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METROPOLITAN EDISON COMPANY
JERSEY CENTRAL POWER & LIGHT COMPANY

AND

PENNSYLVANIA ELECTRIC COMPANY

THREE MILE ISLAND NUCLFAR STATION UNIT 1

Operating License No. DPR-50
Docket No. 50-289
Technical Specification Change Request No. 43, Amendment 1

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 RC Arnold
Vice President-Generation

Sworn and subscribed to me this ____________ day of ________________, 1977.

____
Notary Public

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Three Mile Island Nuclear Station Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289

Technical Specification Change Request No. 43, Amendment 1

The licensee requests that the attached pages replace existing pages or be added to Appendix A of the TMI-1 Technical Specifications.

Reason For Proposed Change

To incorporate steam generators operating and inservice inspection requirements into the TMI-1 Technical Specifications as requested by the USNRC in their letters of September 14, 1976, and December 14, 1976, to Mr. R. C. Arnold.

Safety Analysis Justifying Change

This proposed Technical Specification Change does not involve any unreviewed safety questions in that it only incorporates additional operating and inspection restrictions to ensure the continued integrity of the tube portion of the TMI-1 Once-Through Steam Generators.

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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. Power operation with one idle reactor coolant pump in each loop shall be restricted to 24 hours. If the reactor is not returned to an acceptable RC pump operating combination at the end of the 24-hour period, the reactor shall be in a hot shutdown condition within the next 12 hours.

c. 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. Both steam generators 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.

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Bases

The limitation on power operation with one idle RC pump in each loop has been imposed since the ECCS cooling performance has not been calculated in accordance with the Final Acceptance Criteria requirements specifically for this mode of reactor operation. A time period of 24 hours is allowed for operation with one idle RC pump in each loop to effect repairs of the idle pump(s) and to return the reactor to an acceptable combination of operating RC pumps. The 24 hours for this mode of operation is acceptable since this mode is expected to have considerable margin for the peak cladding temperature limit and since the likelihood of a LOCA within the 24-hour period is considered very remote.

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]

Both steam generators must be operable before heatup of the Reactor Coolant System to insure system integrity against leakage under normal and transient conditions. Only one steam generator is required for decay heat removal purposes.

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

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3.1.6 LEAKAGE

Applicability

Applies to reactor coolant leakage from the reactor coolant system and the makeup and purification system.

Objective

To assure that any reactor coolant leakage does not compromise the safe operation of the facility.

Specification

3.1.6.1 If the total reactor coolant leakage rate exceeds 10 gpm, the reactor shall be placed in hot shutdown within 24 hours of detection.

3.1.6.2 If unidentified reactor coolant leakage (excluding normal evaporative losses) exceeds one gpm or if any reactor coolant leakage is evaluated as unsafe, the reactor shall be placed in hot shutdown within 24 hours of detection.

3.1.6.3 If primary-to-secondary leakage through the steam generator tubes exceeds 1 gpm total for both steam generators, the reactor shall be placed in cold shutdown within 36 hours of detection.

3.1.6.4 If any reactor coolant leakage exists through a nonisolable fault in an RCS strength boundary (such as the reactor vessel, piping, valve body, etc., except the steam generator tubes), the reactor shall be shutdown, and cooldown to the cold shutdown condition shall be initiated within 24 hours of detection.

3.1.6.5 If reactor shutdown is required by Specification 3.1.6.1, 3.1.6.2, 3.1.6.3, or 3.1.6.4, the rate of shutdown and the conditions of shutdown shall be determined by the safety evaluation for each case and reported as required by Specification 6.7.

3.1.6.6 Action to evaluate the safety implication of reactor coolant leakage shall be initiated within four hours of detection. The nature, as well as the magnitude, of the leak shall be considered in this evaluation. The safety evaluation shall assure that the exposure of offsite personnel to radiation is within the guidelines of 10 CFR 20.

3.1.6.7 If reactor shutdown is required per Specification 3.1.6.1, 3.1.6.2, 3.1.6.3, or 3.1.6.4, the reactor shall not be restarted until the leak is repaired or until the problem is otherwise corrected.

3.1.6.8 When the reactor is critical and above 2 percent power, two reactor coolant leak detection systems of different operating principles shall be in operation for the Reactor Building with one of the two systems sensitive to radioactivity. The systems sensitive to radioactivity may be out-of-service for no more than 72 hours provided a sample is taken of the Reactor Building atmposphere every eight hours and analyzed for radioactivity and two other means are available to detect leakage.

