ATTACHMENT TO LICENSE AMENDMENT NO. 4

FACILITY OPERATING LICENSE NO. DPR-73

DOCKET NO. 50-320

Change the following pages of the Appendix "A" Technical Specifications with the enclosed pages as indicated. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Pages

3/4 2-11
3/4 3-12
3/4 3-17
3/4 3-18
3/4 3-19
3/4 3-22

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3. REACTOR BUILDING SPRAY					
a. Reactor Building Pressure High-High	3***	2***	2***	1, 2, 3	10#
b. Automatic Actuation Logic	2	1	2	1, 2, 3	11
4. REACTOR BUILDING SUMP SUCTION					
a. BWST Level-Low	1/train	1	1/train	1, 2, 3, 4	9
5. FEEDWATER LATCHING					
a. Main Steam Pressure Low	4/St. Gen (2/Main St. line)	1/St. Line	2/St. Line	1, 2, 3****	9
6. FEEDWATER LINE RUPTURE DETECTION					
a. Feedwater/Main Steam Line Differential Pressure Low	1/St. Gen	1/St. Gen	1/St.Gen	1, 2, 3	9

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. LOSS OF POWER					
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)					
1. Emergency Bus #2-1E and 2-2E	2/Bus	2/Bus	2/Bus	1, 2, 3	10
2. Emergency Bus #2-3E and 2-4E	2/Bus	1/Bus	2/Bus	1, 2, 3	11
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)					
1. Emergency Bus #2-1E and 2-2E	2/Bus	2/Bus	2/Bus	1, 2, 3	10
2. Emergency Bus #2-3E and 2-4E	2/Bus	1/Bus	2/Bus	1, 2, 3	11
8. CHEMICAL ADDITION SIGNAL					
a. Reactor Building Cooling and Isolation Initiation	(same		as	2. a/b/c)	
b. Safety Injection and	(same		as	1. a/b/c/d/e)	
BWST Level Initiation	1/train	1	1/train	1, 2, 3, 4	9

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-2:

a. Reactor Coolant Hot Leg Temperature.

b. Reactor Coolant Pressure.

c. Reactor Coolant Flow Rate.

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-2 shall be verified to be within their limits at least once per 12 hours.

4.2.5.2 The Reactor Coolant System total flow rate shall be determined to be within its limit by measurement at least once per 18 months.

TABLE 3.2-1

QUADRANT POWER TILT LIMITS

	STEADY STATE LIMIT	TRANSIENT LIMIT	MAXIMUM LIMIT
Measurement Independent QUADRANT POWER TILT	3.69	9.74	20.0
QUADRANT POWER TILT as Measured by:			
Symmetrical Incore Detector System	2.30	7.71	20.0
Power Range Channels	0.96	5.88	20.0
Minimum Incore Detector System	1.72	3.71	20.0

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEMS INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT		ALLOWABLE VALUES	
7. LOSS OF POWER continued				
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)*				
1. Emergency Bus #2-1E and 2-2E	(±) volts with a (±) second time delay		(±) volts with a (±) second time delay	
2. Emergency Bus #2-3E and 2-4E	(±) volts with a (±) second time delay		(±) volts with a (±) second time delay	
8. CHEMICAL ADDITION SIGNAL				
a. Reactor Building Cooling and Isolation Initiation	(same	as	2. a/b/c	)
b. Safety Injection and BWST Level Initiation	(same 53'9" ± 2.9"	as	1. a/b/c/d/e 53'9" ± 3"	)

*Not required until the first refueling per U.S.N.R.C. letter dated August 26, 1977 from S. A. Varga, Light Water Reactors Branch #4, Division of Project Management to Metropolitan Edison Company, subject: Transmittal of Staff Positions 222.46 and 222.47. A proposed change to the Technical Specifications to incorporate trip setpoint and allowable values shall be submitted to the NRC at least 90 days prior to start up after the first refueling.

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. Safety Injection	Not Applicable
b. Reactor Building Cooling	Not Applicable
c. Reactor Building Isolation	Not Applicable
d. Reactor Building Spray	Not Applicable
e. Reactor Building Sump Suction	Not Applicable
f. Feedwater Latching	Not Applicable
2. Reactor Building Pressure High	
a. High Pressure Injection	≤ 25*/25**
b. Low Pressure Injection	≤ 25*/25**
c. Reactor Building Cooling	≤ 125*/125**
d. Reactor Building Isolation	
(1) Reactor Building Purge Isolation	≤ 5*/5**
(2) Reactor Building Isolation	≤ 60*/60**
(3) MU-V25	≤ 75*/75**
e. Control Building Emergency Ventilation	
(1) Control Room Isolation	≤ 6*/6**
(2) Control Building HVAC	≤ 900*/900*
f. Component Cooling Water Systems	
(1) Decay Heat Closed Cooling	≤ 300*/300**
(2) Nuclear Services Closed Cooling	≤ 95*/NA**
g. Service Water System Nuclear Services River Water	≤ 95*/95**
h. Reactor Building Spray Valves	≤ 23*/23**
(1) BS-V1A/B	≤ 23*/23**
(2) DH-V8A/B	≤ 36*/30**(+)

TABLE 3.3-5 (continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Reactor Building Pressure--High-High	
a. Reactor Building Spray Pumps	≤ 31*/31*
4. Reactor Coolant Pressure-Low	
a. High Pressure Injection	≤ 25*/25**
b. Low Pressure Injection	≤ 25*/25**
c. Component Coolant Water System	
(1) Decay Heat Closed Cooling	≤ 300*/300**
(2) Nuclear Services Closed Cooling	≤ 95*/NA**
d. Service Water System (Nuclear Services River Water)	≤ 95*/95**
5. Feedwater Latching	
a. Main Steam Isolation	≤ NA*/124.6**
b. Feedwater Isolation	
(1) FW-V30A/B	≤ NA*/9.2**
(2) FW-V17A/B	≤ NA*/32.6**
(3) FW-V25A/B	≤ NA*/14.6**
(4) FW-V19A/B	≤ NA*/32.6**
6. Emergency Feedwater Pump Actuation	
a. Turbine Driven Pump	≤ NA*/29**
b. Motor Driven Pumps	≤ 29*/12**

TABLE NOTATION

*Diesel generator starting and sequence loading delays included. Response time limit includes movement of valves and attainment of pump or blower discharge pressure.

