U. S. NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT

REGION I

Report No. 50-289/81-21

Docket No. 50-289

License No. DPR-50 Priority -- Category C

Licensee: Metropolitan Edison Company
Post Office Box 480
Middletown, Pennsylvania

Facility Name: Three Mile Island Nuclear Station, Unit 1

Inspection At: Middletown, Pennsylvania

Inspection Conducted: July 20-24 and 27-31, 1981

Inspectors: N. J. Blumberg, Reactor Inspector — 9/11/81 date

date

Approved by: D. Caphton, Chief, Management Programs Section, Engineering Branch — 9/23/81 date

Inspection Summary:
Inspection on July 20-24 and 27-31, 1981 (Report No. 50-289/81-21)
Areas Inspected: Routine, unannounced inspection by one region based inspector of licensee action on previous inspection findings; administrative controls for safety related calibrations and surveillances; program for conduct of Technical Specification tests, and calibrations; program for calibration of balance of plant safety related instruments; program for calibration of test and measurement equipment; and compliance to requirements of the NRC Order dated August 9, 1979. The inspection involved 75 inspector-hours onsite by one region based inspector.
Results: Noncompliances - None in five areas and one in one area (Violation - Failure to calibrate safety related instruments and applicable surveillance procedures did not include certain safet elated instruments, paragraph 5.b(1)).

Region I Form 12
(Rev. April 1977)

DETAILS

1. Persons Contacted

*C. Adams, Quality Assurance (QA) Auditor
*B. Ballard, QA Manager, Modifications/Operations
R. Barth, Unit 1 Fire Protection Engineer
*M. Beers, Modification/Operations, Operations and Radcon Monitoring Supervisor
J. Colitz, Plant Engineering Director
R. Eich, Unit 1 Generation Management Systems Analyst
G. Lawrence, Instrument Preventive Maintenance Foreman
T. O'Connor, Lead Fire Protection Engineer, Units 1 and 2
*V. Orlandi, Lead Instrument and Control (I&C) Engineer
*H. Shipman, Operations Engineer
*C. Smyth, Supervisor Licensing
H. Wilson, Lead I&C Foreman
J. Wright, Quality Control Manager

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*A. Fasano, Chief, Three Mile Island Resident Section
*D. Haverkamp, Senior Resident Inspector, Three Mile Island Unit 1
F. Young, Resident Inspector, Three Mile Island Unit 1

The inspector also interviewed other licensee employees, including staff engineers, clerical personnel, and reactor operators.

*Denotes those present at the exit interview.

2. Licensee Action on Previous Inspection Findings

(Closed) Unresolved Item (289/79-20-03). Licensee to determine if spent fuel pool level switches LS-160 and LS-199 were used to comply with technical specification requirements and if they should be included in the calibration program for such instruments. The inspector verified that level switches LS-160, LS-199, and also LS-159 have been added to the instruments used to verify technical specification compliance (procedure 1302-6) and had been calibrated June 23, 1980. The level switches have also been added to the computer scheduling system to ensure that they are calibrated at the frequencies specified in procedure 1302-6, "Calibration of Non-Tech Spec Instruments Used for Tech Spec Compliance".

Based on the above determination, this item is considered closed.

(Closed) Unresolved Item (289/78-20-03). Various instruments not included in procedure 1303-6, "Calibration of Non Tech Spec Instruments Used for Tech Spec Compliance". This item was partially closed during inspection 289/79-08 but was left open since Control Room Emergency Differential

Pressure (D/P) instruments DPI-695 and DPI-696 were inadvertently omitted from procedure 1302-6 through an oversight. The inspector determined that instrument number DPI-696 listed in Inspection Report 289/79-08 was in error; and should have been DPI-697 as DPI-696 does not exist. This determination was based on discussions with a licensee representative, observation of the instrument, and review of the instrumentation diagram. Additionally, the inspector determined that DPI-695 and 697 have been added to procedure 1302-6 and the computer system for scheduling of calibrations.

Based on the above, this item is considered closed. However, a similar finding was observed during this inspection and is identified as an item of noncompliance which is further detailed in paragraph 5.b(1) below.

3. Administrative trols for Safety Related Calibrations and Surveillances

a. Administrative controls governing the performance of safety related calibrations and surveillances were inspected to determine their conformance with the requirements of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants"; Technical Specification, Section 6, "Administrative Controls"; Regulatory Guide 1.33 - 1978, Quality Assurance Program Requirements (Operation) and ANSI 18.7 - 1976, "Administrative Control and Quality Assurance for the Operational Phase of Nuclear Power Plants". The following procedures were reviewed:

-- AP 1022, Control of Measuring and Test Equipment, Revision 11, July 8, 1980.