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3.1.6.9 Loss of reactor coolant through reactor coolant pump seals and system valves to connecting systems which vent to the gas vent header and from which coolant can be returned to the reactor coolant system shall not be considered as reactor coolant leakage and shall not be subject to the consideration of Specifications 3.1.6.1, 3.1.6.2, 3.1.6.3, 3.1.6.4, 3.1.6.5, 3.1.6.6 or 3.1.6.7, except that such losses when added to leakage shall not exceed 30 gpm. If leakage plus losses exceeds 30 gpm, the reactor shall be placed in hot shutdown within 24 hours of detection.

Bases

Any leak of radioactive fluid, whether from the reactor coolant system primary boundary or not, can be a serious problem with respect to in-plant radioactive contamination and required cleanup or, in the case of reactor coolant, it could develop into a still more serious problem and, therefore, the first indications of such leakage will be followed up as soon as practical. The unit's makeup system has the capability to makeup considerably more than 30 gpm of reactor coolant leakage.

Water inventory balances, monitoring equipment, radioactive tracing, boric acid crystalline deposits, and physical inspections can disclose reactor coolant leaks.

Although some leak rates on the order of gallons per minute may be tolerable from a dose point of view, it is recognized that leaks in the order of drops per minute through any of the barriers of the primary system could be indicative of materials failure such as by stress corrosion cracking. If depressurization, isolation, and/or other safety measures are not taken promptly, these small leaks could develop into much larger leaks, possibly into a gross pipe rupture. Therefore, the nature and location of the leak, as well as the magnitude of the leakage, must be considered in the safety evaluation.

When reactor coolant leakage occurs to the Reactor Building, it is ultimately conducted to the Reactor Building sump. Although the reactor coolant is safely contained, the gaseous components in it escape to the Reactor Building atmosphere. There, the gaseous components become a potential hazard to plant personnel, during inspection tours within the Reactor Building, and to the general public whenever the Reactor Building atmosphere is periodically purged to the environment.

When reactor coolant leakage occurs to the Auxiliary Building, it is collected in the Auxiliary Building sump. The gases escaping from reactor coolant leakage within the Auxiliary Building will be collected in the Auxiliary and Fuel Handling Building exhaust ventilation system and discharged to the environment via the unit's Auxiliary and Fuel Handling Building vent. Since the majority of this leakage occurs within confined, separately ventilated cubicles within the Auxiliary Building, it incurs very little hazard to plant personnel.

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4.18 OTSG TUBE INSERVICE INSPECTION

Applicability

This Technical Specification applies to the inservice inspection of the OTSG tube portion of the reactor coolant pressure boundary.

Objective

The objective of this inservice inspection program is to provide assurance of continued integrity of the tube portion of the Once-Through Steam Generators, while at the same time minimizing radiation exposure to personnel in the performance of the inspection.

Specification

Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 3.1.6.3.

4.18.1 Steam Generator Sample Selection and Inspection

Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.18.1 at the frequency specified in 4.18.3.

4.18.2 Steam Generator Tube Sample Selection and Inspection

The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.18.2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.18.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.18.4. The tubes selected for

each inservice inspection shall include at least 1½% of the total number of tubes in all steam generators:

a. The first inservice inspection of each steam generator included a random sampling of greater than 3% of the tubes in each steam generator.

b. The second and subsequent inservice inspection may be from one steam generator and shall include 1½% of the total installed steam generator heat transfer surface. All tubes to be examined in the second and subsequent intervals shall be from previously examined tubes. All unplugged tubes with indications greater than 20% of the nominal wall thickness shall be included in those tubes selected for reexamination.

The results of each sample inspection shall be classified into one of the following two categories:

Category	Inspection Results
C-1	Less than 10% of the total tubes inspected in a steam generator are degraded tubes exhibiting indications in excess of 20% of the wall thickness.
C-2	10% or more of the total tubes inspected in a steam generator are degraded tubes exhibiting indications in excess of 20% of the wall thickness.

NOTE: In all inspections, previously degraded tubes must exhibit significant (> 10%) further wall penetrations to be included in the above percentage calculations.

