Docket No. 50-289

RECEIVED APR 08 1981 U.S. NUCLEAR REGULATORY COMMISSION

Mr. Henry D. Hukill, Vice President
and Director - TMI-1
Metropolitan Edison Company
P. O. Box 480
Middletown, Pennsylvania 17057

Dear Mr. Hukill:

The Commission has issued the enclosed Amendment No. to Facility Operating License No. DPR-50 for the Three Mile Island Unit No. 1. This amendment consists of changes to the Technical Specifications in response to your application transmitted by letters dated September 17, 1975, October 29, 1975, February 18, 1977 and May 13, 1980. The amendment makes changes to the Technical Specifications related to primary containment leakage testing.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. to DPR-50
2. Safety Evaluation
3. Notice

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| ~~R. Tedesco~~ | E. Blackwood | |

No legal objection to notice & amendment SER not reviewed

WB

P 8104130773 JCD

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|---|---|---|---|---|---|---|---|
| OFFICE | LA-ORB#4:DL | ORB#4:DL | C-ORB#4:DL | AD:OR:DL | OELD | CSB:DSI | |
| SURNAME | [illegible] | DDiIanni:cf | RWReid | TMNovak | Cutchin | W. Butler | |
| DATE | 1/8/81 | 1/9/81 | 1/9/81 | 3/?/81 | 3/6/81 | 2/26/81 | |

MARCH 3 0 1981

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P. O. Box 480
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Sincerely,

Original signed by

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

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| JWetmore | |
| ACRS-10 | |

*SEE PREVIOUS WHITE FOR CONCURRENCES

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| OFFICE | ORB#4:DL | ORB#4:DL | C-ORB#4:DL | AD:OR:DL | OELD | CSB:DSI | |
| SURNAME | *EHylton | *DDiIanni;cf | JStolz | *TNovak | *MCutchin | *WButler | |
| DATE | ~~1/8/81~~ 3/26/81 | 1/9/81 | 3/26/81 3/30 | 3/4/81 | 3/6/81 | 2/26/81 | |

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Docket No. 50-289

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and Director - TMI-1
Metropolitan Edison Company
P. O. Box 480
Middletown, Pennsylvania 17057

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The amendment makes changes to the Technical Specifications related to primary containment leakage testing. In connection with this action, the Commission has granted an exemption which allows the licensee to verify the integrity of airlock door resilient seals following each use by pressurizing the seals to a pressure of Pa. This is an exemption from the portion of Paragraph III.D.2 of Appendix J to 10 CFR 50 which states: "However, airlocks which are opened during such intervals, shall be tested after each opening." The exemption also permits the Type B test pressure for the airlock door seals to be 10 psig rather than at Pa (50.6 psig) the peak calculated accident pressure as required by Paragraph III.B.2 of Appendix J to 10 CFR 50.

We have reviewed the proposed airlock testing and associated acceptance criteria and we find that it adequately demonstrates the leakage integrity of the airlock. We find that granting the proposed exemption from the requirements of Appendix J is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

~~8104150~~

C-CSB:DSI
WButler
12/ /80

Enclosures & cc: See next pg.

| OFFICE | LA-ORB#4:DL | ORB#4:DL | C-ORB#4:DL | AD-OR:DL | D:DL | OELD |
|---|---|---|---|---|---|---|
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| DATE | 12/12/80 | 12/15/80 | 12/16/80 | 11/ /80 | ~~11/ /80~~ | 11/ /80 |

Mr. R. C. Arnold



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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

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Docket File
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EHylton

March 31, 1981

Docket No. 50-289

Docketing and Service Section
Office of the Secretary of the Commission

SUBJECT: THREE MILE ISLAND UNIT NO. 1

Two signed originals of the Federal Register Notice identified below are enclosed for your transmittal to the Office of the Federal Register for publication. Additional conformed copies (12 ) of the Notice are enclosed for your use.

- ☐ Notice of Receipt of Application for Construction Permit(s) and Operating License(s).
- ☐ Notice of Receipt of Partial Application for Construction Permit(s) and Facility License(s): Time for Submission of Views on Antitrust Matters.
- ☐ Notice of Availability of Applicant's Environmental Report.
- ☐ Notice of Proposed Issuance of Amendment to Facility Operating License.
- ☐ Notice of Receipt of Application for Facility License(s); Notice of Availability of Applicant's Environmental Report; and Notice of Consideration of Issuance of Facility License(s) and Notice of Opportunity for Hearing.
- ☐ Notice of Availability of NRC Draft/Final Environmental Statement.
- ☐ Notice of Limited Work Authorization.
- ☐ Notice of Availability of Safety Evaluation Report.
- ☐ Notice of Issuance of Construction Permit(s).
- ☐ Notice of Issuance of Facility Operating License(s) or Amendment(s).
- XX Other: Amendment No. 63
  Referenced documents have been provided PDR.

