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. 40

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, 1976.

F L Sawyer
Notary Public

NOTARY PUBLIC
[illegible] County, Pa
[illegible] Nov. 19, 1979

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Metropolitan Edison Company (Met-Ed)
Three Mile Island Nuclear Station Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289

## Technical Specification Change Request No.40

The Licensee requests that the attached changed pages (v, 3-4, 3-5, 4-11, 4-13, and table 4.2-2) replace pages v, 3-4, 3-5, 4-11, and 4-13 of the existing technical specifications.

## Reason For Proposed Change

As a result of damage reported in our letters of March 18, 1976, and March 23, 1976, the TMI-1 reactor vessel surveillance holder tubes were removed. TMI-1 Cycle 2 operations began in accordance with Amendment 15 and the exemption to 10CFR50 Appendix H issued on May 14, 1976, in response to our requests of March 23, 1976.

Amendment 15 requires that the TMI-1 surveillance specimens be reinstalled prior to TMI-1 Cycle 3 operation. In order to achieve this goal, new surveillance holder tubes must be installed. Babcock and Wilcox (B&W) attempted to design a new holder tube that could be installed without removal of the core barrel. (The design in concept was similar to the original holder tube design except that no push rod assembly was used.) The B&W design did not prove to be adequate since prototype flow testing indicated that unacceptable wear would be experienced.

Since an acceptable design, which could be installed without removing the core barrel, could not be found, other designs were explored. Installation of all these other designs involves core barrel removal and underwater machine work or welding on an irradiated thermal shield as well as numerous other complex operations. We feel that quality assurance and inspection of these operations would be very difficult, that the operations have a high potential for creating significant unforeseen problems, and that the expected man-REM exposure would be excessive. In addition, it is extremely probable that the Cycle 3 refueling outage would be extended by several months. Therefore, we have concluded that a TMI-1 and 2 site integrated reactor vessel surveillance program permitted by 10CFR50 Appendix H paragraph II.C.4 is the only reasonable alternative.

As explained above, this change request is necessary to avoid an extended and costly outage and to avoid excessive personnel radiation exposure, while assuring that changes in the reactor vessel toughness properties are appropriately monitored.

## Safety Evaluations

The purpose of the reactor vessel surveillance program is to monitor changes in the fracture toughness properties to permit the determination of the conditions under which the vessel can be operated with adequate margins of safety throughout service life. To accomplish this objective, Appendix H to 10CFR50:

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1. requires surveillance specimen capsules to be irradiated at fast neutron flux levels one to three times the level existing at the vessel inner surface.

2. specifies the withdrawal requirements for the first four capsules.

3. requires irradiation of a standby capsule.

4. and requires provisions be made for additional surveillance tests to monitor the effects of annealing and subsequent irradiation, if required.

In addition, from an operating standpoint it is desirable to collect data prior to the vessel having reached a given exposure in order to avoid unnecessarily restricting plant operations and to ensure important changes in fracture toughness properties are foreseen.

A site integrated surveillance program to irradiate both TMI-1 and TMI-2 capsules in the TMI-2 reactor vessel achieves the above objectives. The sequence of removal/insertion of the various capsules and their attendant exposure is shown in table II. Several conservative assumptions shown in table I were made to demonstrate that, even given a delay in TMI-2 operations and poor TMI-2 performance combined with excellent TMI-1 performance, the above surveillance program objectives can be met.

As indicated by Tables I and II, the average effective capsule exposure for both TMI-1 and TMI-2 is one to three times that of the reactor vessel at the ¼t location. In addition, all of the required test data will be obtained and available prior to the vessels reaching approximately one-half of their service life. An effort has also been made to minimize the number of capsule insertions/removals while at the same time irradiating two capsules for each unit to lead the vessel prior to reaching ½ of service life (capsule exposures can be equalized at any time). This assures that both the standby capsule provision and the provision for monitoring the effects of annealing and subsequent irradiation are met.

It should be noted that due to the similarity of TMI-1 to TMI-2, no significant differences in the fast neutron flux spectrum, operating temperatures or other environmental conditions will exist for the TMI-1 specimens in TMI-2. As a result, a site integrated surveillance program for the irradiation of both TMI-1 and TMI-2 capsules in TMI-2 will have no adverse effect on the quality of the data obtained.

In summary, the proposed TMI site integrated Reactor Vessel surveillance program provides assurance that both the TMI-1 and TMI-2 reactor vessels will be monitored such that vessel operating limits can be established and demonstrated to be conservative throughout service life. Therefore, this change does not represent undue risk to the health and safety of the public.