**Diesel generator starting and sequence loading delays not included. Offsite power available. Response time limit includes movement of valves and attainment of pump or blower discharge pressure.

(+)Response time applicable for Reactor Building cooling and isolation only.

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEMS INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. SAFETY INJECTION (High Pressure and Low Pressure Injection)				
a. Manual Initiation	NA	NA	M (1)	1, 2, 3, 4
b. RCS Pressure-Low	S	R	M	1, 2, 3
c. R.B. Pressure-High	S	R	M (3)	1, 2, 3
d. R.B. Cooling & Isolation Manual Initiation	NA.	NA	M (1)	1, 2, 3, 4
e. Automatic Acutation Logic	NA	NA	M (2)	1, 2, 3, 4
2. REACTOR BUILDING COOLING AND ISOLATION				
a. Manual Initiation	NA	NA	M (1)	1, 2, 3, 4
b. R.B. Pressure-High	S	R	M (3)	1, 2, 3
c. Automatic Acuation Logic	NA	NA	M (2)	1, 2, 3, 4
3. REACTOR BUILDING SPRAY				
a. Reactor Building Pressure High-High	S	R	M (3)	1, 2, 3
b. Automatic Actuation Logic	NA	NA	M (2)	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEMS INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
4. REACTOR BUILDING SUMP SUCTION				
a. BWST Level-Low	NA	R	NA	1, 2, 3, 4
5. FEEDWATER LATCHING				
a. Main Steam Pressure-Low	NA	R	NA	1, 2, 3
(1) Manual actuation switches shall be tested at least once per 18 months during shutdown All other circuitry associated with manual safety features actuation shall receive a CHANNEL FUNCTIONAL TEST at least once per 31 days.				
(2) Each logic channel shall be tested at least every other 31 days.				
(3) The CHANNEL FUNCTIONAL TEST shall include exercising the transmitter by applying either vaccum or pressure to the appropriate side of the transmitter.				
6. FEEDWATER LINE RUPTURE DETECTION				
a. Feedwater Line/Main steam line differential pressure low	NA	R	NA	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEMS INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
7. LOSS OF POWER				
a. 4.16 kv Emergency Bus Undervoltage (Loss of (Voltage				
1. Emergency Bus #2-1E and 2-2E	S	R	M	1, 2, 3
2. Emergency Bus #2-3E and 2-4E	S	R	R	1, 2, 3
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)				
1. Emergency Bus #2-1E and 2-2E	S	R	M	1, 2, 3
2. Emergency Bus #2-2E and 2-4E	S	R	R	1, 2, 3
8. CHEMICAL ADDITION SIGNAL				
a. Reactor Building Cooling and Isolation Initiation	(same	as	2. a/b/c	)
b. Safety Injection and	(same	as	1. a/b/c/d/e	)
BWST Level Initiation	N/A	R	N/A	1, 2, 3, 4

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-320

METROPOLITAN EDISON COMPANY
JERSEY CENTRAL POWER & LIGHT COMPANY
PENNSYLVANIA ELECTRIC COMPANY

THREE MILE ISLAND NUCLEAR STATION, UNIT 2

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment 4 to Facility Operating License No. DPR-73, issued to the Metropolitan Edison Company, Jersey Central Power & Light Company, and Pennsylvania Electric Company, for operation of the Three Mile Island Nuclear Station, Unit 2 (the facility), located in Dauphin County, Pennsylvania. The amendment is effective as of its date of issuance.

The license is amended by revising certain Technical Specifications.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51(d)(4), an environmental statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) Amendment No. 4, to Facility Operating License No. DPR-73, and (2) the Commission's related safety evaluation supporting Amendment No. 4 to Facility Operating

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License No. DPR-73. These items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., and at the State Library of Pennsylvania, Commonwealth and Walnut Streets, Harrisburg, Pennsylvania 17126. A copy of items (1) and (2) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Project Management.

Dated At Bethesda, Maryland, this 19th day of May 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by
Steven A. Varga

Steven A. Varga, Chief
Light Water Reactors Branch No. 4
Division of Project Management

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
SUPPORTING AMENDMENT NO. 4 TO FACILITY OPERATING LICENSE NO. DPR-73

METROPOLITAN EDISON COMPANY
JERSEY CENTRAL POWER & LIGHT COMPANY
PENNSYLVANIA ELECTRIC COMPANY

MAY 1 9 1978

DOCKET NO. 50-320

THREE MILE ISLAND NUCLEAR STATION, UNIT 2

1. Sodium Hydroxide (NaOH) Injection Signal

Introduction

By letter dated May 10, 1978 transmitting Technical Specification Change Request No. 007, Metropolitan Edison Company (Met Ed) requested amendment of Appendix A to Facility Operating License No. DPR-73 for Three Mile Island Nuclear Station Unit 2 (TMI-2). The requested change would amend the Technical Specifications to permit avoiding injection of NaOH into the reactor coolant system during inadvertent actuations of the emergency core cooling system (ECCS).

Discussion

NaOH is injected into the Borated Water Storage Tank discharge line in the event of a LOCA to provide corrosion control and to enhance iodine removal capability of the reactor building spray system.