Division of Licensing, ORB#4
Office of Nuclear Reactor Regulation

Enclosure:
As Stated

| | | | | | | |
|---|---|---|---|---|---|---|
| OFFICE → | ORB#4:DL | | | | | |
| SURNAME → | EHylton:cf | | | | | |
| DATE → | 3/31/81 | | | | | |

NRC FORM 102 (1-76)

Docket

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 30, 1981

Docket No. 50-289

Mr. Henry D. Hukill, Vice President
and Director - TMI-1
Metropolitan Edison Company
P. O. Box 480
Middletown, Pennsylvania 17057

Dear Mr. Hukill:

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Sincerely,

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

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2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

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U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

* Atomic Safety and Licensing Board Panel
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

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General Counsel
Federal Emergency Management Agency
ATTN: Docket Clerk
1725 I Street, NW
Washington, DC 20472

cc w/enclosure(s) & incoming dtd.:
9/17/75, 10/29/75, 2/18/77 & 5/13/80

Governor's Office of State Planning
and Development
ATTN: Coordinator, Pennsylvania
State Clearinghouse
P. O. Box 1323
Harrisburg, Pennsylvania 17120

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

METROPOLITAN EDISON COMPANY

JERSEY CENTRAL POWER AND LIGHT COMPANY

PENNSYLVANIA ELECTRIC COMPANY

DOCKET NO. 50-289

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 63
License No. DPR-50

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Metropolitan Edison Company, Jersey Central Power and Light Company and Pennsylvania Electric Company (the licensee), dated September 17, 1975 and revised by letters dated October 29, 1975, February 18, 1977 and May 13, 1980 and staff discussions, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (1) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-50 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 63, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 30, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 63

FACILITY OPERATING LICENSE NO. DPR-50

DOCKET NO. 50-289

Revise Appendix A as follows:

| Remove Pages | Insert Pages |
|---|---|
| 4-29 | 4-29 |
| 4-31 thru 4-34a | 4-31 thru 4-34a |

The changed areas on the revised pages are shown by marginal lines.

## 4.4 REACTOR BUILDING

### 4.4.1 CONTAINMENT LEAKAGE TESTS

Applicability

Applies to containment leakage.

Objective

To verify that leakage from the reactor building is maintained within allowable limits.

Specification

4.4.1.1 Integrated Leakage Rate Tests

4.4.1.1.1 Design Pressure Leakage Rate

The design integrated leakage rate, ($L_d$), from the reactor building at the 55 psig design pressure, $P_d$, is .1 weight percent of the building atmosphere at that pressure per 24 hours.

4.4.1.1.2 Allowable Integrated Leakage Rate

The maximum allowable integrated leakage rate, ($L_a$), from the reactor building at the calculated peak reactor building internal pressure of 50.6 psig ($P_a$) associated with the design basis accident, shall not exceed .1 weight percent of the building atmosphere at that pressure per 24 hours.

4.4.1.1.3 Testing at Reduced Pressure

The governing criteria for the periodic integrated leakage rate tests to be performed at the reduced test pressure, $P_t$ (at 30 psig), is the maximum allowable containment test leakage rate, $L_t$. $L_t$ is equal to 0.077 weight percent of the building atmosphere per 24 hrs.

4.4.1.1.5 Frequency of Test

After the initial pre-operational leakage rate test, two integrated leakage rate tests shall be performed at approximately equal intervals between each major shutdown for inservice inspection to be performed at 10 year intervals. In addition, an integrated test shall be performed at each 10 year interval, coinciding with the inservice inspection shutdown. The test shall coincide with a shutdown for major fuel reloading.

4.4.1.1.6 Acceptance Criteria

a. Initial and periodic integrated leakage rate test at $P_t$.

   $L_{tm}$ shall be less than .75 $L_t$.

b. If the initial and periodic integrated leakage rate test fails to meet the acceptance criteria of 4.4.1.1.6a, the test schedule applicable to subsequent tests shall be subject to review and approval by the Commission.

c. If two consecutive periodic integrated leakage rate tests fail to meet the acceptance criteria of 4.4.1.1.6a, a test shall be performed at each plant shutdown for refueling or every 18 months, whichever occurs first, until two consecutive tests meet the criteria of 4.4.1.1.6a.