-- AP 1023, Test Equipment Recall, Revision 4, November 2, 1977.

-- AP 1016, Operations Surveillance Program, Revision 13, December 16, 1980.

-- AP 1010, Technical Specification Surveillance Program, Unit 1 only, Revision 18, October 23, 1980.

-- AP 1036, Instrument Out-of-Service Control, Revision 3, October 23, 1980.

-- AP 1038, Administrative Controls - Fire Protection Program Plan, Revision 2, November 4, 1980.

-- SP 1302-6, Calibration of Non-Tech Spec Instruments Used for Tech Spec Compliance, Revision 6, November 14, 1980.

-- SP 1302-14.1, Calibration of Inservice Inspection Related Instruments, Revision 10, July 17, 1980.

-- AP 1041, Unit #1, ISI Systems List and Retest Requirements, Revision 1, August 24, 1979.

-- SP-1300-1, Inservice Inspection Plan, Revision 3, February 7, 1980.

b. Findings

(1) ANSI N 18.7-1976, Paragraph 5.2.8, requires that a surveillance test and inservice inspection schedule be established. The inspector observed that procedure 1010, "Technical Specification Surveillance Program", states that surveillance test scheduling is performed by the Generation Maintenance System (G.M.S.) Coordinator using a computer program and also observed that this program was being accomplished. Additionally, the inspector noted that calibrations of instruments used to verify Technical Specification surveillance and Inservice Inspection were also being scheduled by the computer system.

Paragraph 1.2 of procedure 1010 states, in part, "The scope of this procedure is limited to the legal requirements and personnel responsibilities of Technical Specification Surveillance...". Procedures 1302-14.1 and 1302-6 which provide for calibration of technical specification and ISI related instruments do not address computer scheduling of these instruments or the responsibilities of the G.M.S. coordinator for this scheduling. Also, procedures 1010, 1302-6, or 1302-14.1 did not provide a means for ensuring that the computer scheduling would be appropriately changed, when procedure changes were made or new procedures were issued which could affect computer scheduling. The inspector expressed a concern that unless computer scheduling were procedurally controlled, inaccuracies could eventually occur in the computer program.

The licensee acknowledged the inspector's concerns and stated that procedure 1010 would be revised to be applicable to "balance of plant" safety related instruments for purposes of computer scheduling and that responsibilities would be assigned to ensure that the computer schedule is changed to reflect any procedure changes which may affect it.

This item is unresolved pending licensee action and subsequent NRC:RI review (289/81-21-02).

(2) Procedure 1022, "Control of Measuring and Test Equipment," Paragraph 4.0.4, references attachment 4, "Calibration Decal"; however, attachment 4 was not included in controlled copies of the procedure reviewed by the inspector. A licensee representative stated that procedure 1022 would be corrected to either delete or include attachment 4. This item will be reviewed during a subsequent NRC:RI inspection (289/81-21-03).

(3) Procedure 1302-6, Table 1, provides required calibration frequencies for listed instruments. However, the calibration frequencies are not given for the Multi Channel Analyzer and Gas Chromatograph, H_2, O_2, Cl Analyzer which are listed in Table 1. A licensee representative stated that procedure 1302-6 would be revised to include the calibration frequencies of these instruments. This item will be reviewed during a subsequent NRC:RI inspection (289/81-21-04).

4. Inspection of Program for Conduct of Surveillance Tests, Inspections and Calibrations Required by Technical Specifications

An inspection was conducted of the program for establishment, control, performance, and documentation of surveillance tests, inspection and calibrations which are required by the Technical Specifications. The program was inspected to determine conformance to the requirements of 10 CFR 50, Appendix B, "Quality Assurance Criteria"; Technical Specification, Section 6, "Administrative Controls"; and Regulatory Guide 1.33-1978, Quality Assurance Program Requirements (Operation) and ANSI 18.7-1976, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants".

The following areas were verified:

-- Master schedule(s) had been established for surveillance testing, inspection and calibrations;

-- Responsibilities had been assigned for performance of surveillance tests, inspection and calibrations and to assure the schedules were satisfied;

-- Methods and responsibilities had been established for review and evaluation of data, for reporting of test deficiencies and failures, and for verification that LCO requirements had been satisfied;

-- Calibration frequency requirements were as established by the Technical Specifications; and

-- Procedures were established for performance of tests and these procedures had been properly reviewed and approved.

No items of noncompliance were identified.

