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 I

Amendment No. 12
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 November 17, 1975, 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 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. An environmental statement or negative declaration need not be prepared in connection with the issuance of this amendment.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment.

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:
February 17, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 12

FACILITY OPERATING LICENSE NO. DPR-50

DOCKET NO. 50-289

Replace pages 2-3, 2-4, 3-1, 3-2, 4-7, and 4-8 of the Appendix A Technical Specifications with the attached pages. Changes on these pages are shown by marginal lines.

The curve of Figure 2.1-1 is the most restrictive of all possible reactor coolant pump-maximum thermal power combinations shown in Figure 2.1-3. The curves of Figure 2.1-3 represent the conditions at which a minimum DNBR of 1.3 is predicted at the maximum possible thermal power for the number of reactor coolant pumps in operation or the local quality at the point of minimum DNBR is equal to 15 percent,[3] whichever condition is more restrictive.

Using a local quality limit of 15 percent at the point of minimum DNBR as a basis for curve 3 of Figure 2.1-3 is a conservative criterion even though the quality at the exit is higher than the quality at the point of minimum DNBR.

The DNBR as calculated by the W-3 correlation continually increases from the point of minimum DNBR, so that the exit DNBR is always higher and is a function of the pressure. Extrapolation of the W-3 correlation beyond its published quality range of +15 percent is justified on the basis of experimental data.[4]

The maximum thermal power for three pump operation is 86.7 percent due to a power level trip produced by the flux-flow ratio (74.7 percent flow x 1.08 = 80.7 percent power) plus the maximum calibration and instrumentation error. The maximum thermal power for other reactor coolant pump conditions is produced in a similar manner.

For each curve of Figure 2.1-3, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.3 or a local quality at the point of minimum DNBR less than 15 percent for that particular reactor coolant pump situation. The 1.3 DNBR curve for four pump operation is more restrictive than any other reactor coolant pump situation because any pressure/temperature point above and to the left of the four pump curve will be above and to the left of the other curves.

REFERENCES

(1) FSAR, Section 3.2.3.1.1

(2) FSAR, Section 3.2.3.1.1.c

(3) FSAR, Section 3.2.3.1.1.k

(4) The following papers which were presented at the Winter Annual Meeting, ASME, November 18, 1969, during the Two-Phase flow and Heat Transfer in Rod Bundles Symposium:

(a) Wilson, et. al., Critical Heat Flux in Non-Uniform Heater Rod Bundles.

(b) Gellerstedt, et. al., Correlation of a Critical Heat Flux in a Bundle Cooled by Pressurized Water.

2.2 SAFETY LIMITS - REACTOR SYSTEM PRESSURE

Applicability

Applies to the limit on reactor coolant system pressure.

Objective

To maintain the integrity of the reactor coolant system and to prevent the release of significant amounts of fission product activity.

Specification

2.2.1 The reactor coolant system pressure shall not exceed 2750 psig when there are fuel assemblies in the reactor vessel.

Bases

The reactor coolant system [1] serves as a barrier to prevent radionuclides in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure, the reactor coolant system is a barrier against the release of fission products. Establishing a system pressure limit helps to assure the integrity of the reactor coolant system. The maximum transient pressure allowable in the reactor coolant system pressure vessel under the ASME Code, Section III, is 110 percent of design pressure.[2] The maximum transient pressure allowable in the reactor coolant system piping, valves, and fittings under ANSI Section B31.7 is 110 percent of design pressure. Thus, the safety limit of 2750 psig (110 percent of the 2500 psig design pressure) has been established.[2] The settings for the reactor high pressure trip (2355 psig) and the pressurizer code safety valves (2435 psig)[3] have been established in accordance with ASME Boiler and Pressurizer Vessel Code, Section III, Article 9, Winter, 1968 to assure that the reactor coolant system pressure safety limit is not exceeded. The initial hydrostatic test was conducted at 3125 psig (125 percent of design pressure) to verify the integrity of the reactor coolant system. Additional assurance that the reactor coolant system pressure does not exceed the safety limit is provided by setting the pressurizer electromatic relief valve at 2255 psig.[4]

REFERENCES

(1) FSAR, Section 4

(2) FSAR, Section 4.3.10.1

(3) FSAR, Section 4.2.4

(4) FSAR, Table 4-1

3. LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR COOLANT SYSTEM

3.1.1 OPERATIONAL COMPONENTS

Applicability

Applies to the operating status of reactor coolant system components.

Objective

To specify those limiting conditions for operation of reactor coolant system components which must be met to ensure safe reactor operations.

