I Form 12
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U. S. NUCLEAR REGULATORY COMMISSION

OFFICE OF INSPECTION AND ENFORCEMENT

REGION I

IE Inspection Report No: 50-289/76-12

Docket No: 50-289

Licensee: Metropolitan Edison Company
P. O. Box 542
Reading, Pennsylvania 19603

License No: DPR-50

Priority:

Category: C

Safeguards Group:

Location: Three Mile Island 1, Middletown, Pennsylvania

Type of Licensee: PWR, 2535, MWt, B&W

Type of Inspection: Routine, Unannounced

Dates of Inspection: June 8-11, 1976

Dates of Previous Inspection: May 4-7, 1976

Reporting Inspector: William J Raymond for
R. O. Hurd, Reactor Inspector
6/23/76
DATE

Accompanying Inspectors:
DATE

DATE

DATE

Other Accompanying Personnel:
DATE

Reviewed By: T. Rebelowski for
E. C. McCabe, Section Chief, Nuclear Support
Section No. 1, RO&NS Branch
6/23/76
DATE

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SUMMARY OF FINDINGS

Enforcement Action

Items of Noncompliance

None.

Licensee Action on Previously Identified Enforcement Items

A. Corrective actions taken in response to Region I Inspection Report 50-289/76-03 were completed. (Detail 11.a)

B. Corrective actions taken in response to Region I Inspection Report 50-289/76-09 were completed. (Detail 11.b)

Other Significant Findings

A. Current Findings

1. Acceptable Areas

These are areas which were inspected on a sampling basis and findings did not involve an Item of Noncompliance, Deviation, or an Unresolved Item:

a. Startup Testing - Refueling Outage Recovery. (Detail 2)

b. Plant Systems. (Detail 3)

c. Rod Program Verification. (Detail 4)

d. Surveillance Testing. (Detail 5)

e. Shift and Daily Checks. (Detail 6)

f. Reactor Coolant Leakage Evaluation. (Detail 7)

g. Boron Concentration During Refueling. (Detail 8)

h. Design Change Testing Review by CTSS. (Detail 9)

i. Corrective Maintenance. (Detail 10)

2. Unresolved Items

None.

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B. Status of Previous Unresolved Items

1. The following items have been resolved:

   a. Limits on Use of Reactor Building Polar Crane. (Detail 12.c)

   b. Local Leak Rate Tests. (Detail 12.d)

   c. Operating Procedure 1102-1, Revision 13, Plant Heatup to 525°F. (Detail 12.e)

   d. Administrative Control of System Lineups. (Detail 12.g)

2. The following items remain unresolved:

   a. Source Range Operability. (Detail 12.a)

   b. Nuclear Overpower Trip Setting. (Detail 12.b)

   c. Emergency Diesel Generator Monthly Test. (Detail 12.f)

   d. Corporate Technical Support Staff Review. (Detail 11.g)

Management Interview

An exit interview was held onsite on June 11, 1976 at the conclusion of the inspection. The following licensee personnel were in attendance:

Mr. J. Colitz, Plant Superintendent
Mr. G. Kunder, Operations Supervisor
Mr. J. O'Hanlon, Engineer, Senior I
Mr. W. Potts, Quality Control Supervisor
Mr. M. Shatto, Procedure Coordinator

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DETAILS

1. Persons Contacted

Discussions were held with the following persons during the conduct of the inspection activities documented in this report:

Mr. R. S. Brown, Technical Analyst, III
Mr. K. P. Bryan, Shift Foreman
Mr. D. Boltz, Shift Foreman
Mr. J. J. Chwastyk, Shift Supervisor
Mr. J. J. Colitz, Unit No. 1 Superintendent
Mr. D. L. Good, Technical Analysist III
Mr. J. Hilbish, Nuclear Engineer
Mr. S. Jules, Mechanical Maintenance Supervisor
Mr. G. A. Kunder, Supervisor Station Operations
Mr. J. P. O'Hanlon, Engineer, Senior I
Mr. W. Sawyer, Maintenance Engineer
Mr. R. Summers, Project Engineer, Mechanical
Mr. M. Shatto, Procedure Coordinator
Mr. D. M. Shovlin, Maintenance Supervisor
Mr. G. Wallace, Shift Supervisor
Mr. H. Wilson, Instrument Foreman