5. Inspection of Program for Conduct of Calibrations of Plant Installed Instruments Used for Verification of Technical Specification Surveillance Tests and Inservice Tests and Inspections

a. An inspection was conducted of the program for (establishment, control, performance, and documentation of the) calibration of plant installed instruments which are used to verify satisfactory performance of Technical Specification surveillance tests and inspections and

for tests and inspections required by the Inservice Inspection Program. The program was inspected to determine conformance to the requirements of 10 CFR 50, Appendix B, "Quality Assurance Criteria...; Technical Specification, Section 6, "Administrative Controls"; Regulatory Guide 1.33-1978, "Quality Assurance Program Requirements (Operation)" and ANSI 18.7-1976, "Administrative Controls...". The following areas were verified:

-- A master schedule had been established for calibration testing;

-- Responsibilities had been assigned for performance of calibrations and to assure that calibration schedules are satisfied;

-- Calibration frequencies had been established by the licensee;

-- Procedures had been established for performance of calibrations and these procedures had been properly reviewed and approved; and

-- Methods and responsiblities had been established for review and evaluation of data, for reporting of test deficiencies and failures, and for verification that LCO requirements had been satisfied.

b. Findings

(1) During the inspection of the program for calibration of instruments used to verify Technical Specification Surveillance and Inservice Inspection Tests, the inspector determined that the following instruments had not been incorporated into the respective calibration procedures in that:

-- Flow instruments DH-FI-299A and 299B which are used to verify Decay Heat Removal Pumps flow per Procedure 1300-3B A/B, "Decay Heat Removal Pump Functional Test, were not included in procedure 1302-14.1, "Calibrations of Inservice Inspection Instruments". The purpose of procedure 1302-14.1 is to ensure that all inservice related instruments are within calibration per ASME Section XI; hence, these instruments should be included in this procedure.

Additionally, procedure 1302-14.1 specifies a calibration frequency of eighteen months. Plant records indicate that DH-FI-299B was last calibrated May 15, 1975. However, DH-FI-299 A&B had recently been placed in a non-safety related preventative maintenance (P.M.) program and DH-FI-299A had last been calibrated June 20, 1980.

-- The Diesel Fire Pumps Fuel Oil Storage Tanks Capacity gages which are used to measure the amount of fuel oil in the tanks as required by Technical Specification 4.18.2.2.a.1 and procedure 3303-MI, "Fire Pump Periodic Operation" were not included in procedure 13U2-6, "Calibration of Non-Tech Spec Instruments used for Tech Spec Compliance".

Additionally, the licensee could produce no records verifying that these gages had ever been calibrated. Licensee representatives interviewed by the inspector had no recollection of these gages being calibrated and stated that failure to calibrate these gages had been an oversight.

-- Flow instrument, FI-155, which is designated by procedure 3303-3Y1, "Fire System Capability Test," to verify fire pump flow was also not included in procedure 1302-6. Technical Specifications require fire pump capacity to be tested every three years.

A licensee representative stated that 3303-3Y1 was a three year test and had not yet been performed but was scheduled to be performed later during 1981. However, there were no plans to calibrate FI-155. A licensee representative stated that use of FI-155 would be re-evaluated in that fire pump capacity is tested every 18 months per procedure 3303-R2. During this test, flow is measured using a test instrument.

Failure to include instruments used for Technical Specification Surveillance Tests and Inservice Inspections into appropriate procedures and failure to calibrate safety related instruments or to calibrate in a timely manner is contrary to 10 CFR 50, Appendix B, Criterions IV and XII; Regulatory Guide 1.33 - 1978; and ANSI 18.7-1976 collectively constitute an item of noncompliance (289/81-21-01).

(2) To inspect Fire Protection System instrumentation, the inspector toured the structure housing Diesel Fire Pump No. 3 located adjacent to the North Side of the Unit 1 Pump and Screen House. During this inspection, two outdated and uncontrolled copies of procedure 3303-MI, "Fire Pump Periodic Operation", were visually observed to be posted near the fire pump. The fire pump was not being operated during this inspection.

There was no evidence that these outdated procedures had been used for recent pump operation. Also station administrative procedures should preclude the use of an outdated procedure to perform surveillance test since working copies are issued from the Control Room files. However, the inspector informed the

licensee's representative that uncontrolled copies of procedures should not be posted at locations where they could inadvertently be used to operate equipment.

Licensee representatives acknowledged the inspector's comments and stated that all fire pump rooms would be inspected and uncontrolled copies of procedures would be removed.