4.4.1.1.7 Corrective Action and Retest

If, during an integrated or supplemental leak rate test, potentially excessive leakage paths are identified which would result in the integrated leak test not meeting the acceptance criteria:

a. terminate the integrated or supplemental leak rate test,

b. measure the subject leakage using local leakage testing methods,

c. make repairs and/or adjustment,

d. run an integrated leakage rate test.

If the test data from a completed leakage rate test does not meet the acceptance criteria, the integrated leakage rate test need not be repeated provided local leakage rate measurements are made at pressure $P_t$ before and after repair to demonstrate that the leakage rate reduction achieved by the repairs reduces the overall measured integrated leakage rate to an acceptable value.

4.4.1.1.8 Report of Test Results

Each integrated leak rate test will be the subject of a summary technical report which will include a description of test methods used and a summary of local leak detection tests. Sufficient data and analysis shall be included to show that a stabilized leak rate was attained and to identify all significant required correction factors such as those associated with humidity and barometric pressure, and all significant errors such as those associated with instrumentation sensitivities and data scatter. This report shall be titled Reactor Containment Building Integrated Leak Rate Test and shall be submitted to the AEC within 3 months of the test.

4.4.1.2 Local Leakage Rate Tests

4.4.1.2.1 Scope of Testing

a. The local leak rate shall be measured for the following components using a type "B" test as defined in 10CFR50, Appendix J.

1. Personnel air lock door gaskets

2. Emergency air lock door gaskets

3. The resilient seals on the equipment hatch and fuel transfer tube blind flanges

4. Reactor Building Purge valves (AH-V1A, B, C, and D)

5. Blind flanges on both ends of pipe through the following penetrations:

   S.1 No. 104 (S/G drains)

   S.2 No. 105 (S/G cleaning)

   S.3 No. 106 (S/G cleaning)

   S.4 No. 210 (S/G annulus drains)

   S.5 No. 211 (S/G annulus drains)

b. The local leak rate shall be measured for the following isolation valves using a type "C" test as defined in 10CFR50, Appendix J.

1. Containment air sample (CM-V1, 2, 3, and 4)

2. Hydrogen purge discharge system (HP-V1 and V6)

3. Make-up and Purification (MU-V18, MU-V20, MU-V116, MU-V2A/B, MU-V25)

4. Industrial Cooler System (RB-V2* and RB-V7)

5. Core Flood (CF-V12A and B, CF-V19A and B, CF-V20A and B)

6. Nuclear Service Closed Cooling (NS-V35)

7. Intermediate Closed Cooling (IC-V2)

8. Sample Valves (CA-V1, CA-V3, CA-V4A/B, CA-V13)

9. Drain Valves (WDG-V3, WDL-V303 and WDL-V534)

10. Chemical Addition (CA-192)

11. Nitrogen Supply (NI-V27)

12. Decay Heat Removal (DH-V69 & V64)

c. The following isolation valves will be tested by testing the Fluid Block System.

1. Nuclear Service Closed Cooling Water (NS-V4 and NS-V15)

2. Intermediate Cooling Water (IC-V3, V4 and V6)

3. Spent Fuel Cooling (SF-V23)

4. Make-up and Purification (MU-V3 and MU-V26)

5. Reclaimed Water (CA-V189)

6. Sample Valves (CA-V5A&B and CA-V2)

7. Drain Valves (WDL-V304, WDG-V4 and WDL-V535)

d. The following isolation valves or blank flanges will be tested by testing the Penetrative Pressurization System.

1. Instrument Air (IA-V6 and IA-V20)

2. Service Air (SA-V2 and SA-V3)

3. Leak rate system (LR-V1, 2, 3, 4, 5, 6, 10, and 49) Blank flanges on Penetrations 414, 415, 416

4. Incore Inst. Transfer Tube - Blank flange on Penetration 241

4.4.1.2.2 Conduct of Tests

a. Local leak rate tests shall be performed pneumatically at a pressure of not less than $P_a$, with the following exception: The access hatch door seal test shall normally be performed at 10 psig and the test every six months specified in 4.4.1.2.5.b shall be performed at a pressure not less than $P_a$.

b. Acceptable methods of testing are halogen gas detection, pressure decay, pneumatic flow measurement or equivalent.

c. The pressure for a valve test shall be applied in the same direction as that when the valve would be required to perform its safety function unless it can be determined that the direction will provide equivalent or more conservative results.

d. Valves to be tested shall be closed by normal operation and without any preliminary exercising or adjustments.