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4.18.3 Inspection Frequencies

The required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first (baseline) inspection was performed after 6 effective full power months but within 24 calendar months of initial criticality. The second inservice inspection shall be performed not less than 9 nor more than 24 calendar months after the previous inspection. The third and subsequent inspections shall be performed within additional periods of 3-1/3 years.

b. If, in any inspection, an excess of 10% of the tubes examined exhibit indications in excess of 20% of the wall thickness, the next two inspections shall be performed at one to two year intervals. If, in these examinations, no more than 10% of the tubes examined exhibit either (a) additional degradation (greater than 10% of wall thickness) of previously degraded tubes, (b) tubes with new indications in excess of 20% wall thickness, or (c) a combination of both, the inspection intervals may continue at 3-1/3 years.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.18.2 during the shutdown subsequent to any of the following conditions:

   1. Primary-to-secondary leaks which exceed the limits of Specification 3.1.6.3.

   2. A seismic occurrence greater than the Operating Basis Earthquake.

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3. A loss of coolant accident requiring actuation of the engineering safeguards, or

4. A major main steam line or feedwater line break.

4.18.4 Acceptance Criteria

a. As used in this Specification:

1. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

3. Degraded Tube means a tube containing imperfections $\geq$20% of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service because it may become unserviceable prior to the next inspection and is equal to 40% of the nominal tube wall thickness, unless higher limits are shown to be acceptable by analysis.

7. Tube Inspection means an inspection of that portion of the steam generator tube from the bottom of the upper tubesheet completely to the top of the lower tubesheet.

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b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plugging including all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.18.2.

4.18.5 Reports

a. Following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission within 15 days.

b. The complete results of the steam generator tube inservice inspection shall be included in the Annual Operating Report for the period in which this inspection was completed. This report shall include:

   1. Number and extent of tubes inspected.

   2. Location and percent of wall-thickness penetration for each indication of an imperfection.

   3. Identification of tubes plugged.

c. Results of steam generator tube inspections which require prompt notification of the Commission shall be reported pursuant to Specification 6.9.2 prior to resumption of plant operation. The written followup of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

Bases

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained.

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The program for inservice inspection of steam generator tubes is based on ASME Code Section XI, Winter 1975 Addenda. Inservice Inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The unit is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these chemistry limits, localized corrosion may likely result in stress corrosion cracking.

The extent of steam generator tube leakage due to cracking would be limited by the secondary coolant activity, Specification 3.1.6.3. Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents.

The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 1 gpm). Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

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Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect would develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for degradation equal to or in excess of 40% of the tube nominal wall thickness, unless higher limits are shown to be acceptable by analysis. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into category C-2 on the 3rd sample inspection, (See Table 4.18.2), these results will be promptly reported to the Commission pursuant to Specification 6.9.2 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy current inspection, and revision of the Technical Specifications, if necessary.

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TABLE 4.18.1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	None
No. of Steam Generators per unit	Two
First Inservice Inspection	Two
Second & Subsequent Inservice Inspections	One[1]

<u>TABLE NOTATION</u>:

1. The Inservice Inspection may be limited to one steam generator on a rotating schedule encompassing 3% of the tubes in that steam generator if the results of the first and subsequent inspections indicate that both steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one steam generator may be found to be more severe than those in the other steam generator. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

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TABLE 4.18.2

STEAM GENERATOR TUBE INSPECTION

1st SAMPLE INSPECTION			2nd SAMPLE INSPECTION		3rd SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of 1½% of total SG Tubes (If both SG's inspected, inspect 1½% of each SG's Tubes If only one SG inspected, inspect 3% of that SG's Tubes)	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes. Inspect additional 3% of tubes in each SG	C-1	None	N/A	N/A
			C-2	Plug defective tubes. Inspect additional 3% of tubes in each SG	C-1	None
					C-2	Plug defective tubes. Notify NRC pursuant to Specification 6.9.2.

NOTE: 1st Sample Inspection as used in this table refers to the required inspections at the frequencies listed in Specification 4.18.13.

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

DOCKET NO. 50-289
LICENSE NO. DPR-50

METROPOLITAN EDISON COMPANY

This is to certify that a copy of Technical Specification Change Request. No. 43, Amendment 1, to Appendix A of the Operating License for Three Mile Island Nuclear Station Unit 1, has, on the date given below, been filed with the U. S. Nuclear Regulatory Commission and been served on the chief executives of Londonderry Township, Dauphin County, Pennsylvania and Dauphin County, Pennsylvania by deposit in the United States mail, addressed as follows:

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METROPOLITAN EDISON COMPANY

By RC Arnold
Vice President-Generation

Dated: 1-7-7

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