At present, either of two signals open the sodium hydroxide injection valves: 1600 psig Reactor Coolant System (RCS) pressure or 4 psig reactor building pressure. As a result of two recent occurrences where sodium hydroxide was inadvertently and unnecessarily injected into the primary system when the RCS pressure went below 1600 psig, the applicant has proposed modifying the sodium hydroxide injection valve actuation signal to require that the valve be actuated by a decreased level of the Borated Water Storage Tank (BWST) simultaneously with an RCS pressure below 1600 psig, or by a 4 psig reactor building pressure. This modification would allow a period of time for safety injection without sodium hydroxide addition for those events where reactor building pressure does not rise above 4 psig before the BWST level initiation, and would allow the operator to manually prevent hydroxide addition for those situations where it was not required.

The license has presented an analysis which concludes that this modification will reduce the probability of spurious NaOH injection without degrading the functional capability of the system. Chemistry control for corrosion

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protection will be maintained, and the plant response to any accident in which to NaOH injection would be necessary to reduce any offsite doses to within acceptable limits, would remain unchanged.

In addition to the above, in a letter dated May 12, 1978, Met Ed stated that the redundant switches which generate the new BWST level signal have appropriate accuracy and repeatability characteristics, that they are qualified to seismic Category I requirements, and that the redundant signal cables are routed in separate safety related and seismically qualified raceways. Met Ed further committed to updating the FSAR in a future amendment to reflect these changes.

Evaluation

We have reviewed the information provided by the licensee, and find that the proposed change is desirable in that it will reduce the probability of unnecessary injection of NaOH into the reactor colant system, and that the proposed change will not degrade the functional capability of the system. We further find that the components provided are appropriate and redundant and conform with seismic Category I requirements.

Based on the above, we conclude that the proposed change in the initiation of NaOH injection is acceptable, and that the facility operating license can be amended by changing the Technical Specifications as shown in the attachment to this license amendment.

2. Quadrant Power Tilt

Introduction

By letter dated May 10, 1978, Metropolitan Edison Company (Met Ed) requested amendment of Appendix A to Facility Operating License No. DPR-73 for Three Mile Island Nuclear Station, Unit 2 (TMI-2). The requested change would amend the Technical Specifications to reduce the maximum allowable value of neutron flux tilt as measured in each quadrant of the reactor core by in-core or out-of-core neutron detectors.

Discussion

Babcock and Wilcox (B&W) performed the initial error analysis for quadrant tilt and axial imbalance derived from incore signals based on data obtained from prototype detectors in 1974. As a result of observations of operating characteristics of these detectors in operating reactors a re-evaluation of the error analysis has recently been performed by B&W. This re-evaluation has resulted in an increase in the measurement uncertainty for tilt and imbalance with a consequent necessity for altering alarm setpoints for these quantities. By letter dated May 11, 1978, B&W has submitted a report on this re-evaluation. This report was used as the basis for the evaluation of the request for the Technical Specification change.

B&W has performed a statistical analysis of the measurement of quadrant tilt and axial imbalance and established an error which assures that the alarm setpoint will be reached or exceeded 95% of the time when the measured quantity is at its limit value. The analysis was performed by the Monte Carlo technique. Individual detector signal components (rhodium signal, background signal, etc.) were chosen from distributions of these quantities which were obtained from critical experiments and operating reactors. Conservative individual uncertainty components were used. Limiting valves of the "real" tilt and imbalance were assured and "measured" values were obtained by performing the same calculations that are done by the online computer. The error value was then chosen so that the alarm setpoint was reached or exceeded 95% of the time. The Monte Carlo analysis was performed with 5000 trials.

B&W has also investigated the effect of the new data base on incore detector uncertainty on the measurements of $F_{\Delta H}$ and F_Q. Comparisons were made between calculations and measurements of these quantities in several operating reactors. The measurement uncertainty was inferred from these comparisons by assuming a conservatively small calculational uncertainity and ascribing the rest of the difference to measurement uncertainty. The results showed that the presently used uncertainties (5% for $F_{\Delta H}$ and 7.5% for F_Q are conservative.

Evaluation

Based on our review of this document we conclude that the method of analysis is acceptable. We further conclude that the values of alarm setpoints for quadrant tilt and axial imbalance recommended for TMI-2 are acceptable.

We note that our review of the B&W submittal of May 11, 1978, has not been fully completed, but that it has progressed sufficiently so that we have been able to evaluate and find acceptable the specific changes in alarm setpoints for TMI-2, as stated above.

Since the proposed changes to the TMI-2 Technical Specifications follow the applicable B&W recommendations, we conclude that the requested changes are likewise acceptable. We further conclude that no changes are necessary in the uncertainty values assigned to measurements of $F_{\Delta H}$ and F_Q, and that the facility operating license can be amended by changing the Technical Specifications as shown in the attachment to this license amendment.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amonts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

H. Silver, Project Manager
Light Water Reactors Branch No. 4
Division of Project Management

MAY 1 9 1978

Steven A. Varga, Chief
Light Water Reactors Branch No. 4
Division of Project Management

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
631 PARK AVENUE
KING OF PRUSSIA, PENNSYLVANIA 19406

Reference 37

Docket No. 50-320

30 NOV 1978

Metropolitan Edison Company
ATTN: Mr. J. G. Herbein
Vice President - Generation
P. O. Box 542
Reading, Pennsylvania 19603

Gentlemen:

Subject: Inspection 50-320/78-33

This refers to the inspection conducted by Mr. D. Haverkamp of this office on November 7-9 and 16-17, 1978, at Three Mile Island Nuclear Station, Unit 2, Middletown, Pennsylvania, of activities authorized by NRC License No. DPR-73 and to the discussions of our findings held by Mr. L. Bettenhausen of this office with Mr. R. Toole of your staff at the conclusion of the inspection.

Areas examined during this inspection are described in the Office of Inspection and Enforcement Inspection Report which is enclosed with this letter. Within these areas, the inspection consisted of selective examinations of procedures and representative records, interviews with personnel, and observations by the inspector.

Within the scope of this inspection, no items of noncompliance were observed.