2. Startup Testing - Refueling Outage Recovery

The inspector reviewed the documentation concerning the testing performed in the areas detailed below to verify that the tests were conducted according to approved procedures and that test data indicated satisfactory test completion.

a. Control Rod Drive Mechanisms and Position Indication

Technical Specification 4.7.1.1 requires that the control rod insertion times shall be measured for each control rod following a refueling outage. The inspector reviewed procedure SP 1303-11.1, Control Rod Drop Time, Revision 3 dated 3/4/76, and the data sheets for this procedure dated 5/21/76. Based on this review, the inspector determined that the 75% insertion time (time measured from coil voltage drop to 75% indication) had been determined for each control rod and that time was recorded on the data sheet. The drop times were all measured with the Reactor Coolant system at 532°F and 2155 psig at rated core flow (147 MLB/R). The longest drop time recorded was 1.296 seconds versus the T.S. 4.7.1.1 allowable maximum time of 1.66 seconds. In addition to the drop times, this test also demonstrated that the rod position indication systems were functioning and that the axial Power Shaping Rods did not drop when de-energized. The latter is a requirement of T.S. 4.7.1.1. The inspector found no inadequacies in this area.

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b. Reactor Coolant Temperature Channels

The inspector reviewed the completed data sheets of SP 1302-5.1, RC Temperature Channel Calibration. Based on this review and discussions with the licensee, the inspector determined that each hot leg RTD resistance bridge circuit calibration had been verified to be within the limits of the acceptance criteria. The inspector found no inadequacies in this area.

c. Core Assembly Verification

The inspector reviewed final core verification form 1505-1D which was developed from TV tapes made of the final core assembly and listed the fuel serial number for each core location. The inspector verified that this form was properly signed off indicating that the Nuclear Engineer and the Core Loading Engineer had verified form 1505-1D and determined that it was identical to form 1505-1C, the desired core map. The inspector independently verified that form 1505-1D and form 1505-1C were identical. The core verification was performed per the instruction of Refueling Procedure 1505-1, Core Assembly. The inspector found no inadequacies in this area.

d. Power Range Nuclear Channels

The inspector reviewed the data sheets of SP 1302-1.1, Power Range Amplifier Calibration, which were completed at the 40%, 75% and 100% power levels during initial power escalation of fuel cycle 2. The inspector verified that, at each of the power level plateaus, the nuclear power range channels were calibrated to be within 1% of the computer calculated core heat balance. The inspector found no inadequacies in this area.

e. Core Physics Test

The core physics testing at the beginning of fuel cycle 2 consisted of the measurement of significant core parameters during low power (less than 1%) and power escalation (at 40%, 75% and 100% of rated power). The reactor was made initially critical by deboration with all control rod groups fully withdrawn with the exception of rod group 7 which was 63% withdrawn. A series of zero power physics tests followed initial criticality to measure the all rods out boron worth, the control rod worths, the shutdown margin, the ejected rod worth and the isothermal temperature coefficients. Additional physics tests were performed at 40%, 75% and 100% of rated power, to determine core power distribution, power doppler coefficient, and the correlation between incore and excore measured power imbalance.

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The inspector reviewed the procedures and the completed test results of the physics tests performed during the return to power following the refueling outage. This review was to verify that the procedure, and any changes to the procedures, were properly approved and that the test results had been reviewed and approved. Tests reviewed follow.

(1) RP-1550-1, Controlling Procedure for Physics Test.

(2) RP-1550-2, Zero Power Physics Test.

(3) RP-1550-3, Dropped Rod Power Distribution Verification.

(4) RP-1550-4, Power Imbalance Detector Correlation Test.