This items is unresolved pending licensee action and subsequent NRC:RI review (289/81-21-05).

6. Inspection of Program for the Calibration and Control of Test and Measurement Equipment

An inspection was conducted of the program for control and calibration of test and measurement equipment. The program was inspected for conformance to the requirements of 10 CFR 50, Appendix B, "Quality Assurance Criteria..."; Technical Specification Section 6, "Administrative Controls"; and Regulatory Guide 1.33-1978, "Quality Assurance Program Requirements (Operation)" and ANSI 18.7 - 1976, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants". The following areas were verified:

-- A master test and measurement equipment list had been established;

-- A test and measurement equipment calibration schedule had been established;

-- Responsibilities had been assigned to assure test and measurement equipment were calibrated and schedules were met;

-- Responsibilities had been established for review and evaluation of test and measurement equipment calibration data;

-- Records were maintained identifying standards used and their traceability to National Bureau of standards or other testing organizations;

-- Test equipment custody control records were maintained;

-- Responsibilities had been assigned to assure proper storage and labeling of test equipment; and

-- Methods had been established to assure traceability of out of calibration test equipment to usage of that equipment.

No items of noncompliance were identified.

7. Review of Compliance to Requirements of NRC Order dated August 9, 1979

The inspector reviewed licensee procedures to determine if the requirements of NRC Order dated August 9, 1979, as evaluated in portions of NUREG-0680, TMI-1 Restart Evaluation Report (RER), and its supplements 1, 2, and 3 had been incorporated. Based on this review, the following was determined:

(1) RER item C.1.a.4.a states "Plants that require local manual realignment of valves to conduct periodic tests of one EFW system train and have only one remaining EFW train available shall provide that a dedicated individual, who is in communication with the Control Room, be stationed at the manual valves, and on instruction from the Control Room, realign the valves from the test mode to their operational alignment."

In their response to this item, the licensee stated that the only local manual valves requiring realignment during periodic inservice inspection pump testing are the motor driven Emergency Feedwater (EFW) Pump discharge valves. The inspector verified that surveillance procedure 1300 F A/B, "Motor Driven Emergency Feedwater Pump Functional Test Surveillance Frequency - 31 Days (M) Pump Test", paragraph 6.6, has been revised to read the following: "6.6 - Establish communication with the control room using local head phones. If the need for emergency feedwater occurs, upon instruction from the control room, open EF-V-10 A (B) EFW pump discharge valve so to provide emergency feed to the steam generators. Maintain communication with control room from step 6.7 through 6.23." Additionally, the inspector verified that a set of sound powered headset phones are maintained at the motor driven EFW pumps station and that there is also a permanently installed phone jack. The inspector also verified that the headset phone wires are of sufficient length for the operator to reach the EFW pump discharge valves while he is wearing the headsets. Based on the above, the procedural aspect of this item is considered to be in accordance with the RER statement.

(2) RER item C.1.a.4.b states the following:

"b. Prior to startup after an extended cold shutdown, a flow test shall be performed to verify the normal EFW flow path to the steam generators." The inspector verified that the pre-heatup checklist of operating procedure 1102-1, "Plant Heatup to 525°F", has been revised to add procedure 1303-11.42, "EFW Flow Test from the Condensate Storage Tank", completion as a prerequisite plant heatup item. However, the inspector determined that procedure 1303-11.42 is in draft form only and has not yet been issued. This item remains open pending issuance of procedure 1303-11.42 and further NRC inspection to determine if it is in accordance with the RER statement.

(3) RER item C.1.a.7, states the following:

"7. To assure that AFW will be aligned in a timely manner to inject on all AFW demand events when in surveillance test mode, procedures will be implemented and training conducted to provide an operator at the necessary location in communications with the Control Room during the surveillance mode to carry out alignment changes necessary upon AFW demand events."

The NRC evaluation of this item states, in part "... A check on the location and verification of adequate communication will be performed prior to restart...". As previously stated in paragraph 7(1) above, the inspector verified the adequacy and location of this communication and this aspect of the item is considered to be in accordance with the RER statement.