In accordance with Section 2.790 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations, a copy of this letter and the enclosed inspection report will be placed in the NRC's Public Document Room. If this report contains any information that you (or your contractor) believe to be proprietary, it is necessary that you make a written application within 20 days to this office to withhold such information from public disclosure. Any such application must be accompanied by an affidavit executed by the owner of the information, which identifies the document or part sought to be withheld, and which contains a statement of reasons which addresses with specificity the items which will be considered by the Commission as listed in subparagraph (b)(4) of Section 2.790. The information sought to be withheld shall be incorporated as far as possible into a separate part of the affidavit. If we do not hear from you in this regard within the specified period, the report will be placed in the Public Document Room.

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No reply to this letter is required; however, if you should have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

Eldon J. Brunner, Chief
Reactor Operations and Nuclear
Support Branch

Enclosure: Office of Inspection and Enforcement Inspection Report Number 50-320/78-33

cc w/encl:
T. Broughton, Safety & Licensing Manager
J. J. Barton, Project Manager
R. C. Arnold, Vice President - Generation
L. L. Lawyer, Manager - Generation Operations - Nuclear
G. P. Miller, Superintendent
J. L Seelinger, Unit 2 Superintendent - Technical Support
I. R. Finfrock, Jr.
Mr. R. Conrad
G. F. Trowbridge, Esquire
Miss Mary V. Southard, Chairman, Citizens for a Safe Environment (Without Report)

bcc w/encl:
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Nuclear Safety Information Center (NSIC)
Technical Information Center (TIC)
REG:I Reading Room
Region Directors (III, IV)(Report Only)
Commonwealth of Pennsylvania
Miss Mary V. Southard, Chairman, Citizens for a Safe Environment

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

Region I

AS OF 30 NOV 1978
NOTICE
REGION I HAS NOT OBTAINED PROPRIETARY CLEARANCE IN ACCORDANCE WITH 10 CFR 2790

Report No. 50-320/78-33

Docket No. 50-320

License No. DPR-73 Priority --

Licensee: Metropolitan Edison Company

P. O. Box 542

Reading, Pennsylvania 19603

Facility Name: Three Mile Island Nuclear Station, Unit 2

Inspection at: Middletown, Pennsylvania

Inspection conducted: November 7-9 and 16-17, 1978

Inspectors: D. R. Haverkamp, Reactor Inspector — 11/28/78 date signed

L. H. Bettenhausen, Reactor Inspector — 11/28/78 date signed

date signed

Approved by: R. R. Keimig, Chief, Reactor Projects Section No. I, RO&NS Branch — 11-29-78 date signed

Inspection Summary:

Inspection on November 7-9 and 16-17, 1978 (Report No. 50-320/78-33)

Areas Inspected: Routine, unannounced inspection by two regional based inspectors of startup test results; power level plateau data; and emergency safeguards actuation on November 7, 1978. The inspection involved 16 inspector-hours onsite by one NRC regional based inspector and one inspector-trainee, and 4 hours by one regional based NRC supervisor.

Results: No items of noncompliance were identified.

DUPLICATE DOCUMENT

Entire document previously entered into system under:

ANO 7901100030

No. of pages: 7pp.

Region I Form 12
(Rev. April 77)

DETAILS

1. Persons Contacted

Metropolitan Edison Company

Mr. J. Floyd, Unit 2 Supervisor of Operations
Mr. J. Hilbish, Station Lead Nuclear Engineer

General Public Utilities Service Corporation

Mr. C. Gatto, Lead Mechanical Test Engineer
Mr. S. Poje, Test Engineer
*Mr. R. Toole, Test Superintendent
Mr. J. Ullrich, Test Engineer

Babcock and Wilcox

Mr. J. Flint, Startup Test Engineer

USNRC

*Mr. D. L. Caphton, Chief, Nuclear Support Section No. 1

* denotes those present at the exit interview on November 17, 1978.

2. Startup Test Results Evaluation

The inspector conducted an evaluation of startup test TP 800/11 (MTX 147.21), Core Power Distribution (75% power level plateau testing completed on October 24, 1978)

The test records were evaluated to verify the following items.

-- Test changes had been approved in accordance with administrative procedures, properly entered into the procedure, accomplished if actions were necessary, and did not change the basic objective of the test.

-- Test deficiencies had been resolved, accepted by appropriate management, retest conducted if required, and any system or process changes necessitated have been properly documented and reviewed.

-- Test summaries and evaluations had been performed by the cognizant engineers, and test results had been compared with established acceptance criteria.

-- "As-run" copies of the test procedures contain completed data sheets (sample), data are recorded where required and are within acceptance tolerances (sample), test deficiencies noted receive appropriate review and evaluation, and individual test steps and data sheets have been properly initialed and dated.

-- Quality Assurance inspection records have been completed to document the adequacy of the test package contents, to indicate independent review of test records and data package contents, and an independent audit was performed during test performance, as required by administrative procedures.

-- Approval of the test results by those personnel charged with responsibility for review and acceptance has been documented, and if the off site review committee has audited the test package, that their comments are included and corrective action has been taken if required.

The inspector used one or more of the following acceptance criteria for the above items.

-- Final Safety Analysis Report

-- Technical Specifications

-- Test Instruction 7, GPU Startup Problem Report

-- Test Instruction 9, Conduct of Test

-- Test Instruction 13, Test Interface Instructions

-- Test Instruction 18, Test Procedure Documents

-- Regulatory Guides

-- Inspector Judgment

-- Quality Assurance Program

Findings were acceptable and no items of noncompliance were identified.

3. Power Level Plateau Data Review

a. Verification of Licensee Evaluation of Test Results

The inspector conducted a review of the following startup tests.