(5) RP-1550-5, Reactivity Coefficients at Power.

(6) RP-1550-8, Core Power Distribution.

The inspector found no inadequacies in this area.

f. Physics Test Results

Based on the inspector's independent review of selected test results and discussions with the licensee, the following tabulation of data results was developed:

(1) Initial Criticality Boron Concentration

Measured: 1337 PPM with rod group 7 @ 63% (5/24/76, RP-1550-2).

Predicted: 1350 PPM with rod group 7 @ 75%.

(2) All Rods Out Boron Concentration

Measured: 1384 PPM (5/24/76, RP-1550-2).

Predicted: 1380 PPM

(3) Moderator Temperature Coefficient

Measured: -10.95×10^{-5} $\Delta K/K/^{\circ}F$ @ 98% power (6/7/76, RP-1550-5).

T.S. 3.1.7 requirement: The Moderator Temperature Coefficient shall not be positive at power levels above 95% of rated power.

(4) Core Power Distribution

Measured: Maximum Linear Heat Rate - 11.67 Kw/ft @ 99.4% power (6/3/76, RP-1550-8).

T.S. Figure 3.5-2J maximum allowable heat rate: 15.5 kw/ft.

The inspector found no inadequacies in the data listed above.

3. Plant Systems

The inspector reviewed the following completed procedures to verify that they were properly approved to indicate that the Reactor Coolant System, Nuclear In-Core Instrumentation and Control Rod Drive System were returned to service in accordance with approved procedures:

a. SP-1308-8.1, Reactor Coolant System.

b. RP-1508-1, Incore Instrumentation.

c. RP-1506-7, CRDM Leadscrew Coupling.

d. RP-1506-9, Shim Safety CRDM Leadscrew Coupling.

e. RP-1506-11, APSR Leadscrew Coupling.

The inspector found no inadequacies in this area.

4. Rod Program Verification

Technical Specification 4.7.2 requires verification that the designated control rod (core positions 1 through 69) is operating in its programmed functional position and group. The inspector reviewed procedure SP-1301-9.2 and the completed data sheets for this procedure which the licensee used to satisfy the Technical Specification. The inspector determined that the documentation indicated that the connections for the control rod drive power and instrumentation cables in the reactor building and the patch connection in the control rod drive cabinets had been verified to be correct. Data sheet 1301-9.2 indicated that each control rod was individually selected and moved and the correct response was verified on the console meters and computer for absolute rod position and relative rod position indication.

The inspector found no inadequacies in this area.

5. Surveillance Testing

The inspector verified that the surveillance tests listed below were performed during the refueling outage per the requirements of the Technical Specifications by reviewing the applicable completed data sheets and documents maintained by the licensee:

a. SP-1303-3, Reactor Coolant Boron Concentration, 2/wk (T.S. Table 4.[illegible], Item 1F).

   NOTE: This test was not performed from 3/10/76 to 4/24/76, due to the vessel being de-fueled and level lowered for Core Support Assembly repair work. (Refer to Inspection Report 76-08, Detail 18.a)

b. SP-1301-5.7, Spent Fuel Pool Boron Concentration, Monthly (T.S. Table 4.1-3, Item 4).

c. SP-1303-4.16, Diesel-Generator Manually Initiated Start, Monthly (T.S. 4.6.1a).

d. SP-1303-5.4, Emergency Feedwater Test, Quarterly (T.S. 4.9).

e. SP-1303-5.5, Control Room Filtering System, Quarterly (T.S. 4.12).

The inspector found no inadequacies in this area.

6. Shift and Daily Checks

The inspector reviewed and witnessed performance of SP-1301-1, Shift and Daily Checks, and verified that its performance documents compliance with the following Technical Specification requirements:

a. T.S. Table 4.1-1, Item 17, LPI Analog Channel Check.

b. T.S. Table 4.1-1, Item 25.b, Core Flood Tank Level Channel Checks.

c. T.S. Table 4.1-1, Item 19, Reactor Building Emergency Cooling and Isolation System Analog Channels Checks.

d. T.S. 2.1.1, Limits on Reactor Coolant Pressure and Temperature.

e. T.S. 2.1.2, Limits on Thermal Power and Power Imbalance.

f. T.S. 2.2.1, Limits on Reactor Coolant System Pressure.