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TABLE I

SCHEDULING DATA FOR THE TMI SITE INTEGRATED RV SURVEILLANCE PROGRAM

| | TMI-1 | TMI-2 | Notes |
|---|---|---|---|
| RV Service Life | 32 EFPY | 32 EFPY | (1) |
| Assumed Capacity Factor | 0.8 | 0.6 | |
| Thermal Power (MWt) | 2535 | 2772 | |
| Assumed 1st Cycle Length (EFPY) | 1.3 (Actual) | 1.20 | |
| Assumed 2nd Cycle Length (EFPY) | 0.8 | 0.8 | |
| Assumed Subsequent Cycle Length (EFPY) | .7 | .7 | |
| Surveillance Specimen Lead Factor @ ¼t in TMI-2 | 3.0 | 2.8 | (8)(9) |
| Specimen Equivalent Exposure to Date (EFPY) | 0 | 0 | (10) |
| Number of Capsules to be Irradiated | 5 | 6 | (2) |
| Req. Withdrawal Times in EFPY Specimen Exposure | 3, 10, 17, 24 | 3, 10, 17, 24 | (3) (4) (5) (6) |
| Unit Startup Date | NA | 6/79 | (7) |

Notes:

(1) 40 yrs. at 0.8 capacity factor = 32 EFPY exposure
(2) First TMI-1 capsule subjected to destructive testing
(3) Predicted shift of the adjusted reference temp. approximately 50 $^{0}$F is 3EFPY
(4) 3/4 service life is 24EFPY
(5) Time interval between first and fourth capsule withdrawal is 21 EFPY therefore 1/3 and 2/3 of this interval is 7 and 14 EFPY respectively which yields a withdrawal at 10 and 17 EFPY
(6) Fifth and sixth capsules are spares to monitor subsequent anealing/irradiation
(7) Startup on 12/77 except 1½ year delay assumed
(8) 2772/2535 x 2.8 = 3.0
(9) Analysis of TMI-1 capsule 1 and other capsules from other B&W-177 plants showed an actual lead factor of 2.8 for the 177 fuel assembly plants
(10) Assumed to be zero for TMI-1

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TABLE II

TMI SITE INTEGRATED RV SURVEILLANCE SCHEDULE

| | | 1 | 2 | 3 | 4 | 5 | 6 | 7 | 8 | 9 | 10 | 11 | 12 | 13 | 14 | [illegible] |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| TMI-2 Cycle No. and Start Date | | 1 / 7/79 | 2 / 7/81 | 3 / 11/82 | 4 / 1/84 | 5 / 3/85 | 6 / 5/86 | 7 / 7/87 | 8 / 9/88 | 9 / 11/89 | 10 / 1/91 | 11 / 3/92 | 12 / 5/93 | 13 / 7/94 | 14 / 9/95 | [illegible] |
| TMI-2 Cycle Length (EFPY)/Elapsed Time From S/U (Mos.) | | 1.2 / 24 | 0.8 / 40 | 0.7 / 54 | 0.7 / 68 | 0.7 / 82 | 0.7 / 96 | 0.7 / 110 | 0.7 / 124 | 0.7 / 138 | 0.7 / 152 | 0.7 / 166 | 0.7 / 180 | 0.7 / 194 | 0.7 / 208 | 0.7 / 222 |
| RV Exposure at Begin Cycle (EFPY) | TMI-1 | 3.9(1) | 5.5 | 6.6 | 7.5 | 8.4 | 9.4 | 10.3 | 11.2 | 12.2 | 13.1 | 14.0 | 15.0 | 15.9 | 16.8 | 17.8 |
| | TMI-2 | 0 | 1.2 | 2 | 2.7 | 3.4 | 4.1 | 4.8 | 5.5 | 6.2 | 6.9 | 7.6 | 8.3 | 9.0 | 9.7 | 10.4 |
| TMI-1 Capsule Effective Exposure at Begin Cycle Lead Factor = 3.0 (EFPY) | 1 | NA | CAPSULE | TESTED | | | | | | | | | | | | |
| | 2 | 0** | 3.6 | 6.0 | 8.1 | 10.2* | NA | TESTED | | | | | | | | |
| | 3 | 0** | 3.6 | 6.0 | 8.1 | 10.2 | 12.3 | 14.4 | 16.5* | | | | | | | |
| | 4 | 0 | 0** | 2.4 | 4.5 | 6.6 | 8.7 | 10.8 | 12.9 | 15.0 | 17.1 | 19.2 | 21.3 | 23.4* | NA TESTED | |
| | 5 | 0 | 0 | 0 | 0 | 0** | 2.1 | 4.2 | 6.3 | 8.4 | 10.5 | 12.6 | 15.7 | 17.8 | 19.9 | 22.0(2) |
| | 6 | 0 | 0 | 0 | 0 | 0** | 2.1 | 4.2 | 6.3 | 8.4 | 10.5 | 12.6 | 15.7 | 17.8 | 19.9 | 22.0(2) |
| TMI-2 Capsule Effective Exposure at Begin Cycle Lead Factor = 2.8 (EFPY) | 1 | 0** | 3.4* | NA CAPSULE TESTED | | | | | | | | | | | | |
| | 2 | 0** | 3.4 | 5.6 | 7.6 | 9.5* | NA | TESTED | | | | | | | | |
| | 3 | 0** | 3.4 | 5.6 | 7.6 | 9.5 | 11.5 | 13.4 | 15.4 | 17.4* | NA | TESTED | | | | |
| | 4 | 0** | 3.4 | 5.6 | 7.6 | 9.5 | 11.5 | 13.4 | 15.4 | 17.4 | 19.3 | 21.3 | 23.2* | NA | TESTED | |
| | 5 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0** | 2.0 | 3.9 | 5.9 | 7.8 | 9.8 | 11.8 | 13.7(2) |
| | 6 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0** | 2.0 | 3.9 | 5.9 | 7.8 | 9.8 | 11.8(2) |
| TMI-1 Capsule Nos. Installed | | 2,3 | 2,3,4 | | → | 3,4,5,6 | | → | 4.5.6 | | | | → | 5,6 | → | (2) |
| TMI-2 Capsule Nos. Installed | | 1,2,3,4 | 2,3,4 | | → | 3,4 | | → | 3,4,5 | 4,5,6 | | → | 5,6 | | → | (2) |