- -- TP 800/5 (MTX 147.19), Reactivity Coefficients at Power (75% power level plateau testing completed on October 25, 1978)
- -- TP 800/35 (MTX 147.36), Effluent and Effluent Monitoring System Test (75% power level plateau testing completed on October 23, 1978)
- -- TP 800/18 (MTX 147.27), Power Imbalance Detector Correlation Test (75% power level plateau testing completed on October 27, 1978)
- -- TP 800/2 (MTX 108.7), Nuclear Instrument Calibration at Power (75% power level plateau testing completed on October 27, 1978)
- -- TP 800/12 (MTX 147.22), Unit Load Steady State Test (75% power level plateau testing completed on October 23, 1978)
- -- TP 800/22 (MTX 147.30), NSS Heat Balance (75% power level plateau testing completed on October 23, 1978)

The test records were reviewed to verify the following items.

- -- Test summaries and evaluations had been performed by the congizant engineers, and test results had been compared with established acceptance criteria.
- -- Approval of the test results by those personnel charged with responsibility for review and acceptance has been documented, and if the off site committee has audited the test package, that their comments are included and corrective action has been taken if required.

The inspector used one or more of the following acceptance criteria for the above items.

-- Final Safety Analysis Report

-- Technical Specifications

-- Test Instruction 18, Test Procedure Documents

-- Regulatory Guides

-- Inspector Judgment

-- Quality Assurance Program

Findings were acceptable and no items of noncompliance were identified.

b. Authorization to Raise Power

The inspector reviewed the licensee's evaluation of the 75% plateau test results and the authorization for proceeding to the next test plateau. This review included discussions with licensee and startup group representatives and review of the following items.

-- Startup tests listed in Paragraphs 3 and 4.a

-- SP 800/21 (MTX 147.29), Unit Startup and Power Escalation Test (TWG approval received to escalate power to 100% on October 27, 1978)

The review was conducted to assure or confirm the following items.

-- All applicable testing has been completed.

-- All testing anomalies have been evaluated and resolved by the licensee.

-- The licensee has reviewed Technical Specification requirements applicable to the next higher power level and has fully implemented them.

-- The licensee performed core and plant surveys to assure safe operation during the increase of power level and arrival at the next plateau; including examination of flux distribution, core performance, reactor heat balance, unexpected radioactivity and radiation leakage, pressure boundary leakage, and reactor coolant chemistry.

-- The licensee has extrapolated the results of tests to applicable plateaus in the power ascension program, has compared the extrapolation with predicted plant performance, and has determined that it is reasonable and prudent to continue the testing program to the next planned power level plateau.

The inspector used one or more of the following acceptance criteria for the above items.

-- Final Safety Analysis Report

-- Technical Specifications

-- Test Instruction 9, Conduct of Test

-- Regulatory Guides

-- Inspector Judgment

Findings were acceptable. No items of noncompliance were identified.

4. Emergency Safeguards Actuation

The inspector reviewed the licensee's corrective measures concerning an emergency safeguards (ES) actuation which occurred on November 7, 1978. While operating at 92% rated thermal power (RTP), startup testing per TP 800/5, Reactivity Coefficients at Power, was in progress. All operating parameters were normal except RC Tave, which had been elevated to 588° F (6° F above normal) for temperature coefficient measurement. A heater drain tank low level alarm was received, which automatically tripped the operating heater drain pumps. The subsequent feedwater flow transient resulted in tripping a condensate booster pump on low suction pressure, which automatically tripped the 1B feedwater pump. The Integrated Control System began

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
631 PARK AVENUE
KING OF PRUSSIA, PENNSYLVANIA 19406

Reference 38

Docket Nos. 50-289
50-320

APR 20 1979

Metropolitan Edison Company
ATTN: Mr. J. G. Herbein
Vice President
P.O. Box 542
Reading, Pennsylvania 19640

Gentlemen:

Subject: Combined Inspections 50-289/79-08 and 50-320/79-07

This refers to the inspection conducted by Mr. D. Haverkamp of this office on March 19-23 and 26, 1979, at Three Mile Island Nuclear Station, Units 1 and 2, Middletown, Pennsylvania, of activities authorized by NRC License Nos. DPR-50 and DPR-73 and to the discussions of our findings held by Mr. Haverkamp with Messrs. J. Logan and J. Seelinger of your staff on March 23, 1979 and with Mr. Seelinger of your staff at the conclusion of the inspection.

Areas examined during this inspection are described in the Office of Inspection and Enforcement Inspection Report which is enclosed with this letter. Within these areas, the inspection consisted of selective examinations of procedures and representative records, interviews with personnel, measurements made by the inspector, and observations by the inspector.

Within the scope of this inspection, no items of noncompliance were observed.

In accordance with Section 2.790 of the NRC's "Rules of Practice", Part 2, Title 10, Code of Federal Regulations, a copy of this letter and the enclosed inspection report will be placed in the NRC's Public Document Room. If this report contains any information that you (or your contractor) believe to be proprietary, it is necessary that you make a written application within 20 days to this office to withhold such information from public disclosure. Any such application must be accompanied by an affidavit executed by the owner of the information, which identifies the document or part sought to be withheld, and which contains a statement of reasons which addresses with specificity the items which will be considered by the Commission as listed in subparagraph (b)(4) of Section 2.790. The information sought to be withheld shall be incorporated as far as possible into a separate part of the affidavit. If we do not hear from you in this regard within the specified period, the report will be placed in the Public Document Room.