The inspector found no inadequacies in this area.

7. Reactor Coolant System Leakage Evaluation

The inspector reviewed SP-1303-1.1, Surveillance Procedure for RCS Leakage. The stated purpose of this report is, "To evaluate reactor coolant system leakage in accordance with T.S. Table 4.1-2, Item 7." The procedure provides for on-line computer calculation or manual calculation using computer generated input or a complete manual calculation and data collection. The inspector witnessed performance of SP-1303-1.1 using the plant computer calculation. Results of that calculation were a total unidentified leakage of 0.026 gpm at 100% power on 6/9/76. The inspector found no inadequacies in this area.

8. Boron Concentration During Refueling

Technical Specification 3.8.4 requires that the boron concentration in the reactor coolant be maintained at a concentration of 1800 ppm or greater during reactor vessel head removal and while loading or unloading fuel. The inspector reviewed the Shift Foreman's log and data sheets for SP-1301-3, RC Chemistry, to verify that, for the period (from 2/24/76 to 5/23/76) during which head removal and fuel unloading and loading occurred, the boron concentration was 1800 ppm or greater. The inspector found no inadequacies in this area.

9. Design Change Testing Review by CTSS

Technical Specification 6.5.2.A.2c states in part that, ".... it shall be the responsibility of the Met-Ed Corporate Technical Support Staff to specify tests that must be performed following a design change ...." The inspector reviewed Item 21 of checklist GPF 1003.002 completed by the cognizant engineer of the Corporate Technical Staff for each of the 13 design changes implemented during March and April, 1976. The appropriate signoff of this item indicated the requirements of T.S. 6.5.2.A.2c were satisfied. The inspector found no inadequacies in this area.

10. Corrective Maintenance

The inspector reviewed the completed Work Request for the six corrective maintenance items performed on the Makeup and Purification System during May, 1976. The inspector verified that the Work Requests were properly reviewed and approved and indicated compliance with the requirements outlined in FSAR Appendix 1A, Operational Quality Assurance Plan, Revision 6, Section 8.2, Maintenance and Repair. The inspector found no inadequacies in this area.

11. Corrective Action for Previously Identified Items of Noncompliance

a. Region I Inspection 50-289/76-03, Region I letter dated March 25, 1976 and Licensee's response dated April 15, 1976.

The inspector reviewed a memorandum from the Unit Superintendent to the plant supervisor dated April 11, 1976 emphasizing the need to thoroughly check data sheets for errors. The inspector also reviewed the data sheets for SP-1302-7.2, Source Range Calibrations, performed 4/20/76, 5/17/76 and 5/27/76 and found no inadequacies.

This satisfactorily completes the licensee's corrective action for this Item of Noncompliance.

b. Region I Inspection Report 50-289/76-09, Region I letter dated May 13, 1976 and Licensee's response dated June 4, 1976.

The inspector reviewed a memorandum from the Operations Supervisor to the Shift Supervisors, Shift Foremen and Control Room Operators dated May 28, 1976 which stated in part, "Technical Specification requirements are to be performed and data sheets filled out accurately and every detail completed." The inspector also reviewed the Shift and Daily Checks (SP-1301-1) data sheets for June 1 through June 7, 1976 and found no inadequacies.

This satisfactorily completes the licensee's corrective action for this Item of Noncompliance.

12. Previously Unresolved Items

a. Source Range Operability

Reference: Report 50-289/76-03, Detail 2.b.

The licensee's procedure change for SP-1303-7.2 to expand Item 6.2.3 to include documentation of recalibration data is still in the review/approval chain. This item remains unresolved pending the incorporation of this PCR. The licensee indicated that this would be completed by July 2, 1976.

b. Nuclear Overpower Trip Setting

Reference: Report 50-289/76-03, Detail 4.b.

