1-16-76

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. 31

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 ______________________
Vice President-Generation

Sworn and subscribed to me this ______ day of ______________,

____
Notary Public

NOTARY PUBLIC
My Commission Expires Nov. 19, 1979

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METROPOLITAN EDISON COMPANY
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THREE MILE ISLAND NUCLEAR STATION UNIT 1

Operating License No. DPR-50
Docket No. 50-289
Technical Specification Change Request No. 31

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 /s/ R. C. Arnold
Vice President-Generation

Sworn and subscribed to me this 16th day of January, 1976

Lawrence L. Lawyer
Notary Public

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Regulatory Docket File

1-16-76

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

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

METROPOLITAN EDISON COMPANY

This is to certify that a copy of Technical Specification Change Request No. 31 to Appendix A of the Operating License for Three Mile Island Nuclear Station, Unit 1, dated January 16, 1976, and filed with the U.S. Nuclear Regulatory Commission January 16, 1976, has this 16th day January, 1976, been served on the chief executives of Londonderry Township, Dauphin County, Pennsylvania, and of Dauphin County, Pennsylvania, by deposit in the United States Mail, addressed as follows:

Mr. Weldon B. Arehart, Chairman
Board of Supervisors of
Londonderry Township
R.D. #1, Geyers Church Road
Middletown, Pennsylvania 17057

Mr. Charles P. Hoy, Chairman
Board of County Commissioners of
Dauphin County
Dauphin County Courthouse
Harrisburg, Pennsylvania 17120

METROPOLITAN EDISON COMPANY

By R C Arnold
Vice President-Generation

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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. 31

The licensee requests that the attached changed pages replace pages 2-5, 2-6, 2-9, and table 2.3-1 of the existing technical specifications.

Reasons for Proposed Change

Metropolitan Edison is currently involved in a program with Babcock and Wilcox to improve the capability of the TMI-1 plant to withstand a loss of electrical load (LOEL) from 100% power without tripping the reactor.

The ability to ride through a LOEL without tripping the reactor is important for several reasons. In the first place, if the reactor trips the plant cannot resume power generation for a considerable length of time following the load loss. In order to ensure continuity of electric power supplied to the public it is desirable for the plant to be able to pick up load again as soon as possible after the initial interruption. In the second place, if the reactor trips following a separation of the station from the grid, emergency sources of power must be relied upon to run vital station auxiliaries. If reactor trip is prevented, the plant can continue to supply all auxiliaries in the normal fashion.

In order to improve the ability of the TMI-1 plant to avoid reactor trip on a loss of load, Babcock and Wilcox has recommended several relatively minor plant modifications. For the most part, the modifications are aimed at increasing the effectiveness of the secondary plant steam relief systems and improving the response of the steam generator controls. In addition to these changes, however, B&W has recommended an increase in the high reactor coolant pressure reactor trip setpoint.

The attached supplementary safety analysis justifies a revised trip setting of 2405 psig. Accordingly, Met-Ed ultimately intends to increase the high pressure reactor trip setting from 2355 psig to 2405 psig on a permanent basis.

In addition, in order to preserve the margin between the high pressure trip setting and the setting of the pressurizer code safety valves, Met-Ed intends to increase the set pressure of the safety valves, at the same time the permanent change to the high pressure trip setting is made. Preliminary study indicates that unnecessary conservatism can be removed from the calculation which was used to determine the current safety valve setting of 2435 psig, with the result that a revised setting of 2500 psig can be implemented without compromising the plant's protection from overpressure. A final analysis justifying an increased safety valve setpoint will be completed in the near future, at which time a request for the revised safety valve setting and a permanent reactor trip setting of 2405 psig will be submitted. If approved, these changes will be made during the refueling outage scheduled to commence in February of this year.

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Just prior to the refueling outage in February, Met-Ed intends to perform a test of the loss of electrical load transient at TMI-1. The purpose of the test is to obtain sufficient data on the LOEL transient to evaluate the effectiveness of recent plant modifications in improving plant response to a LOEL. In order to ensure that the test is successful, i.e., data is obtained during a successful runback from 100% to 15% power without reactor trip, it is desired to raise the high pressure reactor trip setting to 2405 psig just prior to the test. However, it is not practical to increase the pressurizer safety valve setting before the test. Therefore, Met-Ed requests approval for a change in the trip setting for the purpose of the test. A request for permanent increased settings of both the reactor trip and the code safety valves will be submitted in the near future, to be implemented during the refueling outage as discussed above.

