METROPOLITAN EDISON COMPANY
JERSEY CENTRAL POWER & LIGHT COMPANY

AND

PENNSYLVANIA ELECTRIC COMPANY

THREE MILE ISLAND NUCLEAR STATION UNIT

Operating License No. DPR-73
Docket No. 50-320
Technical Specification Change Request No. 014

This Technical Specification Change Request is submitted in support of Licensee's request to change Appendix A to Operating License No. DPR-73 for Three Mile Island Nuclear Station Unit 2. As a part of this request, proposed replacement pages for Appendix A are also included.

METROPOLITAN EDISON COMPANY

By H. D. Herbein
Vice President

Sworn and subscribed to me this 7th day of July, 1978.

George J. Troffey
Notary Public

GEORGE J. TROFFEY
Notary Public, Reading, Berks Co.
My Commission Expires Jan. 25, 1982

7904200 129

57-201

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

DOCKET NO. 50-320
LICENSE NO. DPR-73

METROPOLITAN EDISON COMPANY

This is to certify that a copy of Technical Specification Change Request No. 014 to Appendix A of the Operating License for Three Mile Island Nuclear Station Unit 2, 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:

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

Mr. Harry B. Reese, Jr.
Board of County Commissioners
of Dauphin County
Dauphin County Court House
Harrisburg, Pennsylvania 17120

METROPOLITAN EDISON COMPANY

By J. G. Herbein
Vice President

Dated: July 7, 1978

57 202

Three Mile Island Nuclear Station Unit 2 (TMI-2)
Operating License No. DPR-73
Docket No. 50-320

Technical Specification Change Request

The licensee requests that the attached changed pages replace 2-5, 2-6, 3/4 2-13, 3/4 3-3, and 3/4 3-13, and that the attached changed figures replace figures 2.1-1, 2.1-2, 2.2-1 and 2.2-2. Also included for your information and use are changed pages B 2-1, B 2-2, B 2-6 and figure 2.1, all of which are part of the "Bases" Section of the TMI-2 Technical Specifications. These pages will be distributed to Technical Specifications (T.S.) copyholders upon approval of this change request. However, in accordance with 10 CFR 50.36, the bases are not considered part of the T.S., and therefore, changes in them do not require your approval.

Reasons for the Change Request

Due to the recent concern over wear encountered in the Fuel Assembly (FA) holddown latch assemblies in units similar to TMI-2, caused by vibration of Burnable Poison Rod Assemblies (BPRAs) and Orifice Rod Assemblies (ORAs), it is believed necessary to install retainers on the BPRAs and on two modified ORAs and to remove the remaining ORAs. Installation of the BPRA retainers reduces hot assembly flow by less than 1% and removal of all 40 ORAs would increase bypass flow by only 1.6%. These effects on the system flow characteristics are very slight and when they are combined with the increased flow presented in the attached proposed changes, the DNBR safety margin is actually increased. Although the present safety margins are adequate to compensate for the changes in flow distribution brought about as a result of these core alterations, we feel that system flow should be increased to ensure even larger DNBR safety margin.

Several changes included in this change request are not related to ORA removal and BPRA retainer installation. The first of these changes is the increase in RCS Pressure - low trip setpoint in Table 2.2-1 from 1800 psig to 1900 psig. This change is being made for greater operating flexibility and to allow for a backup function in case of a small break LOCA although it will most likely never be used because the RCS pressure - variable low should occur first. It is also being made to increase the margin to HPI so that a rapid depressurization will not unnecessarily inject HPI as frequently as would occur with less margin. As a result of this increase in the RCS Pressure - low trip setpoint, it is correspondingly necessary to increase the manual bypass by 100 psi to $\leq$1820 psig to incorporate 1820 psig as the new high pressure trip during startup in shutdown bypass. This will enable startup to be performed more easily and will continue to maintain the same margin previously used to allow for instrument errors. Since the change in RCS Pressure - low trip setpoint is in a conservative direction, it is bounded by previous analyses and therefore does not require an additional safety analysis.

The rod bow penalty which is not directly related to ORA removal has been revised to correctly reflect the NRC Rod Bow Model. The original Technical Specification was prepared using the B&W Rod Bow Model and during investigation into the removal of the ORAs was discovered and is corrected herein. The imbalance curves in Figures 2.2-1 and 2.2-2 are also being corrected to account for an initial misinterpretation of the positive offset error equation.

Since core power and core burnup have never attained levels that could have caused these errors to be of significance, they did not result in operation in a manner less conservative than assumed in the analyses and therefore are of no safety concern.

Also, incorporated herein is the addition of the allowable values in Table 2.2-1 for the Channel Functional Test which previously were not included in the Technical Specifications. These values have been added to account for instrument calibration error, instrument drift and instrument error.

Safety Analysis Justifying Change

As previously indicated, installation of the retainers and removal of the ORAs is necessary due to wear experienced in the FA holddown latch assemblies in other units similar to TMI-2. The BPRA Retainer Design Report BAW-1496, Babcock & Wilcox, Lynchburg, Virginia, May 1978, shows that the retainers are structurally adequate to withstand all expected conditions within the system and their addition creates only a minute (<1%) change in the core flow distribution which is more than offset by the DNBR safety margin.

The attached Justification for Removal of Orifice Rod Assemblies in the Three Mile Island Unit 2, Cycle 1, BAW-1497, Babcock & Wilcox, Lynchburg, Virginia, June 1978, shows that core bypass flow is increased by only 1.6% with all forty ORAs removed. When the removal of all 40 ORAs is combined with the increased reactor coolant system flow presented in the attached changed pages and figures, core safety margins are improved relative to those defined in the TMI-2 FSAR and the TMI-2 Fuel Densification Report, BAW-1455, Babcock & Wilcox, Lynchburg, Virginia, July 1977.