4.4.1.2.3 Acceptance Criteria

The combined leakage from all items listed in 4.4.1.2.1, except leakage from those valves or devices sealed by the Fluid Block System or Penetration Pressurization System, shall not exceed .6 $L_a$ (the maximum allowable leakage rate at $P_a$).

4.4.1.2.4 Corrective Action and Retest

a. If at any time it is determined that the criterion of 4.4.1.2.3 above is exceeded, repairs shall be initiated immediately.

b. If conformance to the criterion of 4.4.1.2.3 is not demonstated within 48 hours following detection of excessive local leakage, the reactor shall be shutdown and depressurized until repairs are effected and the local leakage meets the acceptance criterion as demonstrated by retest.

4.4.1.2.5 Test Frequency

Local leak detection tests shall be performed at a frequency of at least each refueling period, except that:

a. The equipment hatch and fuel transfer tube seals shall be tested every other refueling period but in no case at intervals greater than 3 years. If they are opened they will be tested after being closed.

b. The entire personnel and emergency airlocks shall be tested once every six months. When the airlocks are opened during the interim between six month tests, the airlock door resilient seals shall be tested within 72 hours of the first of each of a series of openings. This requirement exists whenever containment integrity is required.

c. The reactor building purge isolation valves shall be tested yearly.

d. Readings of the rotameters in each manifold of the penetration pressurization system shall be recorded at periodic intervals not to exceed three months.

4.4.1.3 Isolation Valve Functional Tests

Every three months, remotely operated reactor building isolation valves shall be stroked to the position required to fulfill their safety function unless such operation is not practical during plant operation. The valves not stroked every three months shall be stroked during each refueling period.

4.4.1.4 Annual Inspection

A visual examination of the accessible interior and exterior surfaces of the containment structure and its components shall be performed annually and prior to any integrated leak test to uncover any evidence of deterioration which may affect either the containment's structural integrity or leak-tightness. The discovery of any significant deterioration shall be accompanied by corrective actions in accord with acceptable procedures, nondestructive tests, and inspections, and local testing where practical, prior to the conduct of any integrated leak test. Such repairs shall be reported as part of the test results.

4.4.1.5 Reactor Building Modifications

Any major modification or replacement of components affecting the reactor building integrity shall be followed by either an integrated leak rate test or a local leak test, as appropriate, and shall meet the acceptance criteria of 4.4.1.1.5 and 4.4.1.2.3, respectively.

4.4.1.6 Operability of Access Hatch Interlocks

1. At least once per six months the operability of the personnel and emergency hatch door interlocks and the associated control room annunciator circuits shall be determined. If the interlock permits both doors to be open at the same time or does not provide accurate status indication in the control room, the interlock shall be declared inoperable.

2. During periods when containment integrity is required and an interlock in inoperable, each entry and exit via that airlock shall be locally supervised by a member of the unit operating maintenance or technical staffs, to assure that only one door is open at any time and that both doors are properly closed following use. A record of supervision and verification of closure shall be maintained during periods of interlock inoperability in an appropriate station log.

3. If an interlock is inoperable for more than 14 days following determination of inoperability, use of the airlock, except for emergency purposes, shall be suspended until the interlock is returned to operable status.

**Bases**[1]

The reactor building is designed for an internal pressure of 55 psig and a steam-air mixture temperature of 281 F. Prior to initial operation, the containment was strength tested at 115 percent of design pressure and leak rate tested at the design pressure. The containment was also leak tested prior to initial operation at approximately 50 percent of the design pressure. These tests established the acceptance criteria of 4.4.1.1.3.

The performance of periodic integrated and local leakage rate tests during the plant life provides a current assessment of potential leakage from the containment in case of an accident that would pressurize the interior of the containment. In order to provide a realistic appraisal of the integrity of the containment under accident conditions "as found" local leakage results must be documented for correction of the integrated leakage rate test results. Containment isolation valves are to be closed in the normal manner prior to local or integrated leakage rate tests.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

# RELATED TO AMENDMENT NO. 63 TO FACILITY OPERATING LICENSE NO. DPR-50

# METROPOLITAN EDISON COMPANY

# JERSEY CENTRAL POWER AND LIGHT COMPANY

# PENNSYLVANIA ELECTRIC COMPANY

# THREE MILE ISLAND NUCLEAR STATION UNIT NO. 1

# DOCKET NO. 50-289

## Introduction

On August 7, 1975 (Reference 1), the NRC requested Metropolitan Edison Company (Met-Ed) to review its containment leakage testing program for Three Mile Island, Unit 1 (TMI-1) and the associated Technical Specifications (TSs), for compliance with the requirements of Appendix J to 10 CFR Part 50.