Safety Analysis Justifying Change

The attached supplementary safety analysis provides justification for the revised high pressure reactor trip setpoint of 2405 psig. The analysis indicates that, with the revised trip setting, for the end of life conditions which will exist for the LOEL test, reactor coolant system pressure as well as other critical plant parameters are held below the safety limits for the most limiting accidents treated in the TMI-1 FSAR. (It also shows that for all conditions which will be encountered in fuel cycle ?, critical parameters will be maintained below these limits.) Note that the accidents analysed in Section 14.1.2.3 of the FSAR, Rod Withdrawal from Rated Power, and Section 14.1.2.4, Moderator Dilution, have less severe consequences and, therefore, have not been specifically analyzed in the attachment. The Loss of Electric Power treated in FSAR Section 14.1.2.8 also has less severe consequences and requires no further analysis.

Since the pressurizer code safety valve setpoint must remain at the current setting of 2435 psig during the LOEL test, it is conceivable that some postulated accident could cause lifting of the safety valves prior to the reactor coolant pressure reaching the revised high pressure reactor trip setting. Our analysis of the consequences of such an accident indicate that no hazard to public safety would result. However, to provide additional assurance in this regard, special precautions will be taken during the period of time the trip setting is raised to protect against any possible adverse consequences of lifting the safety valves prior to high pressure trip.

With regard to the LOEL transient itself, increasing the reactor trip setting to 2405 psig will not present a risk of reactor coolant system overpressure during the test. Data obtained from previous full power load rejections show that the pressure reached following the LOEL did not appreciably overshoot the present trip point (2355 psig). Raising the high pressure trip setting to 2405 psig can increase the peak pressure by at most 50 psi (the amount of the proposed increase in the trip setting). Therefore, the maximum reactor coolant pressure will still be limited to well below the plant design pressure.

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTION INSTRUMENTATION

Applicability

Applies to instruments monitoring reactor power, reactor power imbalance, reactor coolant system pressure, reactor coolant outlet temperature, flow, number of pumps in operation, and high reactor building pressure.

Objective

To provide automatic protection action to prevent any combination of process variables from exceeding a safety limit.

Specifications

2.3.1 The reactor protection system trip setting limits and the permissible bypasses for the instrument channels shall be as stated in Table 2.3-1 and Figure 2.3-2.

2.3.2 For the 24 hour period prior to shutdown for refueling the high pressure trip setpoint may be increased to 2405 psig provided that the following precautions are taken during this period.

a. If pressurizer level exceeds 315 inches as indicated by the high-high level alarm, the reactor shall be manually tripped.

b. If the reactor drain tank pressure exceeds 15 psig, the reactor shall be manually tripped.

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Bases

The reactor protection system consists of four instrument channels to monitor each of several selected plant conditions which will cause a reactor trip if any one of these conditions deviates from a pre-selected operating range to the degree that a safety limit may be reached.

The trip setting limits for protection system instrumentation are listed in Table 2.3-1. The safety analysis has been based upon these protection system instrumentation trip set points plus calibration and instrumentation errors.

Nuclear Overpower

A reactor trip at high power level (neutron flux) is provided to prevent damage to the fuel cladding from reactivity excursions too rapid to be detected by pressure and temperature measurements.

During normal plant operation with all reactor coolant pumps operating, reactor trip is initiated when the reactor power level reaches 105.5% of rated power. Adding to this the possible variation in trip set points due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which is more conservative than the value used in the safety analysis(1).

a. Overpower trip based on flow and imbalance

   The power level trip set point produced by the reactor coolant system flow is based on a power-to-flow ratio which has been established to accommodate the most severe thermal transient considered in the design, the loss-of-coolant flow accident from high power. Analysis has demonstrated that the specified power to flow ratio is adequate to prevent a DNBR of less than 1.3 should a low flow condition exist due to any malfunction.