Dupe of
7906200334

No reply to this letter is required; however, if you should have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

Eldon J. Brunner, Chief
Reactor Operations and Nuclear
Support Branch

Enclosure: Office of Inspection and Enforcement Combined Inspection Report Numbers 50-289/79-08 and 50-320/79-07

cc w/encl:
E. G. Wallace, Licensing Manager
J. J. Barton, Project Manager
R. C. Arnold, Vice President - Generation
L. L. Lawyer, Manager - Generating Operations
G. P. Miller, Manager - Generating Station - Nuclear
J. L. Seelinger, Unit 1 Superintendent
W. E. Potts, Unit 1 Superintendent - Technical Support
J. B. Logan, Unit 2 Superintendent
G. A. Kunder, Unit 2 Superintendent - Technical Support
I. R. Finfrock, Jr.
Mr. R. Conrad
G. F. Trowbridge, Esquire
Miss Mary V. Southard, Chairman, Citizens for a Safe Environment
(Without Report)

bcc w/encl:
IE Mail & Files (For Appropriate Distribution)
Central Files
Public Document Room (PDR)
Local Public Document Room (LPDR)
Nuclear Safety Information Center (NSIC)
Technical Information Center (TIC)
REG:I Reading Room
Director, Region IV (Report Only)
Commonwealth of Pennsylvania
Miss Mary V. Southard, Chairman, Citizens for a Safe
Environment

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

Region I

AS OF 20 APR 1979
NOTICE
REGION 1 HAS NOT OBTAINED PROPRIETARY
CLEARANCE IN ACCORDANCE WITH 10 CFR 2.790

Report No.	50-289/79-08 50-320/79-07				
Docket No.	50-289 50-320				
License No.	DPR-50 DPR-73	Priority	--	Category	C

Licensee: Metropolitan Edison Company

P.O. Box 542

Reading, Pennsylvania 19640

Facility Name: Three Mile Island Nuclear Station, Units 1 and 2

Inspection at: Middletown, Pennsylvania

Inspection conducted: March 19-23 and 26, 1979

Inspectors: D. R. Haverkamp — 4/17/79
D. R. Haverkamp, Reactor Inspector — date signed

date signed

date signed

Approved by: R. R. Keimig — 4/19/79
R. R. Keimig, Chief, Reactor Projects Section No. 1, RO&NS Branch — date signed

Dupe of
7906200339

Inspection Summary:

Inspection on March 19-23 and 26, 1979 (Combined Report Nos. 50-289/79-08 and 50-320/79-07)

Areas Inspected: Routine, unannounced inspection of previous inspection findings (Unit 1); selected licensee events (Units 1 and 2); facility tour (Unit 1); and licensee followup to a prompt reportable occurrence identified during the inspection (Unit 1). The inspection involved 27 hours onsite for Unit 1 and 17 hours onsite for Unit 2 by one NRC regional based inspector.

Results: No items of noncompliance were identified.

DUPLICATE DOCUMENT

Entire document previously entered into system under:

ANO 7906200339

No. of pages: 10pp.

Region I Form 12
(Rev. April 77)

DETAILS

1. Persons Contacted

Metropolitan Edison Company

Mr. T. Acker, Unit 1 Shift Foreman
Mr. R. Barley, Unit 1 Lead Mechanical Engineer
Mr. M. Benson, Station Nuclear Engineer
Mr. R. Bensel, Unit 2 Lead Electrical Engineer
Mr. M. Bezilla, Unit 2 PORC Secretary
Mr. J. Chwastyk, Shift Supervisor
Mr. R. Dubiel, Supervisor of Radiation Protection and Chemistry
Mr. C. Hartman, Unit 1 Lead Electrical Engineer
Mr. T. Hawkins, Unit 1 Maintenance Supervisor
* Mr. J. Logan, Unit 2 Superintendent
Mr. T. Mackey, Supervisor of Quality Control
Mr. L. Noll, Unit 1 Shift Foreman
Mr. V. Orlandi, Unit 1 Lead Instrumentation and Controls Engineer
Mr. D. Pilsitz, Unit 1 Shift Foreman
Mr. W. Potts, Unit 1 Superintendent - Technical Support
Mr. M. Ross, Unit 1 Supervisor of Operations
** Mr. J. Seelinger, Unit 1 Superintendent
Mr. M. Shatto, Unit 1 PORC Secretary
* Mr. R. Warren, Unit 2 Lead Mechanical Engineer

Other Personnel

Mr. T. Szymanski, Instructor, Career Management Branch, NRC Headquarters

The inspector also interviewed several other licensee employees during the inspection. They included control room operators, maintenance personnel, engineering staff personnel and general office personnel.

* denotes those present at the exit interview on March 23, 1979.

** present at the exit interviews on March 23 and 26, 1979.

2. Licensee Action on Previous Inspection Findings (Unit 1)

(Open) Unresolved Item 289/77-09-02: Adequacy of Snubber Visual Inspection Surveillance Procedure. Licensee review and approval of the proposed PCR to SP 1301-9.9 is scheduled for completion by May 1, 1979. A special tool has been manufactured to measure snubber piston positions for sufficient stroke to allow for thermal growth without hitting the mechanical stops. This item remains unresolved pending revision of SP 1301-9.9.

(Open) Unresolved Item 289/78-17-01: Licensee Review of IE Circular 78-06 and IEC 78-07. Licensee review of these circulars for applicability and determination of appropriate action has been completed. With respect to IEC 78-07, "Damaged Components on a Bergen-Patterson Series 25000 Hydraulic Test Stand," applicable test stand inspection requirements have been incorporated in SP 1303-9.9. With respect to IEC 78-06, "Potential Common Mode Flooding of ECCS Rooms," a periodic preventive maintenance (PM) inspection is planned for back flow check valves located in safeguards equipment vaults drain lines. This item remains unresolved pending preparation and approval of the PM procedure, scheduled for completion by May 15, 1979.

(Closed) Unresolved Item 289/78-14-01: Adequacy of Alarm Circuits to Monitor Operability of the Reactor Building Access Hatch Interlocks. New limit switches were installed during the current refueling outage, as documented by Work Request #24246 completed March 14, 1979. The limit switches were located to provide proper monitoring of Reactor Building personnel and equipment hatch door interlocks. The inspector had no further questions concerning this item.

