod position indication for several odd positions of CRD 30-15 were missing as indicated in TEDR #1. Resolution was to use "as is" for the test and that an operations work order was to provide for rework of the position indicating probe cable. The licensee has indicated that they are currently searching for evidence that the resolution was actually completed. Thus far they have been unable to provide the PN-21 that was used for the rework. This is to remain an open item (341/84-36-04(DRS)) until the PN-21 has been found and provided to the inspector for his review.

d. PRET.R3201.001 130/260 VDC Station Batteries. The inspector reviewed portions of the completed preoperational test data for the Division 1 and 2 safety-related station batteries for compliance with the FSAR, IEEE Standard 450-1975 and the current Proof and Review Technical Specification. Additionally, the inspector reviewed the Duke Final Assessment of Construction report (CAT). The Duke CAT report (significant finding No. 20) identified that the specific gravities for the safety-related batteries were higher than designed. The station batteries are designed with, normally, a specific gravity concentration between 1.210 - 1.220. At the present time, the specific gravity is between 1.240 - 1.260. This increase is the result of the licensee adding 1.400 specific gravity battery acid to correct low specific gravities recorded on Nonconformance Report (NCR) 83-526 dated June 3, 1983.

NCR 83-526 recorded that the Division 1 and 2 safety-related battery pilot cell specific gravities were out of the minimum tolerance of 1.195. As a corrective action, C and D Batteries Division (manufacturer) prescribed a procedure for changing specific gravity in C and D Service Bulletin No. 1 by adding either 1.300 or 1.400 specific gravity acid to raise the specific gravities from 1.185 to 1.215. However, after the licensee had withdrawn and added 1.400 electrolyte to each cell in May of 1984 the measured gravities of each cell were above 1.240.

An Engineering Evaluation Request (EER) 84-130 was written on June 19, 1984, addressing the high specific gravity concentrations in CAT Significant Finding No. 20. The disposition to EER 84-130 was per verbal conversation with DECo Engineering Research (ERD) and C and D Batteries that the high specific gravity concentration in the safety-related batteries is not a concern at this time. Therefore, the licensee's conclusion was to use "as is." Justification was supplied in a C and D Batteries letter to DECo dated August 16, 1984, which stated that "the result of the increased specific gravity will be a slight increase in battery ampere hour capacity" and that "cells with nominal 1.250 specific gravity electrolyte normally operate at a higher voltage than 1.215 cells The float voltage should be raised to 2.25 - 2.30." C and D also stated that they manufacture batteries with 1.215 and 1.250 specific gravities regularly and that "the aging characteristics of batteries with 1.215 and 1.250 specific gravity are similar enough" so that the standard 20 year product life time covers both types.

The licensee is currently submitting a Technical Specification change request to raise the minimum specific gravity limits of Table 4.8.2.1-1 from 1.195 to 1.225 for each designated pilot cell, from 1.190 to 1.220 for each connected cell, and from 1.200 to 1.230 for the average of all connected cells. This change increases the parameters to a base cell specific gravity of 1.250 from the previous base of 1.215. However, the change request did not address or change the float voltage parameters, which is stated as greater than 2.13 volts. C and D Batteries had recommended that the float voltage be increased to 2.25 - 2.30 volts. Since battery performance is based upon the relationship of cell specific gravities, float voltage and battery capacity amperes, this is part of unresolved item (341/84-36-05(DRS)) until the operational float voltage in Technical Specification is addressed.

Additionally, the inspector reviewed the surveillance data collected by the Fermi 2 Technical Group prior to and after the 1.400 specific gravity acid was added to the safety-related batteries. The review of the data taken prior to the addition, which was also used for calculating the amount of acid addition, revealed the following concerns:

(1) Maintenance Instruction MI-E0039, which was used to perform the equalizing charge of the batteries for the addition, required collecting the precharge data after the equalizing voltage is applied to the cells. This resulted in the precharge parameters of specific gravities to be in the range of 1.194 to 1.209 and individual cell voltages to be from 2.36 to 2.40 volts. The post charge data, however, was taken after the equalizing voltage was removed and the float voltage applied. This method of data collection resulted in the post charge to be at lower individual cell voltages and specific gravities: specific gravities from 1.193 to 1.207 and cell voltages from 2.19 to 2.21. Because of the method of data collection, it is not possible to make an independent evaluation of the data to determine the battery's performance or state of charge.

