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METROPOLITAN EDISON COMPANY
JERSEY CENTRAL POWER & LIGHT COMPANY

AND

PENNSYLVANIA ELECTRIC COMPANY

THREE MILE ISLAND NUCLEAR STATION UNIT 1

Operating License No. DPR-50
Docket No. 50-289
Technical Specification Change Request No. 22

This Technical Specification Change Request is submitted in support of Licensee's request to change Appendix A to Operating License No. DPR-50 for Three Mile Island Nuclear Station Unit 1. As a part of this request, proposed replacement pages for Appendix A are also included.

METROPOLITAN EDISON COMPANY

By R C Arnold
Vice President-Generation

Sworn and subscribed to me this 29th day of October, 1975

Richard I Ruth
Notary Public
RICHARD I. RUTH
Notary Public, Muhlenberg Twp., Berks Co.
My Commission Expires September 23, 1978

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Three Mile Island Nuclear Station Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289

Technical Specification Change Request No. 22

The licensee requests that the attached changed pages replace pages 4-29, 4-31, 4-32, 4-33, 4-34, 4-34a, and 4-34b of the existing technical specifications appendix A.

Reasons for Proposed Change

These changes to the technical specifications are necessary in order to be consistent with the requirements of 10CFR50 Appendix J and the limitations/ design of equipment installed at TMI-1. Specifically, Specifications 4.4.1.2.2 and 4.4.1.2.5 are changed in that normal use of the personnel and emergency air locks' outer doors is relatively high, often once per day. This high usage has necessitated frequent testing to comply with Technical Specifications. Since the hatch doors are designed to seat under accident conditions with reactor building pressure, this pressure must be simulated under periodic door seal test conditions by adjusting the door operating mechanisms for higher than usual seal compression in order to meet acceptance criteria. The resulting stress on the mechanisms has caused mechanism gear failure and seal damage during normal hatch use.

Testing that is required every six months does not require excessive, higher than usual tight mechanism adjustment for two reasons:

(1) the inner hatch door is fitted with strong backs during the six month test and the outer door is seated tightly by test pressure, therefore, the operating mechanism is not involved in seating of the doors, and
(2) the seal interspace is not exposed to test pressure during the six month test.

These changes will allow door operating and latching mechanisms to be adjusted in accordance with manufacturer's recommendations, therefore increasing the reliability of the seals and mechanism and will prevent excessive wear and tear on the system.

Specification 4.4.1.1.3 is changed to include the value of L_t which was determined prior to reactor operation and in accordance with 10CFR50 Appendix J and our present technical specifications.

Specification 4.4.1.2.1 is changed to correct clerical errors, include additional valves that require testing, and delete testing requirements for valves MU-V116, CF-V12A&B, DH-V63, and DH-V64 in that:

a) MU-V116 and CF-V12A&B as check valves are not required to close automatically upon receipt of a containment isolation signal, are not required to operate intermittently under post accident conditions, and are in systems which do not provide a direct connection between the inside and outside atmosphered of the containment under normal operations. Therefore, Appendix J does not require Type "C" testing of these valves, and

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b) DH-V63 and DH-V64 are manual locked closed valves which do not provide a direct connection between the inside and outside containment atmosphere under normal operations. Therefore, Type "C" testing is not required by Appendix J. In addition, under accident conditions leakage through these valves is prevented by the Low Pressure Injection System.

Valves DH-V4A&B have not been included as these valves fit into the same category as valves RR-V4A thru D and exemptions for RR-V4A-D have already been granted, therefore, exclusion of DH-V4A&B is considered to be an extension of this exemption.

Specification 4.4 Bases is changed to include the acceptance criteria for the Fluid Block and Penetration Pressurization Systems required by Appendix J III. B. 1 and III. C. 3.

Safety Analysis Justifying Change

This change meets the requirements of 10CFR50 Appendix J, subject to the interpretations and agreements referenced in our letter of September 17, 1975, except as noted below.

The Licensee hereby requests an exemption from 10CFR50 Appendix J in accordance with 10CFR50.12 for testing of the Reactor Building personnel and emergency air locks' door seals. As stated above, frequent testing at higher pressure (Pa) than necessary to assure proper seal seating and leak tightness has resulted in degradation. The testing pressure and frequency required by this technical specification change request will provide assurance that the subject door seals are in good condition and are capable of proper sealing in the event of a LOCA.