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The power level trip set point produced by the power-to-flow ratio provides both high power level and low flow protection in the event the reactor power level increases or the reactor coolant flow rate decreases. The power level trip set point produced by the power to flow ratio provides overpower DNB protection for all modes of pump operation. For every flow rate there is a maximum permissible power level, and for every power level there is a minimum permissible low flow rate. Typical power level and low flow rate combinations for the pump situations of Table 2.3-1 are as follows:

1. Trip would occur when four reactor coolant pumps are operating if power is 108 percent and reactor flow rate is 100 percent, or flow rate is 92.6 percent and power level is 100 percent.

2. Trip would occur when three reactor coolant pumps are operating if power is 80.7 percent and reactor flow rate is 74.7 percent or flow rate is 69.2 percent and power level is 75 percent.

3. Trip would occur when one reactor coolant pump is operating in each loop (total of two pumps operating) if the power is 52.9 percent and reactor flow rate is 49.0 percent or flow rate is 45.4 percent and the power level is 49 percent.

For safety analysis calculations the maximum calibration and instrumentation errors for the power level were used.

The power-imbalance boundaries are established in order to prevent reactor thermal limits from being exceeded. These thermal limits are either power peaking kW/ft limits or DNBR limits. The reactor power imbalance (power in the top half of core minus power in the bottom half of core) reduces the power level trip produced by the power-to-flow ratio so that the boundaries of Figure 2.3-2 are produced. The power-to-flow ratio reduces the power level trip and associated reactor-power reactor-power-imbalance boundaries by 1.08 percent for a one percent flow reduction.

b. Pump monitors

The redundant pump monitors prevent the minimum core DNBR from decreasing below 1.3 by tripping the reactor due to the loss of reactor coolant pump(s). The pump monitors also restrict the power level for the number of pumps in operation.

c. Reactor coolant system pressure

During a startup accident from low power or a slow rod withdrawal from high power, the system high pressure trip set point is reached before the nuclear overpower trip set point. The trip setting limit shown in Figure 2.3-1 for high reactor coolant system pressure (2355*psig) has been established to maintain the system pressure below the safety limit (2750 psig) for any design transient.

* Except as specified in 2.3.2

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TABLE 2.3-1

REACTOR PROTECTION SYSTEM TRIP SETTING LIMITS

	Four Reactor Coolant Pumps Operating (Nominal Operating Power - 100%)	Three Reactor Coolant Pumps Operating (Nominal Operating Power - 75%)	One Reactor Coolant Pump Operating in Each Loop (Nominal Operating Power - 49%)	Shutdown Bypass
1. Nuclear power, Max. % of rated power	105.5	105.5	105.5	5.0(3)
2. Nuclear power based on flow(2) and imbalance, max. of rated power	1.08 times flow minus reduction due to imbalance(s)	1.08 times flow minus reduction due to imbalance(s)	1.08 times flow minus reduction due to imbalance(s)	Bypassed
3. Nuclear power based(5) on pump monitors, max. % of rated power	NA	NA	91%	Bypassed
4. High reactor coolant system pressure, psig, max.	2355 *	2355 *	2355 *	1720(4)
5. Low reactor coolant system pressure, psig, min.	1800	1800	1800	Bypassed
6. Variable low reactor coolant system pressure, psig, min.	$(16.25T_{out} - 7756)^{(1)}$	$(16.25T_{out} - 7756)^{(1)}$	$(16.25T_{out} - 7756)^{(1)}$	Bypassed
7. Reactor coolant temp. F., Max.	619	619	619	619
8. High Reactor Building pressure, psig, max.	4	4	4	4

(1) T_{out} is in degrees Fahrenheit (F)