(4) RER item C.1.d, states in part, that the low pressure reactor trip setpoint be changed from 1800 psig to 1900 psig and the High Pressure Injection actuation setpoint be changed from 1500 psig to 1600 psig. The inspector verified that the following procedures had been revised to include the 1900 psig and/or 1600 psig setpoint, as applicable:

-- 1101-2, "Plant Setpoints", Revision 17, January 15, 1981;

-- 1103-5, "Pressurizer Operation", Revision 15, February 25, 1981;

-- 1105-2, "Reactor Protection System", Revision 15, April 10, 1981;

-- 1105-3, "Safeguards Actuation System", Revision 8, February 10, 1981;

-- 1202-4, "Reactor Trip", Revision 20, March 13, 1981;

-- 1202-6A, "Loss of RC/RCP Within Capability of Makeup System (RC Pressure above ESAS Set Point)", Revision 8, March 13, 1981;

-- 1202-6B, "Loss of RC/RC Pressure (Small Break LOCA) Causing Automatic High Pressure Injection", Revision 8, June 24, 1981;

-- 1202-6C, "Loss of RC/RCP Causing Automatic High Pressure Injection, Core Flood, and Low Pressure Injection", Revision 7, March 13, 1981.

-- 1202-14, "Loss of Reactor Coolant Flow/RCP Trip", Revision 11, March 18, 1981;

-- 1202-29, "Pressurizer Systems Failure", Revision 16, June 29, 1981;

-- 1302-5.2, 5.3, "RPS High and Low RC Pressure Channel Calibration Required Interval Refueling", Revision 4, January 15, 1981;

-- 1303-4.1, "Reactor Protection System", Revision 5, September 26, 1980;

-- 1303-4.11, "High and Low Pressure Injection Analog Channels", Revision 36, January 5, 1981; and

-- Alarm Response Procedure FI-1.

Based on the above, the procedural aspect of this item is considered to be in accordance with the RER statements.

(5) RER item C.2,IE Bulletin 79-05B part 3, states, in part, the following:

"3. Following detailed analysis, describe the modifications to design and procedures which you have implemented to assure the reduction of the likelihood of automatic activation of the pressurizer PORV during aniticipated transients. This analysis shall include consideration of a modification of the high pressure scram set point and the PORV opening set point such that reactor scram will preclude opening of the PORV for the spectrum of anticipated transients ... Changes developed by this analysis shall not result in increased frequency of pressurizer safety valve operation for these anticipated transients" and, RER item C.1.d, requires, in part, that the set point of the pressurizer electromatic relief valve (PORV) be changed from 2255 psig to 2450 psig and the high pressure reactor trip setpoint be changed from 2390 psig to 2300 psig.

The inspector verified that the following procedures had been revised to include the new 2450 psig and 2300 psig setpoints, as applicable:

-- 1101-2, "Plant Setpoints", Revision 17, January 15, 1981;

-- 1103-5, "Pressurizer Operation", January 15, 1981;

-- 1105-2, "Reactor Protection System", Revision 15, April 10, 1981;

-- 1202-4, "Reactor Trip", Revision 20, March 13, 1981;

-- 1202-29, "Pressurizer Systems Failure", Revision 16, June 29, 1981;

-- 1203-1, "Load Rejection", Revision 8, January 17, 1981;

-- 1302-5.2, 5.3, "RPS High and Low RC Pressure Channel Calibration Required Interval Refueling". Revision 4, January 15, 1981; and,

-- 1303-4.1, "Reactor Protection System", Revision 5, September 26, 1980.

However, the inspector determined that procedure 1302-6.16, "PORV setpoint calibration", is in draft form and has not yet been issued. The procedural aspect of this item remains open pending issuance of 1302-6.16 and further NRC inspection to determine if it is in accordance with the RER statements.

(6) RER item C.2,IE Bulletin 79-05, part 5, states:

"5. Provide for NRC approval a design review for implementation of a safety grade automatic anticipatory reactor scram for loss of feedwater, turbine trip or significant reduction in steam generator level."

In response to their item, the licensee stated that a safety grade reactor trip system would be installed which will implement a reactor trip on loss of both main feedwater pumps or on a turbine trip. The inspector determined that these trips had only recently been installed and had not yet been tested. Additionally, applicable procedures have not yet been revised to include these new scram signals. This item remains open pending completion of licensee action and further NRC inspection to determine if the licensee is meeting commitment stated in the RER.

8. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable, deviations, or items of noncompliance. Two unresolved items were identified during this inspection and are detailed in paragraphs 3.b(1), 3.b(2), 3.b(3) and 5.b(2).

9. Management Meetings

Licensee Management was informed of the scope and purpose of the inspection at the entrance interview conducted on July 20, 1981. The findings of the inspection were periodically discussed with licensee representatives during the course of the inspection. An exit interview was conducted on July 31, 1981 (see paragraph 1 for attendees) at which time the findings of the inspection were presented. A subsequent telephone discussion concerning the inspection findings was conducted between the inspector and Mr. J. Colitz on August 5, 1981.