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.36
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 licensees), dated October 28, 1977, 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 (i) 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. 36 , 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

Robert W. Reid

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 7, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 36

FACILITY OPERATING LICENSE NO. DPR-50

DOCKET NO. 50-289

Revise Appendix A as follows:

Remove Pages	Insert Pages
1-5	1-5
4-34a & 4-34b	4-34a & 4-34b

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

1.6 POWER DISTRIBUTION

1.6.1 QUADRANT POWER TILT

Quadrant power tilt is defined by the following equation and is expressed in percent.

$$100 \left[\frac{\text{Power in any core quadrant}}{\text{Average power of all quadrants}} - 1 \right]$$

The quadrant tilt limits are stated in Specification 3.5.2.4.

1.6.2 REACTOR POWER IMBALANCE

Reactor power imbalance is the power in the top half of the core minus the power in the bottom half of the core expressed as a percentage of rated power. Imbalance is monitored continuously by the RPS using input from the power range channels. Imbalance limits are defined in Specification 2.1 and imbalance setpoints are defined in Specification 2.3.

1.7 CONTAINMENT INTEGRITY

Containment integrity exists when the following conditions are satisfied:

a. The equipment hatch is closed and sealed and both doors of the personnel hatch and emergency hatch are closed and sealed except as in "b" or "f" below.

b. At least one door on each of the personnel hatch and emergency hatch is closed and sealed during refueling or personnel passage through these hatches.

c. All nonautomatic containment isolation valves and blind flanges are closed as required by the "Containment Integrity Check List" attached to the operating procedure "Containment Integrity and Access Limits."

d. All automatic containment isolation valves are operable or locked closed.

e. The containment leakage determined at the last testing interval satisfies Specification 4.4.1.

f. One door of the personnel hatch or emergency hatch may be open for up to 24 hours for maintenance, repair or modification provided the other door of the hatch is maintained closed and has been leak tested and found to meet the local leak rate criteria for door seals within 24 hours prior to the maintenance, repair or modification.

1.8 FIRE SUPPRESSION WATER SYSTEM

A FIRE SUPPRESSION WATER SYSTEM shall consist of: a water source, gravity tank or pump and distribution piping with associated sectionalizing control or isolation valves. Such valves include yard hydrant curb valves, and the first valve upstream of the water flow alarm device on each sprinkler, hose standpipe or spray system riser.

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 is 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 will be strength tested at 115 percent of design pressure and leak rate tested at the design pressure. The containment will also be leak tested prior to initial operation at approximately 50 percent of the design pressure. These tests will verify that the leakage rate from reactor building pressurization satisfies the relationships given in the specifications.

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.

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 of 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.

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-stroke tested will be part-stroke tested during operation and full-stroke tested during each normal refueling shutdown.

Periodic surveillance of the airlock interlock systems is specified to assure continued operability and preclude instances where one or both doors are inadvertently left open. When an interlock is inoperable and containment integrity is required, local supervision of airlock operation is specified.

REFERENCE

(1) FSAR, Section 5.