Since the probability of door seal failure under accident conditions is increased by such frequent testing and by testing at such unnecessarily high pressure required by Appendix J, the reliability of this system will be increased by this change and there will be an increased measure of safety for the public and environment.

It should be noted that the "10 psig" test pressure has been recommended by the door manufacturer (Chicago Bridge and Iron Company) for door seal testing.

Since this change either meets the requirements of 10CFR50 Appendix J or provides adequate assurance that the intent of Appendix J (i.e. containment leak tightness) is met, it does not pose undue risk to the health and safety of the public.

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 (of not less than 27.5 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.

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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

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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-V2A/B, MU-V25)

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

5. Core Flood (CF-V2A 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)

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

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).

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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 resilient seal of the personnel and emergency air locks' outer doors shall be tested at six month intervals, except when the air locks are opened during that interval. In this case, they shall be tested after each use but no more than once every three days. During cold shutdown the minimum test frequency is one per week.

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.

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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.

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 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, this periodic test is to be performed without preliminary leak detection surveys or leak repairs and containment isolation valves are to be closed in the normal manner. The minimum test pressure of 27.5 psig for the periodic integrated leakage rate test is sufficiently high to provide an accurate measurement of the leakage rate and it duplicates the pre-operational leakage rate test at the reduced pressure. The specification provides a relationship for relating the measured leakage of air at the reduced pressure to the potential leakage at 55 psig. The minimum of 24 hours was specified for the integrated leakage rate test to help stabilize conditions and thus improve accuracy and to better evaluate data scatter. The frequency of the periodic integrated leakage rate test is keyed to the refueling schedule for the reactor, because these tests can best be performed during refueling shutdowns.

The specified frequency of periodic integrated leakage rate tests is based on three major considerations. First is the low probability of leaks in the liner, because of conformance of the complete containment to a 0.10 percent leakage rate at 55 psig during pre-operational testing and the absence of any significant stresses in the liner during reactor operation. Second is the more frequent testing, at design pressure, of those portions of the containment envelope that are most likely to develop leaks during reactor operation (penetrations and isolation valves which are not continuously pressurized by the penetration pressurization system or are not fluid blocked post-accident by the fluid block system) and the low value (0.06 percent) of leakage that is specified as acceptable from penetrations and isolation valves. Third is the tendon stress surveillance program which provides assurance that an important part of the structural integrity of the containment is maintained.

More frequent testing of various penetrations is specified as these locations are more susceptible to leakage than the reactor building liner due to the mechanical closure involved. Particular attention is given to testing those penetrations and process lines not serviced by the penetration pressurization system or the fluid block system. The basis for specifying a total leakage rate of 0.06 percent from those penetrations and isolation valves is that more than one-half of the allowable integrated leakage rate will be from these sources.

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Valve operability tests are specified to assure proper closure or opening of the reactor building isolation valves to provide for isolation or functioning of Engineered Safety Features systems. Valves will be stroked to the position required to fulfill their safety function unless it is established that such testing is not practical during operation. Valves that cannot be full-stoke tested will be part-stroke tested during operation and full-stroke tested during each normal refueling shutdown.

Specification 4.4.1.2.1(c) specifies those containment isolation valves which will be tested using the Fluid Block System. Fluid leakage from valves sealed with water from Tank A shall not exceed 0.00354 gpm and from Tank B shall not exceed 0.00347 gpm. These leak rates ensure that the criteria of III. C. 3 of Appendix J are satisfied.

Specification 4.4.1.2.1(d) specifies those isolation valves or blank flanges which will be tested using the Penetration Pressurization System. Tests conducted using this system shall be considered acceptable if the total system leakage does not exceed 81 SCFH. With this leak rate, sufficient excess air in the instrument and service air receivers and nitrogen from the nitrogen manifold system will be available to maintain the system pressure in excess of 57 psig for approximately 10.5 hrs. This is sufficient time to allow the containment cooling systems to return containment pressure to normal and to ensure that the instrument air compressor can be manually loaded onto the Engineered Safeguards Bus should the accident occur at the same time off-site power is lost. Thereafter, the instrument air compressor can maintain system pressure for the required 30 days.

<u>REFERENCE</u>

(1) FSAR, Section 5.

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