The licensee's procedure change to OP-1102-1 and OP-1102-11 to specify performance of the specific steps in SP-1303-4.1 relating to the 5% overpower trip settings is still in the review/approval chain. This item remains unresolved pending the incorporation of this PCR. The licensee stated that this should be completed by July 2, 1976.

c. Limits on Use of Reactor Building Polar Crane

Reference: Report 50-289/76-03, Detail 22.

The inspector reviewed Work Request No. 13497 which controlled the replacement of a CRDM stator on 1/15/75 with the RCS above 300 psig and 200°F. T.S. 3.12.3 requires that the Reactor Building Polar Crane hoists shall not be operated over the Steam Generator Compartments with the RCS above 300 psig and 200°F. The Work Request included a specific requirement that the polar crane bridge remain lined-up over the fuel transfer canal during this work. This prevents movement of the hoist over the Steam Generator Compartments. The licensee stated that such requirements are contained in all Work Requests involving use of the Polar Crane when the RCS is above 300 psig and 200°F. This item is resolved.

d. Local Leak Rate Tests

Reference: Report 50-289/76-06, Detail 8.f.

The inspector reviewed the Licensee's Event Report, ER 76-19/3L (June 9, 1976) and the data sheets for SP-1303-11.18 which contained the individual value leak rate data and a summation of total leak rate. The licensee has concluded a final review of this data and the measured total leakage for the valves and penetrations is 75,140 sccm which is 67% of 0.6 L_a (111,899 sccm), the limit specified by T.S. 4.4.1.2.3. This item is resolved.

e. Operating Procedure 1102-1, Revision 13, Plant Heatup to 525°F

Reference: Report 50-289/76-06, Detail 7.

The inspector reviewed Enclosure II, Refueling Startup Surveillance Procedure Checkoff of OP-1102-1, Plant Heatup to 525°F, Revision 14. The Reactor Local Leak Rate Test, SP-1303-11.18 is correctly listed and referenced in Enclosure II. This item is resolved.

f. Emergency Diesel Generators Monthly Test

Reference: Report 50-289/76-09, Detail 10.

The licensee stated that the Diesel vendor had not provided new temperature limits for use on data sheets SP-1303-4.16. This item is unresolved until SP-1303-4.16 is modified to reflect realistic temperature limits. The licensee stated that this should be completed by July 2, 1976.

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g. Corporate Technical Support Staff Review

Reference: Report 50-289/76-09, Detail 17.

The inspector determined by review of memoranda from the Manager, Generation Engineering, to the Unit No. 1 Superintendent, that of the 36 PCR's for which CTSS review had not been documented at the time of the referenced report, review had been completed on 16 of the PCR's. This item is unresolved pending documentation of CTSS review of the remaining 20 PCR's.

h. Administrative Control of System Lineups

Reference: Report 50-289/76-10, Detail 5.

The inspector reviewed the pre-heatup checklist of OP-1102-1, Plant Heatup to 525^{o}F, completed prior to the plant heatup following the refueling outage. The inspector determined the following:

(1) Valve lineup for feed system completed 4/25/76 per OP-1106-3.

(2) Valve lineup for Emergency Feed Water System completed 4/15/76 per OP-1106-6.

(3) Valve lineup for Reactor Building Spray System completed 5/21/76 per OP-1104-5.

(4) Valve lineup for Decay Heat River Water System completed 3/5/76 per 1104-32.

(5) Valve lineup for Decay Heat Closed Cooling Water System completed 5/19/76 per 1104-13.

(6) Valve lineup for Intermediate Cooling System completed 5/6/76 per OP-1104-8.

(7) Valve lineup for RB Emergency Cooling and RW System completed 7/22/76 per OP-1104-38.

The inspector noted that for each system listed above, the individual valves in each lineup were verified. This item is resolved.