Specification

3.1.1.1 Reactor Coolant Pumps

a. Pump combinations permissible for given power levels shall be as shown in Specification Table 2.3-1.

b. The boron concentration in the reactor coolant system shall not be reduced unless at least one reactor coolant pump or one decay heat removal pump is circulating reactor coolant.

3.1.1.2 Steam Generator

a. One steam generator shall be operable whenever the reactor coolant average temperature is above 250 F.

3.1.1.3 Pressurizer Safety Valves

a. The reactor shall not remain critical unless both pressurizer code safety valves are operable with a lift setting of 2435 PSIG ±1%.

b. When the reactor is subcritical, at least one pressurizer code safety valve shall be operable if all reactor coolant system openings are closed, except for hydrostatic tests in accordance with ASME Boiler and Pressure Vessel Code, Section III.

Bases

A reactor coolant pump or decay heat removal pump is required to be in operation before the boron concentration is reduced by dilution with makeup water. Either pump will provide mixing which will prevent sudden positive reactivity changes caused by dilute coolant reaching the reactor. One decay heat removal pump will circulate the equivalent of the reactor coolant system volume in one half hour or less..

The decay heat removal system suction piping is designed for 300 F and 370 psig; thus, the system can remove decay heat when the reactor coolant system is below this temperature.[2,3]

One pressurizer code safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than that required by the sum of the available heat sources which are pump energy, pressurizer heaters, and reactor decay heat.[4] Both pressurizer code safety valves are required to be in service prior to criticality to conform to the system design relief capabilities. The code safety valves prevent overpressure for a rod withdrawal accident.[5] The pressurizer code safety valve lift set point shall be set at 2435 psig ±1 percent allowance for error and each valve shall be capable of relieving 311,700 lb/h of saturated steam at a pressure not greater than three percent above the set pressure.

REFERENCES

(1) FSAR, Tables 9-10 and 4-3 through 4-7

(2) FSAR, Sections 4.2.5.1 and 9.5.2.3

(3) FSAR, Section 4.2.5.4

(4) FSAR, Sections 4.3.10.4 and 4.2.4

(5) FSAR, Section 4.3.7

TABLE 4.1-1 (Continued)

	CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
38.	Steam Generator Water Level	W	NA	R	
39.	Turbine Overspeed Trip	NA	R	NA	
40.	Sodium Thiosulfate Tank Level Indicator	NA	NA	R	
41.	Sodium Hydroxide Tank Level Indicator	NA	NA	R	
42.	Diesel Generator Protective Relaying	NA	NA	R	
43.	4 KV ES Bus Undervoltage Relays (Diesel Start)	NA	M(1)	R	(1) Relay operation will be checked by local test pushbuttons.
44.	Reactor Coolant Pressure DH Valve Interlock Bistable	S(1)	M	R	(1) When reactor coolant system is pressurized above 300 psig or Taves is greater than 200°F.

S - Each Shift

D - Daily

W - Weekly

M - Monthly

T/W - Twice per week

B/M - Every 2 months

Q - Quarterly

P - Prior to each startup if not done previous week

R - Each Refueling Period

NA - Not Applicable

B/W - Every two weeks

TABLE 4.1-2

MINIMUM EQUIPMENT TEST FREQUENCY

	Item	Test	Frequency
1.	Control Rods	Rod drop times of all full length rods	Each refueling shutdown
2.	Control Rod Movement	Movement of each rod	Every two weeks, when reactor is critical
3.	Pressurizer Safety Valves	Setpoint *	50% each refueling period
4.	Main Steam Safety Valves	Setpoint	25% each refueling period
5.	Refueling System Interlocks	Functional	Start of each refueling period
6.	Main Steam Isolation Valves	(See Section 4.8)	
7.	Reactor Coolant System Leakage	Evaluate	Daily, when reactor coolant system temperature is greater than 525°F
8.	Charcoal and high efficiency filters for Control Room, and RB Purge Filters	DOP test on HEPA filters, freon test on charcoal filter units	Each refueling period and at any time work on filters could alter their integrity
9.	Spent Fuel Cooling System	Functional	Each refueling period prior to fuel handling
10.	Intake Pump House Floor (Elevation 262 Ft 6 in.)	(a) Silt Accumulation-Visual inspection of Intake Pump House Floor	Each refueling period
		(b) Silt Accumulation Measurement of Pump House Flow	Quarterly
11.	Hydraulic Shock Suppressors on Safety Related Systems	Inspection of Hydraulic Shock Suppressors (Snubbers) on Safety Related Systems.	Refueling Interval

* The setpoint of the pressurizer code safety valves shall be in accordance with ASME Boiler and Pressurizer Vessel Code, Section III, Article 9, Winter, 1968.