(Closed) Noncompliance 289/78-19-01: Administrative Controls for Operating and Surveillance Procedures. The licensee's specific corrective actions were completed as described in MEC letter to NRC:Region I Serial GQL 2071, dated December 29, 1978. The general corrective action included a complete audit by the Operations Engineer of the Control Room file of operating procedures. Additional discrepancies were identified during that audit concerning nonconformance with administrative procedural controls and were corrected by initiating about 35 procedure change requests. Selected operating procedures were reviewed by the inspector and were determined to contain appropriate revisions. The inspector had no further questions concerning this item.

(Closed) Unresolved Item 289/78-19-04: LER 78-27 Corrective Actions. Change/Modification 1165 was approved to replace the core flood tank level transmitters with those of a different design. Work associated with C/M 1165 was performed under Work Request #25057 during the current refueling outage. C/M 1165 has been fully completed with the exception of final drawing revisions. The inspector had no further questions concerning this item.

(Closed) Unresolved Item 289/78-20-01: SP 1302-5.13 Discrepancies. SP 1302-5.13 has been superseded in its entirety by TCN's 79-40 and 79-46. The previous comments concerning SP 1302-5.13 were no longer applicable. The inspector had no further questions concerning this item.

(Open) Unresolved Item 289/78-20-03: SP 1302-6 Discrepancies. Surveillance Procedure 1303-5.5, Revision 7, dated January 30, 1979, correctly identified six D/P instruments, used to perform surveillance of the Control Room Emergency Filters. SP 1302-6, "Calibration of Non Tech Spec Instruments Used for Tech Spec Compliance," Revision 1, included calibration requirements for four of the D/P instruments (DPI-698, -699, -700 and -701), but did not list calibration requirements for DPI-695 and DPI-696, due to an apparent oversight. The referenced calibration procedure for the four listed filter D/P instruments, IC-76, provided for a multipoint check of the D/P indicators. (The inspector determined that all six D/P instruments had in fact recently been calibrated per IC-76). SP 1302-6, Revision 1, also listed calibration requirements for fire protection instrumentation used to comply with Tech Spec requirements.

The Unit 1 Lead Instrumentation and Controls Engineer stated that SP 1302-6 would be further revised to include calibration requirements for DPI-695 and DPI-696. In addition, the method of scheduling (by computer printout) and documenting completion of SP 1302-6 calibration requirements would be reviewed. This item remains unresolved pending completion of these additional actions.

(Closed) Unresolved Item 289/78-20-04: Gage Calibration Scheduling. Decay Heat Pump Flow Instruments DH-1-FI-1 and DH-1-FI-2, Diesel Generators 1A and 1B Megawatt and Volt Meters and Control Room Emergency Ventilation Filter D/P Indicators were satisfactorily calibrated in January, 1979. The inspector had no further questions concerning this item.

(Closed) Unresolved Item 289/78-20-05: Thermocouple Calibrations. SP 1302-14.1, Revision 5, dated March 1, 1979 incorporated changes which resolved the referenced concerns. The inspector had no further questions concerning this item.

3. In-Office Review of Licensee Event Reports (LERs) (Units 1 and 2)

The LERs listed below were reviewed in the Region I office promptly following receipt to verify that details of the event were clearly reported including the accuracy of the description of cause and the adequacy of corrective action. The LERs were also reviewed to determine whether further information was required from the licensee, whether generic implications were involved, whether the event should be classified as an Abnormal Occurrence, whether the information involved with the event should be submitted to Licensing Boards, and whether the event warranted onsite followup.

The following Unit 1 LERs were reviewed.

* -- LER 79-03/3L, dated March 9, 1979 (High Pressure Injection Pump MU-P-1C tripped on overload during surveillance testing, due to a failed lead that connects sections of the motor internal windings).

* -- LER 79-04/3L, dated March 14, 1979 (Emergency Diesel EG-Y-1B tripped on overspeed during surveillance testing, due to misadjusted linkage following governor replacement).

** -- Nonroutine 10 Day Environmental Report, dated February 26, 1979 (Measured level of tritium in river water at stations 9A2 and 9B1 exceeded ten times the control station value, due to location and sampling methods).

The following Unit 2 LERs were reviewed.

** -- NPDES Noncompliance Notification 78-26, dated January 3, 1979 (IWFS discharge pH of 9.1 exceeded permit limitations which allows a pH range of 6.0-9.0).

-- LER 78-73/3L, dated January 15, 1979 (Containment atmosphere particulate radioactivity monitor air pump for HP-R-227 was seized, due to accumulation of water in the sample lines).

* -- LER 78-74/3L, dated January 23, 1979 (Diesel Generator DF-X-1B did not start during surveillance testing, apparently due to partially clogged fuel oil filter).

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* denotes those LERs selected for onsite followup.

** denotes those environmental reports subject to generic and selective onsite followup during a subsequent environmental inspection.

* -- LER 79-01/3L, dated February 1, 1979 (RB Pressure Hi-Hi Channel A monthly functional test was not performed when scheduled, due to technician error).

* -- LER 79-02/3L, dated January 23, 1979 (Adequate documentation was not retained to verify T.S. 3.3.1 surveillance performance, due to personnel error).

-- LER 79-03/3L, dated February 2, 1979 (Quadrant power tilt steady state and transient limits were exceeded when Control Rod #6-12 dropped into the core, due to a blown fuse in the B phase).

* -- LER 79-04/3L, dated February 2, 1979 (Valve BS-V-1B position indication was inoperable due to a bent valve stem).

* -- LER 79-05/3L, dated February 2, 1979 (Small crack in decay heat piping weld due to vibration).

* -- LER 79-06/3L, dated January 31, 1979 (Borated water source - BWST - boron concentration surveillance was not performed when scheduled, due to personnel error).

* -- LER 79-07/3L, dated February 26, 1979 (Travelling Water Screens were inoperable in Mode 5, due to significant build-up of debris causing a high differential level across the idle screen system).

-- LER 79-08/3L, dated February 9, 1979 (Setpoints of two feedwater line rupture detection pressure switches were outside allowable limits due to instrument drift or steam leakage).