Appendix J to 10 CFR Part 50 was published on February 14, 1973. Since by this date there were already many operating nuclear plants and a number more in advanced stages of design or construction, the NRC decided to have these plants re-evaluated against the requirements of this new regulation. Therefore, beginning in August 1975, requests for review of the extent of compliance with the requirements of Appendix J were made to each licensee. Following the initial responses to these requests, NRC staff positions were developed which would assure that the objectives of the testing requirements of the above cited regulation were satisfied. These staff positions have since been applied in our review of the submittals filed by the TMI-1 licensee. The results of our evaluation are provided below.

## Evaluation

Our consultant, the Franklin Research Center, has reviewed the licensee's submittals (References 2, 3, 4, and 5) and prepared the attached evaluation of containment tests for TMI, Unit 1 (Reference 6). We have reviewed this evaluation and concur in its bases and findings.

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In its report, the staff's consultant recommended that proposed TS 4.4.1.1.3 be modified in accordance with Reference (5) to include the corrected value of Pt (30 psig) instead of 27.5 psig for the containment integrated leakage rate test as follows:

> 4.4.1.1.3, Testing at Reduced Pressure
>
> The governing criteria for the periodic integrated leakage rate tests to be performed at the reduced test pressure, Pt (at 30 psig), is the maximum allowable containment test leakage rate, Lt. Lt is equal to 0.077 weight percent of the building atmosphere per 24 hours.

The basis for this change is that the Lt value of .077 corresponds to the reduced 30 psig test pressure that will be used for any future test. This change proposed by our consultant was discussed with and agreed to by the licensee.

The staff's consultant recommended that proposed T.S. 4.4.1.2.1.b be modified to include valves CA-V192, NI-V26, NI-V27, MU-V116, CF-V12A and B, DH-V69 and DH-V64 in the local leak rate testing program for isolation valves. In Reference (5), the licensee stated that valves MU-V116, CF-V12A and B, DH-V69 and DH-V64 are included in the local leak testing program. With regard to valves CA-V192, NI-V26 and NI-V27, the staff's consultant took exception to the licensee's interpretation of Appendix J in Reference (2) to exclude the valves from testing. We have discussed this matter with the licensee and he has agreed to the modification of T.S. 4.4.1.2.1.b that includes valves CA-V192, NI-V27 in the local leak rate testing program. The licensee intends to cap the line containing valves NI-V26 and NI-V27 inside of containment in place of leak testing valve NI-V26. We conclude that capping the line inside containment and testing valve NI-V27 outside containment meet the guidelines for local leak testing and meets the intent of Appendix J's requirements. Therefore, the need for not testing valve NI-V26 is acceptable.

By letter dated October 29, 1975 (Reference 3) the licensee requested a change to TS 4.4.1.2.5.b which, at the time the request was made, would have required an exemption to the requirements of Appendix J to 10 CFR 50 if the change was found acceptable. However, the portion of Appendix J pertaining to containment building airlock leak testing has been revised (effective October 22, 1980), and an exemption to the requirements of Appendix J is no longer necessary in regard to the licensee's request. Presently, Specification 4.4.1.2.5.b requires the licensee to test the resilient seal of the personnel and emergency air locks' outer door after each use but no more than once daily. The proposed change to Specification 4.4.1.2.5.b as shown in Referency (3) would permit the airlock door resilient seals to be tested within 72 hours of the first of a series of openings. Based on plant operating experience, requiring an airlock to be leak tested after each opening is impractical when frequent airlock usage is necessary over a short period of time. Furthermore, the TMI-1 airlock design incorporates dual seals on the airlock doors with the capability to pressurize the volume between the seals. Therefore, the applicant proposes to leak test the airlock door seals within 72 hours after opening an airlock. This will permit door seal integrity to be demonstrated without pressurizing the entire airlock. This is an acceptable test method for these tests.