(2) Additional uncertainty of the data is possibly attributed to the instrument used to collect the specific gravity data, the Anton Paar Digital Density Meter DMA-35, which gives a temperature corrected specific gravity readout. Since the DMA-35 does not indicate uncorrected or raw specific gravity data, the independent evaluation of the recorded corrected data must be made upon the calibration of the instrument. The DMA-35 is calibrated by DECo Engineering Research (ERD) to 20°C (68°F) and compensated to 60°F, which is the reference temperature of the calibrating standard. However, the base temperature correction applied to the battery is 77°F. This results in an uncertainty of the data collected in that it is unknown if the documented data is compensated for 60°F, 68°F or 77°F.

The licensee is currently addressing these concerns with a task force. The task force will construct a history of the batteries from the time of manufacturing to present, including assembling and reviewing all related data. Based upon the review, a determination of the acceptability of the batteries will be made by the licensee. This is an unresolved item (341/84-36-05 (DRS) until the determination is made by the licensee and reviewed by the regional staff.

No other items of noncompliance or deviations were identified.

4. Preoperational Test Result Evaluation Verification

The inspector verified that the following preoperational test procedures were written, reviewed and approved by the licensee in accordance with the requirements of Regulatory Guide 1.68 and the QA Manual and found them satisfactory:

PRET N2002.001 Condenser Polishing Demineralizer System
PRET N6100.001 Condenser/Auxiliary Systems
PRET F1300.001 Reactor Assembly and Servicing Equipment

The inspector noted that during PRET F1300.001 preoperational testing, unauthorized operation of equipment by maintenance personnel resulted in various equipment being damaged. The equipment was inspected, repaired and sufficiently retested. Additionally, personnel were retrained and adequate administrative controls implemented to control use of equipment on the refueling floor. Because of the dates of the occurrence, late 1982, and the corrective measures taken by the licensee to prevent any recurrence, this is not an item of noncompliance.

No items of noncompliance or deviations were identified.

5. Preoperational Test Witnessing

The inspector witnessed the following preoperational tests and reviewed associated records to ascertain that testing was conducted in accordance with approved procedures and found them satisfactory. Additionally, the performance of licensee personnel was evaluated during the test and found satisfactory.

a. PRET P4400.001 Emergency Equipment Cooling Water (EECW) System. The inspector witnessed Division II EECW design basis accident verification consisting of isolating the nitrogen supply to the EECW makeup tank, control air to the EECW makeup tank level control valve, and control air to the EECW pressure control valve and periodically recordir ystem operating parameters.

b. PRE .001 Residual Heat Removal (RHR) System. The inspector wit d RHR pump suction valve interlock logic which consisted of verifying correct action of relays with various combinations of suppression pool and shutdown cooling RHR pump suction valve positions. Testing also verified correct open and close permissives for pump suction valves with various suction and discharge valve positions.

c. PRET A8100.001 ECCS Integrated Test. The inspectors witnessed the performance of Section 6.1 simulated LOCA (loss of coolant accident) with normal power available, Section 6.2 simulated LOCA with simultaneous loss of station power (LOSP), Section 6.3 LOSP and simulated LOCA with balance of plant (BOP) AC and DC power supply and Division 2 essential safety feature (ESF) and battery out of service, and Section 6.4 LOSP and simulated LOCA with BOP AC, BOP Battery, Division 1 ESF AC and Division 1 Battery out of service.

The objectives of the test are to demonstrate the response of the Emergency Core Cooling System (ECCS) and supporting systems to LOCA conditions with and without normal offsite power available. Additionally, the test demonstrated: independence between redundant onsite power supplies and their loads, primary and secondary containment isolation functions, Reactor Pressure Vessel Level Low Level 1 trips, and the verification of LPCI and core spray injection. During the performance of this test, the inspectors observed that the licensee's staff conducted the test in a professional manner reflecting improvements in the knowledge of the plant and systems, and the ability of the Nuclear Shift Supervisors, Assistant Shift Supervisors, Technical Shift Advisors, and operators to handle adverse and complex situations.

The following equipment problems were observed by the inspectors:

(1) During the performance of Section 6.2: The "A" Core Spray Pump failed to start with LOCA initiation logic. The licensee troubleshot the initiation logic and was unable to determine any cause except a possible intermittent problem with the time delay relay which did not recur during the retest.

(2) During the performance of Section 6.1, Low Level 2 actuation, the following problems were observed:

(a) The Reactor Core Isolation Cooling System (RCIC) Turbine Steam Inlet Valve (E51-F045) did not position open resulting in the RCIC Pump Inboard Isolation Valve failing to open because of the steam inlet valve/pump isolation valve interlock. The licensee determined that E51-F045 had received the logic signal and that the valve was prevented from opening by a mechanical failure of the limitorque switch. Since the logic was verified to be operable and the failure was mechanical, the licensee will only retest the E51-F045 portion of the preoperational test.

(b) The High Pressure Coolant Injection System (HPCI) Barometric Condenser Pump started as required but was running with high operating amperes. The licensee determined that, even though the piping between the Condensate Storage Tank (CST) and the barometric condenser had been drained prior to performing the test, the CST butterfly-type check valve had leaked and filled the piping through the Lube Oil Cooling Water Supply Valve E41-F059. E41-F059 positions open with a Low Level 2 initiation signal, giving a flow path back to the barometric condenser. The operator stopped the pump to prevent damage and since valve E41-F059 position logic had already been verified, it was subsequently closed to allow draining of the barometric condenser. This action in turn permitted the barometric condenser pump to be restarted so that the Low Level 1 initiation pump trip could later be verified.

(c) The Torus Water Management System Primary Containment Division 1 and 2 Isolation Block Valves did not perform as required. The licensee was not able to fully determine the cause prior to the end of the inspection period. This is an unresolved item (341/84-36-06(DRS)) until the licensee has determined the reason that a temporary modification jumper had prevented the valves from closing as designed and why the effects of the temporary modification were not understood during the test.

(d) The RHR Heat Exchanger "B" Bypass Valve (E11-F048B) positioned to only 95 per cent open during the test. It was determined that the three minute timer had timed out prior to the valve fully opening. The valve is a throttle type valve which requires a continuous automatic or manual signal to open to the 100 percent position. The valve was still operable by the manual pushbutton. The stroke time of the valve is also three minutes and the timer had timed out a few seconds before the valve had fully opened, the licensee will verify the timer's setting to ensure that the valve is still within the required tolerance and will make any needed adjustments. Since the valve is normally open and is given an open signal only as a backup function, the test was able to be continued.

d. PRET T9200.001 Secondary Containment System. The inspector witnessed the preoperational test, including verifying that the Standby Gas Treatment (SBGT) trains obtained required flow and that the containment maintained the desired vacuum. Several problems did develop during the test which will result in the licensee retesting the secondary containment. They included excessive inleakage through the personnel access doors and failures of the access doors' interlocks. Additionally, the ten minute SBGT hold-down required to meet 10 CFR 100 requirements was marginal and will be retested. The inspector will witness the retesting during subsequent inspections.

No items of noncompliance or deviations were identified.

6. Operational and Maintenance Program Implementation

The inspector reviewed various Nuclear Production department programs and procedures to verify that systems that had been preoperational tested or accepted by Nuclear Production were being maintained for configuration control by either scheduled preventive or corrective maintenance in accordance with Regulatory Guides, FSAR and SER.

a. Equipment Protection

In general, the licensee's program to implement equipment protective tagging has not been fully effective throughout all departments within Nuclear Productions. Examples of this were demonstrated by two unresolved items: (341/84-36-05(DRS), jumpers used during the PRET A8100.001 preoperational test tha had prevented the Torus Water Management System Containment Block Valves from performing as required, and (341/84-36-11(DRS) EDG 11 not able to achieve full load. The licensee's current equipment protective tagging program as described in a letter to all Nuclear Production personnel from the superintendent dated July 9, 1984, states that the yellow "Notice" information tag performs the function of protecting equipment. In addition, the licensee has two other systems utilized in the Control Room: the color coded, by panel, numbered discs and information cards assigned to each panel.

Historically, DECo has utilized various systems for personnel protection as implemented in policy order, PPO 77: PN-21 work orders, personnel and operator red tags, and protective barriers. Additionally, the yellow "Notice" tag is used for many of these functions, including informing personnel of equipment status, maintenance or modifications being performed, and equipment protection. However, until a recent change to administrative procedure 12.000.12, Tagging and Protective Barrier System, these tags were uncontrolled. Since, the implemented system was primarily for personnel protection and the equipment tagging was uncontrolled and with many different functions, the licensee has experienced problems with protecting safety-related equipment.

During preoperational testing of the Core Spray System one such problem required removal and inspection of the Core Spray Pump "D" for possible damage. Inspection report 50-341/83-30(DPRP) paragraph 7.a describes the event and identifies that lack of equipment protection contributed to the core spray pump inadvertently running approximately 45 minutes in a dry condition without detection.