* -- LER 79-09/3L, dated February 26, 1979 (Boration system flow path verification surveillance was not performed in Mode 5 after the makeup pumps were tagged out, due to inadequate procedure).

* -- LER 79-10/1T, dated February 26, 1979 (Boron concentration for boric acid mix tank was in excess of the T.S. limit, and appropriate corrective action was not taken due to personnel error).

* denotes those LERs selected for onsite followup.

The above LERs were closed based on satisfactory review in the Region I office, except those selected for onsite followup.

4. Onsite Licensee Event Followup (Units 1 and 2)

For those LERs selected for onsite followup (denoted in Paragraph 3), the inspector verified that the reporting requirements of Technical Specifications and GP 4703 (Original) had been met, that appropriate corrective action has been taken, that the event was reviewed by the licensee as required by Technical Specifications, and that continued operation of the facility was conducted in conformance with Technical Specification limits.

The inspector's findings regarding these licensee events were acceptable, unless otherwise noted below.

-- Unit 2 LER 78-74/3L described the failure of Diesel Generator DF-X-1B to start during surveillance testing. The event cause was attributed to be a partially clogged fuel oil filter, although the cause could not be positively determined. The corrective actions included changing the fuel oil filters, changing the air intake filter, and draining and refilling the fuel oil day tank. The LER did not fully describe the corrective actions taken. This LER will remain open pending additional review of corrective and preventive actions.

-- Unit 2 LER 79-04/3L described the inoperability of Valve BS-V-1B due to a bent valve stem. The valve was temporarily repaired and returned to service by installing a spacer between the valve and the operator. Permanent repair is scheduled under Work Request C-0647 and Change/Modification 2-0400, as tracked by PORC Action Item 2-79-010. The permanent repair will include removal of the temporary spacer and replacement of the stem with a stem of improved material. The inspector determined that BS-V-1B was an eight-inch Aloyco manufactured valve, and there are about 18 Aloyco valves of different sizes used in safety-related applications at the facility. Licensee representatives stated that the need to replace the stems of other Aloyco valves with improved stems, as a precautionary measure, would be evaluated. This item is unresolved pending permanent repair of BS-V-1B, licensee evaluation of the need for additional generic corrective action, and submission of an Update LER. (320/79-07-01)

-- Unit 2 LER 79-05/3L described a small crack that had developed in a piping weld upstream of the B Decay Heat Pump discharge relief valve. The crack was in the heat affected zone of the weld and was attributed to vibration. The AE is evaluating if additional pipe hangers are required to reduce vibration, as tracked by PAI 2-79-011. This item is unresolved pending completion of the AE's vibration evaluation, final PORC disposition of long term corrective action and submission of an Update LER. (320/79-07-02)

-- Unit 2 LER 79-10/1T described the out-of-specification condition of the boric acid mix tank and subsequent facility operation in violation of Technical Specification 3.1.2.9 requirements. The inspector determined that appropriate immediate and long term corrective actions were taken, but not adequately described in the LER. The report failed to identify the cause of high boron concentration and corrective action to restore the concentration to within specification. Additionally, the basis for the conclusion that the event did not adversely affect health and safety was insufficiently described. This item is unresolved pending submission of an Update LER that fully describes the event, cause and corrective actions. (320/79-07-03)

5. <u>BWST Dome Damage (Unit 1)</u>

On March 19, 1979, the Unit 1 Borated Water Storage Tank (BWST) dome was observed to be partially collapsed. The center section of the dome had collapsed about 2-3 feet. The plant was in cold shutdown for a scheduled refueling outage at the time of discovery of the BWST damage. This event was determined to be prompt reportable by plant management on March 22, 1979, and the inspector was informed of the event description, apparent cause and planned corrective action. Details of the event will be reported to Region I in the 14-day LER.

The inspector reviewed C/M 1309 (Work Request 0784) dated March 24, 1979, which requested modification or replacement of the 24-inch manway cover on top of the BWST with a venting device. The modification was considered necessary to ensure that no significant vacuum is created when drawing down water from the tank. The inspector also reviewed MEC letter GEM 1607 dated March 23, 1979, "Structural and Functional Adequacy of BWST," MEC letter GEM 1615 dated March 23, 1979, "BWST Atmospheric Vent," and other correspondence and documentation related to C/M 1309. Additionally, the inspector observed work in progress on March 26, 1979, to modify the manway cover for continuous venting. The inspector noted that the licensee's corrective actions concerning the BWST dome damage appeared acceptable and had no further questions concerning this matter at this time.

6. <u>In-Office Review of Special Reports (Unit 2)</u>

The special reports listed below were reviewed in the Region I office to verify that the report included information required to be reported and that test results and/or supporting information discussed in the report were consistent with design predictions and performance specifications, as applicable. The reports were also reviewed to ascertain whether planned corrective action was adequate for resolution of identified problems, where applicable, and to determine whether any information contained in the report should be classified as an Abnormal Occurrence.

The following TMI-2 special reports were reviewed.

-- LER 78-65/99X dated January 30, 1979 (ECCS actuation which occurred on November 7, 1978).

-- LER 78-69/99X dated February 28, 1979 (ECCS actuation which occurred on December 2, 1978).

The above reports were closed based on satisfactory review at the Region I office and previous review of the events during prior inspections.

7. <u>Plant Tour (Unit 1)</u>

At various times during the inspection, the inspector conducted tours of the Unit 1 auxiliary building, turbine building, and reactor building. The tours were conducted to observe general housekeeping and cleanliness conditions and the readiness of systems/equipment for plant startup. Findings were acceptable.

8. Unresolved Items (Unit 2)

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 this inspection are discussed in Paragraph 4.

9. Exit Interviews

The inspector met with the licensee representatives (denoted in Paragraph 1) at the conclusion of the inspection on March 23 and 26, 1979. The inspector summarized the purpose and scope of the inspection and the findings.