A second part of the change request addresses the test pressure applied during testing the airlock door resilient seals within 72 hours of the first of a series of openings. TS 4.4.1.2.2a currently requires the licensee to apply a test pressure Pa equal to 50.6 psig when testing airlock door seals after each use. By letter dated October 29, 1975 (Reference 3) the licensee requested a change to TS 4.4.1.2.2a to reflect the reduced test pressure of 10 psig. The change will permit the licensee to apply a reduced test pressure of 10 psig when performing the 72 hour test interval. The lower test pressure of 10 psig is sufficient to verify that door seal integrity is being maintained and that the door seals are free of dirt and foreign objects. The test pressure is recommended by the air lock manufacturer, and testing at lower pressure is expected to extend the seal life. This change is also in conformance with the revised Appendix J.

The staff's consultants have found the reduced test pressure and 72 hour test interval requirement acceptable, however, the wording in TS 4.4.1.2.5b does not agree with standard specification wording. Applying the standardized wording, TS 4.4.1.2.5b would be modified to read as follows:

> The entire personnel and emergency airlocks shall be tested once every six months. When the airlocks are opened during the interim between six month tests, the airlock door resilient seals shall be tested within 72 hours of the first of each of a series of openings. This requirement exists whenever containment integrity is required.

This modified statement being different from the initial proposed statement by the licensee (note Reference 3) has been discussed with and agreed to by the licensee. We have reviewed our consultant's evaluation and the basis (Reference 6) for the airlock test changes. Based on the above, we conclude that the proposed changes in the testing frequency and lower test pressure for the airlock door resilient seals are acceptable.

Our consultant's report addresses the licensee's request to exempt 19 valves (note Table 3.1-1 Reference 6) from the Type C testing in 10 CFR Part 50 Appendix J. Our consultant has reviewed the valve list and concluded that an exemption is not required since these valves are not needed in meeting the requirements of 10 CFR Part 50 Appendix J. We concur with our consultant that these valves are not needed in meeting the requirements of Appendix J of 10 CFR Part 50 and therefore, the valves need not be subject to Type C testing.

The licensee also requested a TS change concerned with Type A test which is the overall integrated leak rate test of the containment. Specifically, the change addresses repairs or adjustments to valves and components that can be made prior to this periodic integrated leak rate test in TS 4.4.1.4a. We have issued Amendment No. 27 by letter dated March 23, 1977 providing our evaluation regarding this matter. The amendment finds the change request to TS 4.4.1.4a acceptable which is in agreement with our consultant's conclusion.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts 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.

Dated: March 30, 1981

REFERENCES

1. NRC generic letter regarding implementation of 10 CFR Part 50, Appendix J, at TMI-1, dated August 7, 1975.

2. Metropolitan Edison Company letter dated September 17, 1975, from Mr. R. C. Arnold (MEC) to Director, Division of Reactor Licensing, NRC.

3. Metropolitan Edison Company letter dated October 29, 1975, from Mr. R. C. Arnold (MEC) to Mr. R. Reid (ORB#4); Technical Specification Change Request No. 48.

4. Metropolitan Edison Company letter dated February 18, 1977, from Mr. R. C. Arnold (MEC) to Mr. R. Reid (ORB#4); Technical Specification Change Request No. 48.

5. Metropolitan Edison Company letter dated May 13, 1980, from Mr. J. G. Herbein (MEC) to Mr. R. Reid (ORB#4).

6. Consultant's report, Franklin Research Center letter dated July 3, 1980 to E. J. Butcher, Jr., from C. P. Cafagno.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-289

METROPOLITAN EDISON COMPANY

JERSEY CENTRAL POWER AND LIGHT COMPANY

PENNSYLVANIA ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 63 to Facility Operating License No. DPR-50, issued to Metropolitan Edison Company, Jersey Central Power and Light Company and Pennsylvania Electric Company (the licensees), which revised Technical Specifications for operation of the Three Mile Island Nuclear Station, Unit No. 1 (the facility) located in Londonderry Township, Dauphin County, Pennsylvania. The amendment makes changes to the Technical Specifications related to primary reactor containment leakage testing. The amendment is effective as of its date of issuance.

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. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact 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) the application for amendment dated September 17, 1975, as revised October 29, 1975, February 18, 1977 and May 13, 1980, (2) Amendment No. 63 to License No. DPR-50, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W. Washington, D.C. 20555, and at the Government Publications Section, State Library of Pennsylvania, Box 1601 (Education Building), Harrisburg, Pennsylvania 17126. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, the 30th day of March 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing