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 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 15
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 March 23, 1976, 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. 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

P. Kooney for

Robert W. Reid, Chief
Operating Reactors Branch No. 4
Division of Operating Reactors

Attachment:
Changes to the
Technical Specifications

Date of Issuance: MAY 14 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 15

FACILITY OPERATING LICENSE NO. DPR-50

DOCKET NO. 50-289

Revise Appendix A as follows:

Remove pages 3-3, 3-4, 3-5, 4-11, 4-12, 4-13, and 4-14 and insert attached pages 3-3, 3-4, 3-5, 4-11, 4-12, 4-13 and 4-14.

Changes on the revised pages are shown by marginal lines. Pages 4-12 and 4-14 are unchanged and are included for convenience only.

3.1.2 PRESSURIZATION, HEATUP, AND COOLDOWN LIMITATIONS

Applicability

Applies to pressurization, heatup, and cooldown of the reactor coolant system.

Objective

To assure that temperature and pressure changes in the reactor coolant system do not cause cyclic loads in excess of design for reactor coolant system components.

Specification

3.1.2.1 For the first 1.7×10^6 thermal megawatt days (approximating two years) the reactor coolant pressure and the system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1-1 and Figure 3.1-2 and are as follows:

Heatup:
Allowable combinations of pressure and temperature shall be to the right of and below the limit line in Figure 3.1-1. Heatup rates shall not exceed those shown on Figure 3.1-1.

Cooldown:
Allowable combinations of pressure and temperature for a specific cooldown shall be to the left of and below the limit line in Figure 3.1-2. Cooldown rates shall not exceed those shown on Figure 3.1-2.

Hydro Tests:
For isothermal system hydrotests during the first two years of operations, the system may be pressurized to the limits set forth in Specification 2.2, when there are fuel assemblies in the vessel and to ASME Code Section III limits when no fuel assemblies are present if the system temperature is 215 F or greater. The system may be tested to a pressure of 1150 psig provided system temperature is 175 F or greater. Initial system hydrotests prior to criticality may be conducted if the reactor coolant system temperature is 118 F or greater.

3.1.2.2 The secondary side of the steam generator shall not be pressuirzed above 200 psig if the temperature of the steam generator shell is below 100 F.

3.1.2.3 The pressurizer heatup and cooldown rates shall not exceed 100°F in any one hour. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 430F.

3.1.2.4 Within two effective full power years of operation, Figure 3.1-1 and 3.1-2 shall be updated in accordance with criteria acceptable to the NRC.

Bases

All reactor coolant system components are designed to withstand the effects of cyclic loads due to system temperature and pressure changes.[1] These cyclic

loads are introduced by unit load transients, reactor trips, and unit heatup and cooldown operations. The number of thermal and loading cycles used for design purposes are shown in Table 4-8 of the FSAR. The maximum unit heatup and cooldown rate of 100 F in any one hour satisfies stress limits for cyclic operation.[2] The 200 psig pressure limit for the secondary side of the steam generator at a temperature less than 100°F satisfies stress levels for temperatures below the DTT.[3] The reactor vessel plate material and welds have been tested to verify conformity to specified requirements and a maximum NDTT value of 30 F has been determined based on Charpy V-notch tests. The maximum NDTT value obtained for the steam generator shell material and welds was 40 F.

The heatup and cooldown rate limits in this specification are not intended to limit instantaneous rates of temperature change, but are intended to limit temperature changes such that there exists no one hour interval, in which a temperature change greater than the limit takes place.

Figures 3.1-1 and 3.1-2 contain the limiting reactor coolant system pressure-temperature relationship for operation at DTT[4] and below to assure that stress levels are low enough to preclude brittle fracture. These stress levels and their bases are defined in Paragraph 4.3.3 of the FSAR.

As a result of fast neutron irradiation in the region of the core, there will be an increase in the NDTT with accumulated nuclear operation. The predicted maximum NDTT increase for the 40-year exposure is shown on Figure 4-10.[4] The actual shift in NDTT will be determined periodically during plant operation by testing of irradiated vessel material samples located in this reactor vessel.[5] The results of the irradiated sample testing will be evaluated and compared to the design curve (Figure 4-11 of the FSAR) being used to predict the increase in transition temperature.

The design value for fast neutron (E > 1 MeV) exposure of the reactor vessel is 3.1×10^{10} n/cm^2 sec at the reference design power of 2568 MWt and an integrated exposure of 3.0×10^{19} n/cm^2 for 40 years operation.[6] The calculated maximum values are 2.2×10^{10} n/cm^2 sec and 2.2×10^{19} n/cm^2 integrated exposure for 40 years operation at 80 percent load.[4] Figure 3.1-1 is based on the design value which is considerably higher than the calculated value. The DTT value for Figure 3.1-1 is based on the projected NDTT at the end of the first two effective full power years of operation. During these two years, the energy output has been conservatively estimated to be 1.7×10^6 thermal megawatt days, which is equivalent to 655 days at 2568 MWt core power. The projected fast neutron exposure to the reactor vessel for the two years is 1.7×10^{18} n/cm^2 which is based on the 1.7×10^6 thermal megawatt days and the design value for fast neutron exposure.

The actual shift in NDTT will be established periodically during plant operation by testing vessel material samples which are irradiated by securing them periodically near the inside wall of the vessel in the core area to achieve an average effective exposure between 1 and 3 times that of the reactor vessel inner surface. To compensate for the increases in the NDTT caused by irradiation, the limits on the pressure-temperature relationship are periodically changed to stay within the established stress limits during heatup and cooldown.

The NDTT shift and the magnitude of the thermal and pressure stresses are sensitive to integrated reactor power and not to instantaneous power level. Figures 3.1-1 and 3.1-2 are applicable to reactor core thermal ratings up to 2568 MWt.

The pressure limit line on Figure 3.1-1 has been selected such that the reactor vessel stress resulting from internal pressure will not exceed 15 percent yield strength considering the following:

a. A 25 psi error in measured pressure

b. System pressure is measured in either loop

c. Maximum differential pressure between the point of system pressure measurement and reactor vessel inlet for all operating pump combinations

For adequate conservatism, in lieu of portions of the Operational Requirements of Appendix G to 10 CFR 50, a maximum pressure of 550 psig and a maximum heatup rate of 50°F in any one hour has been imposed below 275 F as shown on Figure 3.1-1.

The spray temperature difference restriction, based on a stress analysis of the spray line nozzle is imposed to maintain the thermal stresses at the pressuirzer spray line nozzle below the design limit. Temperature requirements for the steam generator correspond with the measured NDTT for the shell.

REFERENCES

(1) FSAR, Section 4.1.2.4

(2) ASME Boiler and Pressure Code, Section III, N-415

(3) FSAR, Section 4.3.10.5

(4) FSAR, Section 4.3.3

(5) FSAR, Section 4.4.5

(6) FSAR, Sections 4.1.2.8 and 4.3.3

4.2 REACTOR COOLANT SYSTEM INSERVICE INSPECTION

Applicability

This technical specification applies to the inservice inspection of the reactor coolant system pressure boundary and portions of other safety oriented system pressure boundaries as shown on Figure 4.2-1.

Objective

The objective of this inservice inspection program is to provide assurance of the continuing integrity of the reactor coolant system while at the same time minimizing radiation exposure to personnel in the performance of inservice inspections.

Specification

4.2.1 The inservice inspection program to be followed is outlined in Table 4.2-1. Except as provided for in this Table and as discussed herein, the inservice inspection program is in accordance with the ASME Code, Section XI, Rules for Inservice Inspection of Nuclear Reactor Coolant Systems, dated January 1, 1970, as modified by the Winter 1970 Addenda. Prior to initial plant operation a pre-operational inspection of the plant will be performed of at least the areas listed in the ASME Code, provided accessibility and the necessary inspection techniques are available for each of these areas. The only exception to this will be areas where the necessary base line data is already available and has been obtained by the same techniques as will be used during inservice inspection.

4.2.2 Reactor vessel irradiation capsules are planned to be withdrawn for testing at specimen exposures (E > 1MeV) equivalent to 3, 9.5, 16, and 22.5 effective full power years of operation. Withdrawal schedules for testing may be modified to coincide with those refueling outages most closely approaching the testing withdrawal schedule and may be adjusted following evaluation of data from each withdrawal in accordance with 10 CFR 50 Appendix H paragraph II.C.3.g. Specimen capsules not subjected to destructive testing after Cycle 1 operation will be removed and stored during Cycle 2 operation, but shall be re-installed prior to Cycle 3 operation.

4.2.3 The accessible portions of one reactor coolant pump motor flywheel assembly will be ultrasonically inspected within 3-1/3 years, two within 6-2/3 years, and all four by the end of the 10 year inspection interval. However, the U.T. procedure is developmental and will be used only to the extent that it is shown to be meaningful. The extent of coverage will be limited to those areas of the flywheel which are accessible without motor disassembly, i.e., can be reached through the access ports. Also, if radiation levels at the lower access ports are prohibitive, only the upper access ports will be used.

4.2.4 The inspection schedule may be modified to coincide with those refueling or maintenance outages most closely approaching the inspection schedule.

4.2.5 Sufficient records of each inspection shall be kept to allow comparison and evaluation of future inspections.

4.2.6 The inservice inspection shall be reviewed at the end of 5 years to consider incorporation of new inspection techniques and equipment which have been proven practical, and a possible extension of the program to additional examination areas. The conclusions of this review shall be submitted to the AEC for evaluation.

Inspection Bases

a. The nuclear plant was designed prior to the issuance of Section XI of the ASME Code, Rules for Inservice Inspection of Nuclear Reactor Coolant Systems, dated January 1, 1970. However, sufficient accessibility was included in the design to perform most inspections discussed in the Code. The proposed inspection program follows the Code except that inspections are focused on areas which engineering analysis has indicated are subject to the more critical stress, radiation, or transient conditions. The areas selected for inspection on this basis are listed in Table 4.2-1. These areas are exposed to the more severe conditions (which are still well within Code limits) in the reactor coolant system. Therefore, they are expected to indicate potential problems before significant flaws develop in the selected areas or in other areas. It is considered that the focused approach specified herein will result in a meaningful inspection program in that it will provide assurance of continuing plant integrity.

In those areas where inspection methods are developmental, such as for remote inspection of the reactor vessel welds, reactor vessel nozzle inside radii and welds, and ultrasonic inspection of pressurizer support bracket welds, the inspection methods will be developed and tested to the extent practicable during pre-operational inspections. (Development of inspection techniques will not be attempted on radioactive equipment, unless necessary to explore a specific problem.) A pre-operational inspection is planned of areas listed in the ASME Code which are within the inservice inspection boundaries and which are accessible for inspection. However, as discussed above, in areas where inspection methods are developmental, the inspections will only be performed to the extent practicable. Once an inspection method is selected for a particular inspection (e.g., U.T. for most volumetric inspections), it is intended that all subsequent inservice inspections be performed using the identical method and on the same component parts wherever practicable.

In addition to the above inspection, if any of the components within the inservice inspection boundary are disassembled for maintenance, the accessible parts will be given a normal visual examination as part of the routine plant maintenance operations.

b. The vessel specimen surveillance program is based on specimen equivalent exposure years of 3, 9.5, 16, and 22.5 EFPY referenced to 1/4 t[2]. These times were selected to meet the requirements of Appendix H to 10 CFR 50.

The specimen capsules not subjected to destructive testing after cycle 1 operation are to be stored to permit the redesign of the capsule holders. The stored specimen capsules will be re-installed following completion of cycle 2 operation in a manner such that a specimen equivalent exposure (E > 1MeV) between 1 and 3 times that of the reactor vessel inner surface as required by 10 CFR 50 Appendix H is achieved.

c. The reactor coolant pump motor flywheel ultrasonic test procedure is being developed to detect flaws of a small enough size to provide assurance of continued integrity, based upon a conservative fracture mechanics evaluation.

REFERENCE

(1) FSAR, Section 4.4

(2) BAW-10100A February 1975

TABLE 4.2-1

INSTRUMENT SURVEILLANCE REQUIREMENTS

Item IS - 261*	Examination Category-IS - 251*	Areas to be Examined	Examination Methods	Inspection Schedule and Extent	Remarks
	Reactor Vessel and Closure Head				
1.1	A	Longitudinal and circumferential welds in core region	Volumetric	At or near the end of the 10 year inspection interval, 10% of each longitudinal and 5% of each circumferential weld will be inspected. When the neutron fluence exceeds 10^{19} nvt (E 1 Mev or greater) the length of each weld which is inspected will be increased to 50%.	Note 1**
1.2	B	Longitudinal and circumferential welds in the shell and heads (other than those covered in Items 1.1, 1.3, and 1.4)	See remarks	See Remarks	The welds in this category are the circumferential weld just above the support skirt junction with the vessel, the lower head to vessel weld which is only accessible from the bottom of the vessel, and the circumferential nozzle belt weld. The stresses in these welds are lower than in the other reactor vessel welds to be inspected per Items 1.1 and 1.3. Therefore, no inspections are planned.
1.3	C	Vessel to flange and head to flange circumferential welds	Volumetric	1/3 about every 3-1/3 years.	This inspection will be performed using automated U.T. Note 1 applies.

* IS-261 and IS-251 refer to Tables in Section XI of the ASME Code.