The implemented program has weaknesses in the areas of control and information transfer of system status. The color coded tags and information cards utilized by the control room do not adequately identify all components that are repositioned, deenergized, or removed by a PN-21. Therefore, the operator is not able to fully know the status of the system that is tagged out of service. Additionally, the controlled yellow "Notice" information tags may be utilized without operations fully knowing the extent or effects because of the lack of a master status system which would track all components tagged and would be located such that operators could readily obtain an accurate status of any system.

Since the implemented program does not appear to fully meet the intent of both 10 CFR 50, Appendix B, Criterion XIV and ANSI N45.2-1977, Section 15, paragraph 3, which requires that measures shall be established for indicating the operating status of systems and components, such as by tagging valves and switches, to prevent inadvertent operation, this is an unresolved item (341/84-36-07(DRS)) and will be further inspected by the regional staff for compliance to the above codes and regulations.

b. Emergency Diesel Generator (EDG) Start-Failure Log

The inspector reviewed the implementing program and the EDG Start-Failure Logs for EDGs 11, 12,13 and 14 to determine compliance with Regulatory Guide 1.108. The guide requires that a log shall be maintained recording all start attempts, including those from bona fide signals. The log should also describe each occurrence in detail such as to permit an independent determination for valid tests and failures. The regulatory position as outlined in the guide determines valid tests as "successful starts, including those initiated by bona fide signals, followed by successful loading either sequential or manual, to at least 50 per cent (1425KW) of continuous rating and continued operation for at least one hour." The guide also outlines determination of tests that were performed as verification of corrective maintenance as valid tests if they met the load and time requirements as prescribed. Additionally, if the test failed to meet these requirements or was intentionally terminated before completion because of an abnormal condition that would ultimately have resulted in diesel generator damage or failure the test is considered to be a valid test failure. These requirements were not prescribed by the DECo implementing program and resulted in the logs not containing adequate information to permit an independent determination for valid tests or failures.

The log documented 79 starts and start attempts between four diesel generators, of these, 39 entries do not provide the load data or the time loaded or both. In general, less than twenty starts can be independently determined as valid tests. This is an item of noncompliance (341/84-36-08(DRS)) in that the logs failed to collect the data needed to make an independent determination for valid tests or failures as required by Regulatory Guide 1.108 which is committed to by the DECo FSAR.

c. Maintenance Surveillance: Reactor Building Crane

The inspector reviewed the approved temporary maintenance procedure 34.000.43T, Reactor Building Crane Interim Inspection Procedure and data sheets for hoist lifts and crane movements for the month of May 1984. The data collected is utilized to calculate run times of the bridge, main and auxiliary hoists for determination of frequency of required preventive maintenance. However, review of the data revealed that not all lifts were documented and that not all lifts documented bridge and hoist movements as required by procedure 34.000.43T. This is an item of noncompliance (341/84-36-09(DRS)) for failing to follow an approved and implemented procedure. Subsequent to the finding, the maintenance department is conducting an audit, with assistance from Operational Quality Assurance, of a sample of similar work packages for compliance to administrative procedures. This effort will be reviewed during a subsequent inspection period.

d. Emergency Diesel Generator Surveillance

On September 16, 1984, during routine surveillance, EDG 11 was unable to be fully loaded above 2750 KW indicated load. Inspection by the licensee revealed that a knife-type switch to the generator load indicator was in an abnormal open position. The switch in an open position removes the ground connection of the Wye connection to the generator load meter, functionally masking the indication such that the indicated kilowattage was lower than actual load. The operator unloaded the engine and secured the diesel in accordance with the operating procedure. Inspection of the diesel generator did not reveal any damage; however, the licensee has not been able to determine the cause that resulted in the indicator's ground switch being in the open position. This is an unresolved item (341/84-36-10(DRS)) until the licensee determines the cause for the abnormal switch position.

The event with EDG 11 is similar to unresolved item (341/84-36-06) because of the possible involvement of a maintenance work order PN-21. The licensee was unable to completely determine prior to the end of the inspection:

(1) That the ground switch was positioned open by an approved PN-21 and that the diesel generator surveillance was conducted without removing the PN-21 adequately without an independent verification, or

(2) that the switch was positioned open without an approved PN-21.

These events are examples of the licensee's staff not being able to adapt from construction to the operational administrative controls currently being implemented for preparation of fuel load.

e. General Summary

The inspector reviewed four areas of the Nuclear Production program: equipment protection of systems and components, Emergency Diesel Generator Start-Failure Log, maintenance surveillance of the Reactor Building Crane and Emergency Diesel Generators. In those areas, two items of noncompliance and two unresolved items were identified.