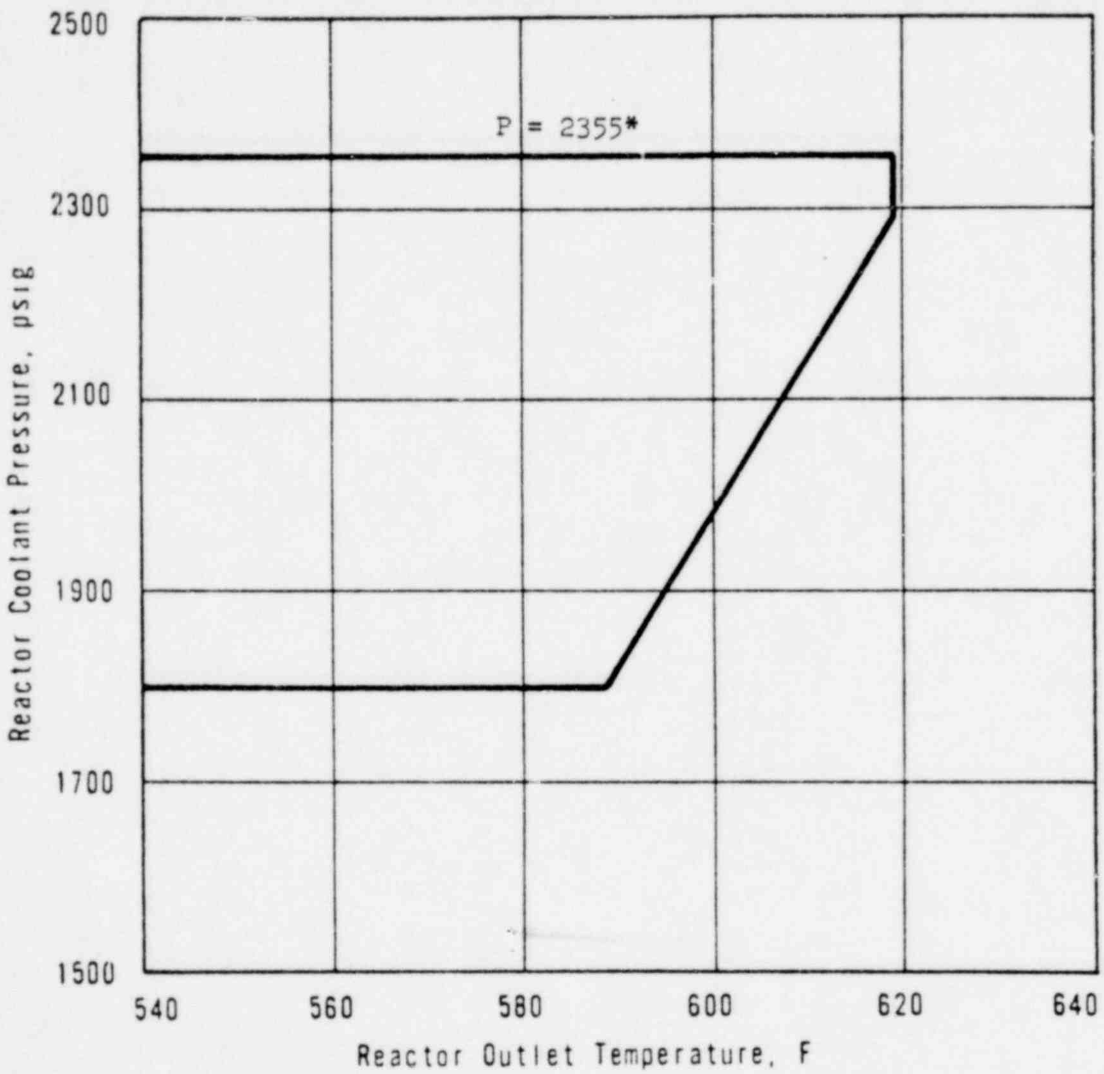
* Except as specified in 2.3.2

(2) Reactor coolant system flow, %

(3) Administratively controlled reduction set on during reactor shutdown

(4) Automatically set when other segments of the RPS (as specified) are bypassed

(5) The pump monitors also produce a trip on: (a) loss of two reactor coolant pumps in one reactor coolant loop, and (b) loss of one or two reactor coolant pumps during two-pump operation.



* Except as specified in 2.3.2

PROTECTION SYSTEM MAXIMUM
ALLOWABLE SET POINTS
THREE MILE ISLAND NUCLEAR STATION UNIT 1

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FIGURE 2.3-1