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. 32

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 23rd day of March, 1976.

Lawrence L. Lawyer
Notary Public

My Commission Expires Nov. 19, 1979

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

TECHNICAL SPECIFICATION CHANGE REQUEST NO. 32

The licensee requests that the attached Technical Specification proposed changed pages replace pages 3-3, 3-4, 3-5, 4-11 and 4-13 of the existing Technical Specifications.

REASON FOR PROPOSED CHANGE

As a result of damage to the reactor vessel surveillance capsule holder tubes (see attached Surveillance Holder Tube Report) it is necessary to remove these tubes and therefore the surveillance capsules to permit safe Cycle 2 operation. Cycle 2 operation without removal of these tubes would increase the probability of loose parts and thereby create the potential for Reactor Internals damage given the extent of Cycle 1 tube damage. Removal of the surveillance capsules prior to and for the duration of Cycle 2 requires an exemption to paragraph II.C.2 of 10CFR50 Appendix H. Part of this proposed change is therefore in response to the Commission's desire that the technical specifications address the fact that the surveillance capsules are to be removed for Cycle 2 operation but must be re-installed prior to Cycle 3 operation.

As a separate issue, our present specifications do not conform to the requirements of Appendix H for withdrawals subsequent to the first capsule withdrawal (since Appendix H was issued after TMI-1 technical specifications were issued). Therefore, this change will also serve to revise the withdrawal times to conform to Appendix H and to clarify other specifications.

In light of the two separate issues explained above, we have determined that this proposed Technical Specification change is necessary.

SAFETY EVALUATION JUSTIFYING CHANGE

Appendix H to 10CFR50 requires for the TMI-1 reactor vessel that the first surveillance capsule be withdrawn during the refueling outage most closely approaching the time the capsule specimens reach an exposure equivalent to the reactor vessel exposure when the reference temperature is expected to shift by $50^{\circ}F$. This shift is predicted to occur at about 3EFPY* exposure (based on high copper content weld material) to the specimens or at about the first refueling outage.** Our present specifications require withdrawal at the refueling outage nearest 2 calendar years which was intended to correspond to approximately 4EFPY*** of specimen exposure and is sufficiently consistent with this refueling outage at 1.8 calendar years from April 1974 and the approximately 3EFPY* present specimen expsoure. In addition, the future withdrawal times specified in this proposed change were chosen to comply with the requirements of Appendix H.

Specification 3.1.2.4 has been changed to clarify that the specified two years means <u>2EFPY</u> of reactor vessel exposure which is consistent with the intent of the present specification and consistent with standard Technical Specifications.

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* Referenced to 1/4t.
** Note: First cycle=1.3EFPY; 1.3x2.4(i.e. lead factor from specimen to ¼t)=3.12
*** Note: First cycle predicted to be 1.6EFPY; 1.6x2.4=3.84

Removal of all specimens prior to and for the duration of Cycle 2 is justified in that all specimens to be reinstalled prior to Cycle 3 have been exposed to the equivalent of greater than 3EFPY referenced to 1/4t and approximately 2.2EFPY referenced to the vessel inner wall. The combined length of Cycle 1 and 2 is about 2.1EFPY, therefore, the specimens will have a cumulative exposure, at beginning of Cycle 3, which is between 1 and 3 times that of the vessel inner wall as required by Appendix H.

Although our Reactor vessel surveillance program differs from our original plans we find that our proposed actions and these proposed specifications accomplish the intent of 10CFR50 Appendix H. Additional amplifying information justifying this change is contained in the attached Surveillance Holder Tube Report.

Based on the above we conclude that this change does not represent a threat to the health and safety of the public and does not represent an unreviewed safety question.

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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.

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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.

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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 $50^{o}F$ 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

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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-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 may 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.

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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

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