No other items of noncompliance or deviations were identified.

7. <u>Meeting with DECo on September 26, 1984</u>

a. Attendance

(1) Detroit Edison (DECo)

R. S. Lenart, Superintendent, Nuclear Production
L. E. Schuerman, Supervisor, Nuclear Engineering
J. E. Conen, Engineer, Licensing
M. K. Deora, Engineer, Nuclear Engineering
R. V. Kezenius, Engineer, Nuclear Engineering

(2) NRC

R. D. Walker, Chief, Operations Branch
R. F. Warnick, Chief, Reactor Projects Branch
L. A. Reyes, Chief, Test Programs Section
S. G. DuPont, Reactor Inspector, Test Programs Section
P. L. Eng, Reactor Inspector, Operational Programs Section
J. McCormick-Barger, Project Inspector, Reactor Projects

b. Meeting Summary

This meeting was conducted at the request of DECo on September 26, 1984, in the Regional Office, to discuss the NRC concern with a recent Design Change Notice (DCN-10481) which changed the stroke time values of motor operated valves. The NRC representatives stated that many of the stroke time requirements were increased significantly by the DCN and that this may affect the transient and safety analysis, specifically the ECCS flow diverting valves such as the full flow bypass test valves. Additionally, the NRC representatives questioned the usage and tolerance of plus or minus 50 per cent stroke time limits on non critical valves.

The licensee stated that the 50 per cent tolerance was obtained from various valve manufacturers and was considered an industrial practice. They also demonstrated that the tolerance was not applied to the valves that they had considered as part of their Appendix K of 10 CFR 50 review or the transient analysis.

The licensee stated that a review will be conducted to determine the acceptability of using the industry practice of 50 per cent tolerance of valve stroke times.

In addition, the licensee will review their transient analysis for any possible impact from changing the stroke times or the usage of the 50 per cent tolerance upon such valves as the minimum flow bypass valves and the feedwater warm-up valves.

Additionally, concerns were discussed by the NRC representatives pertaining to the utilization of preoperational test valve stroke times, with a 50 per cent tolerance for base or reference values for Section XI testing of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME), 1980 edition and addendum. In addition to the licensee reviewing the acceptability of such a large tolerance, the regional staff has requested assistance from the Office of Nuclear Reactor Regulation (NRR) to determine the acceptability of the industrial tolerance and the impact upon Section XI inservice testing. This is an unresolved item (341/84-36-11(DRS)) until NRR has completed their evaluation.

8. Review of Startup Phase Test Procedures

The following Startup Phase Test procedure review has been started but not completed during this inspection period. The review will be completed during a subsequent inspection.

STUT.HUO.004	Full Core Shutdown Margin
STUT.HUA.006	SRM Performance - Initial Criticality
STUT.HUA.010	IRM Performance - SRM/IRM OVERLAP
STUT.HUB.025	MSIV Functional Testing
STUT.HUA.005	CRD Insert and Withdrawal Testing
STUT.HUB.005	CRD Frictional Testing
STUT.HUC.005	Scram Timing Test (Sequence B)
STUT.HUE.005	Scram Timing Test (Sequence A)
STUT.020.026	Relief Valve Testing - Condition 2
STUT.HUO.026	Relief Valve Testing - Heatup
STUT.02A.027	Load Reject Test - within Bypass
STUT.06B.027	Generator Load Reject
STUT.030.018	Core Power Distribution - Condition 3
STUT.060.018	Core Power Distribution - Condition 6
STUT.01A.019	BUCLE Determination
STUT.02B.019	Process Computer Determination
STUT.03B.019	Process Computer Determination
STUT.04B.019	Process Computer Determination
STUT.058.019	Process Computer Determination
STUT.06B.019	Process Computer Determination
STUT.040.021	Core Power Void Response

No items of noncompliance or deviations were identified.

9. Open Items

Open 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 items disclosed during the inspection are discussed in Paragraph 3.

10. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, items of noncompliance, or deviations. Unresolved items disclosed during the inspection are discussed in Paragraphs 3.d, 5.c, 6.a, 6.d and 7.b.

11. Exit Meetings

The inspectors met with site representatives (denoted in Paragraph 1) at the conclusion and during the inspection on September 25 and 28, 1984. The inspector summarized the scope and findings of the inspection.