** See the notes at the end of this Table.

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 15 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 1

DOCKET NO. 50-289

INTRODUCTION

By letter dated March 23, 1976, Metropolitan Edison Company (the licensee) requested an exemption from the requirements of 10 CFR Part 50, Appendix H, Section II.C.2 to permit the operation of Three Mile Island Nuclear Station Unit 1 (TMI-1), Cycle 2 with the reactor vessel surveillance capsules removed from the reactor vessel. The licensee requested corresponding changes to the Technical Specifications appended to Facility Operating License No. DPR-50 for the TMI-1. These changes would reflect (a) the removal of the reactor vessel surveillance capsules for Cycle 2 operation (b) a revision in the surveillance capsule withdrawal times to conform with 10 CFR Part 50, Appendix H, Section II.C.3.C and (c) clarification of the term "effective full power years (EFPY)" to be consistent with standard Technical Specifications.

The licensee also advised that the capsule holder tubes would be removed for Cycle 2.

DISCUSSION

The Three Mile Island Unit 1 design includes three reactor vessel surveillance capsule holder tubes located adjacent to the reactor inside vessel wall. Each holder tube contains two surveillance capsules which hold the specimens to be irradiated in accordance with the requirements of the reactor vessel material surveillance program as described in Appendix H to 10 CFR Part 50. The purpose of the surveillance program is to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel regions resulting from their exposure to neutron irradiation and the thermal environment.

In a recent inspection of the surveillance capsule holder tubes, conducted during the current refueling outage at TMI-1, damage to the holder tubes was observed. The damage was attributed to contact and flow induced relative motion between the holder tubes and various components of the surveillance capsule train (push-rod spacers, holddown springs, and surveillance capsule rings) which position and hold the surveillance capsules in place during reactor operation. Two of the three holder tubes were found to be severed at the axial location of the second push-rod spacer from the top and one of these tubes was so severely worn at the axial location of the first push-rod spacer that it became separated at that location during capsule removal. The third holder tube was intact following capsule removal.

To preclude the possibility of loose parts occurring during Cycle 2, the surveillance capsules and holder tubes will be removed prior to Cycle 2 operation. Engineering of new holder tube and push-rod assembly design modifications and material procurement will be completed during Cycle 2 to allow installation of the revised holder tubes prior to the start of Cycle 3.

As a separate issue, the present schedule for withdrawals, subsequent to the first capsule withdrawal, is being modified to conform with Appendix H (which was issued after the TMI-1 Technical Specifications were developed).

In addition, Specification 3.1.2.4 is being changed to clarify that the first withdrawal would occur at a refueling outage nearest 2 EFPY of reactor exposure.

EVALUATION

As required by Paragraph II.C.2 of Appendix H to 10 CFR Part 50, the surveillance capsules of TMI-1 are positioned during reactor operation so that the neutron flux received by the specimens is at least as high, but not more than three times as high, as that received by the vessel inner surface. A recent calculation by the licensee, acceptable to the staff, indicates that the neutron flux is 2.4 times greater at the specimen location than at the reactor vessel wall at 1.4 wall thickness (1/4t). Cycle 1 of TMI-1 accumulated approximately 1.3 EFPY of actual exposure to the reactor vessel wall at 1.4t; therefore the specimen accumulated approximately 3.2 EFPY of equivalent irradiation. Cycles 2 and 3 are planned to accumulate 0.8 EFPY and 0.75 EFPY of actual exposure respectively. Therefore, the specimens removed after Cycle 1 have already received an irradiation equivalent to 3.12 EFPY (1.3 x 2.4) which is more irradiation than the vessel wall at 1/4t will accumulate during the first three cycles of operation (2.85 EFPY).

The irradiation effects accumulated during Cycle 1 will not be altered in those specimens stored throughout Cycle 2. When these specimens are reinstalled at the beginning of Cycle 3, the Technical Specifications will be revised, based on information acquired with the specimens tested after Cycle 1, to be applicable at least through the first 5 EFPY of reactor vessel exposure. The specimen surveillance program is now based on equivalent exposure years of 3, 9.5, 16, and 22.5 EFPY referenced to 1/4t so as to meet the requirements of Appendix H to 10 CFR Part 50.

In view of the above, we consider it acceptable to allow the licensee to remove the surveillance specimen capsules and holder tubes during Cycle 2 of TMI-1. The specimen capsules not subjected to destructive testing after Cycle 1 operation may be stored until the beginning of Cycle 3 to permit redesign of the capsule holders. We also concur in the removal of the three capsule holder tubes during Cycle 2.

The withdrawal schedule that is based on specimen equivalent exposure years of 3, 9.5, 16, and 22.5 EFPY referenced to 1/4t meets the requirements of Appendix H to 10 CFR Part 50.

The clarification of wording in Technical Specification 3.12 relating to effective full power years of operation is acceptable.

We have determined that this 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 this 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 statement, negative declaration, or 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 change 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 change 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: MAY 14 1976