(1) Exposure begins 7/79, 1.3 from TMI-1 Cycle 1 plus 3.25 yrs @ 0.8 capacity (3/76-7/79) = 3.9 EFPY

(2) Capsules 5 & 6 for both units planned to be removed and stored when exposed to 32 EFPY

* Removed

** Inserted

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## LIST OF TABLES



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loads are introduced by unit load transients, reactor trips, and unit heatup and cooldown operations. The number of thermal and loading cycles are 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 in accordance with specification 4.2.2. 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 $3.1 \times 10^{10}$ n/cm$^2$ sec at the reference design power of 2568 MWt and an integrated exposure of $3.0 \times 10^{19}$ n/cm$^2$ for 40 years operation.(5) The calculated maximum values are $2.2 \times 10^{10}$ n/cm$^2$ sec and $2.2 \times 10^{19}$ 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 $1.7 \times 10^6$ 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 $1.7 \times 10^{18}$ n/cm$^2$ which is based on the $1.7 \times 10^6$ 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 TMI-2 reactor vessel in the core area to achieve an average effective exposure between 1 and 3 times that of the reactor vessel at ¼t.(6) 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.



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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 pressurizer 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, Sections 4.1.2.8 and 4.3.3

(6) BAW-10100A

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## 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-operat  al 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, 10, 17, and 24 effective full power years. 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 shall be irradiated in the TMI-2 reactor vessel in accordance with the schedule given in table 4.2-2.

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.

b. The vessel specimen surveillance program is based on specimen equivalent exposure years of 3, 10, 17, and 24 EFPY referenced to 1/4 t[2]. These times were selected to meet the requirements of Appendix H to 10 CFR 50 for the 32 EFPY service life of the reactor vessel.

The planned withdrawal schedule is based on a TMI-1 specimen lead factor of 3.0 when installed in the TMI-2 reactor vessel surveillance capsules holder tubes. The schedule provides periodic exposure of the various capsules such that a lead factor between 1 and 3 as required by Appendix to 10 CFR 50 is obtained. Provision has also been made to monitor the effects of annealing and subsequent exposure should it ever be necessary. Therefore, the surveillance program provides sufficient data to substantiate the conservatism of the pressurization, heatup, and cooldown limits.

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

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TABLE 4.2-2
THREE MILE ISLAND NUCLEAR STATION SITE INTEGRATED
REACTOR VESSEL SURVEILLANCE PROGRAM

| Required Capsule exposure when withdrawn | Capsule No. TMI-1 | | Capsule No. TMI-2 | |
|---|---|---|---|---|
| | inserted | withdrawn | inserted | withdrawn |
| 0 EFPY(1) | 2,3 | | 1,2,3,4 | |
| ~3 EFPY | 4 | | | 1 |
| ~ 10 EFPY | 5,6 | 2 | | 2 |
| ~ 17 EFPY | | 3 | 5,6 | 3 |
| ~ 24 EFPY | | 4 | | 4 |
| ~ 32 EFPY | | 5.6 | | 5,6 |

(1) initial installation at the beginning of TMI-2 cycle 1