The two ORAs that will be modified and left in the core, held in place by the identical retainers used on the BPRAs are described in the attached letter to S. A. Varga from J. H. Taylor, dated June 7, 1978. This letter and Appendix "B" of BAW-1496 "Retainer Design Report", also justify use of the retainers on a Modified Orifice Rod Assembly (MORA) and indicate that the margin for component lift with the retainer in use is greater than that required in the report. FA growth is less than that considered in the report and all stress and fatigue margins are acceptable.

Since the changes presented above actually enhance the conservative operation of the unit, it has been determined that as a result of these changes:

1. The probability of occurrence for an accident or malfunction of equipment important to safety previously evaluated in the TMI-2 FSAR is not increased.

2. The probable consequences of any accident analyzed in the TMI-2 FSAR is not increased.

3. The potential for the occurrence of an accident not previously analyzed in the TMI-2 FSAR is not introduced.

Amendment Classification (10 CFR 170.22)

Because the proposed changes involve a single safety issue which does not involve a significant hazards consideration, it constitutes a Class III License Amendment (as defined by 10 CFR 170.22). Therefore, enclosed please find a check in the amount of $4,000.00.

Attachments: 1) Justification for Removal of Orifice Rod Assemblies in TMI-2, Cycle 1, BAW-1497 Babcock & Wilcox, Lynchburg, Virginia, June 1978

2) Letter to S. A. Varga from J. H. Taylor, dated June 7, 1978

TABLE 2.2-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Nuclear Overpower	$\leq$ 105.5% of RATED THERMAL POWER with four pumps operating	$\leq$ 105.6% of RATED THERMAL POWER with four pumps operating #
	$\leq$ 78.1% of RATED THERMAL POWER with three pumps operating	$\leq$ 78.2% of RATED THERMAL POWER with three pumps operating #
	$\leq$ 50.9% of RATED THERMAL POWER with one pump operating in each loop	$\leq$ 51.0% of RATED THERMAL POWER with one pump operating in each loop #
3. RCS Outlet Temperature-High	$\leq$ 619°F	$\leq$ 619.08°F #
4. Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBLANCE (1)	Trip Setpoint not to exceed the limit line of Figure 2.2-1.	Allowable Values not to exceed the limit line of Figure 2.2-2. #
5. RCS Pressure-Low (1)	$\geq$ 1900 psig	$\geq$ 1899.0 psig*; $\geq$ 1891.5 psig**
6. RCS Pressure-High	$\leq$ 2355 psig	$\leq$ 2356.0 psig*; $\leq$ 2363.5 psig**
7. RCS Pressure-Variable Low (1)	$\geq$ (13.00 T_{out} °F - 5887) psig	$\geq$ (13.00 T_{out} °F - 5887.64) psig#

8007080368

TABLE 2.2-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTION UNIT	TRIP SETPOINT	ALLOWABLE VALUES
8. Nuclear Overpower based on Pump Monitors(1)	≤ 125% of RATED THERMAL POWER with three pumps operating	≤ 125% of RATED THERMAL POWER with three pumps operating #
	≤ 56.9% of RATED THERMAL POWER with one pump operating in each loop	≤ 57.18% of RATED THERMAL POWER with one pump operating in each loop#
	≤ 0% of RATED THERMAL POWER with two pump operating in one loop and no pump operating in the other loop	≤ 0.28% of RATED THERMAL POWER with two pumps operating in one loop and no pump operating in the other loop#
	≤ 0% of RATED THERMAL POWER with no pumps operating or only one pump operating	≤ 0.28% of RATED THERMAL POWER with no pumps operating or only one pump operating #
9. Reactor Containment Vessel	≤ 4 psig	≤ 4 psig #

(1) Trip may be manually bypassed when RCS pressure ≤ 1820 psig by actuating Shutdown Bypass provided that:

a. The Nuclear Overpower Trip Setpoint is ≤ 5% of RATED THERMAL POWER
b. The Shutdown Bypass RCS Pressure - High Trip Setpoint of ≤ 1820 psig is imposed, and
c. The Shutdown Bypass is removed when RCS Pressure >1900 psig.

*Allowable value for Channel Functional Test
**Allowable value for Channel Calibration
#Allowable value for Channel Functional test and Channel Calibration

2.1.1 and 2.1.2 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime would result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the BAW-2 DNB flux correlation. The DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. This value corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curve presented in Figure 2.1-1 represents the conditions at which a minimum DNBR of 1.30 is predicted for the maximum possible thermal power 112% when the reactor coolant flow is 377,000 gpm, which is 102% of the design flow rate for four operating reactor coolant pumps. This curve is based on the following nuclear power peaking factors with potential fuel densification effects:

$$F_Q^N = 2.67; \quad F_{\Delta H}^N = 1.78; \quad F_Z^N = 1.50$$

The design limit power peaking factors are the most restrictive calculated at full power for the range from all control rods fully withdrawn to minimum allowable control rod withdrawal, and form the core DNBR design basis.

The reactor trip envelope appears to approach the safety limit more closely than it actually does because the reactor trip pressures are measured at a location where the indicated pressure is about 30 psi less than core outlet pressure, providing a more conservative margin to the safety limit.

The curves of Figure 2.1-2 are based on the more restrictive of two thermal limits and include the effects of potential fuel densification:

1. The 1.30 DNBR limit produced by a nuclear power peaking factor of $F_Q^N = 2.67$ or the combination of the radial peak, axial peak and position of the axial peak that yields no less than a 1.30 DNBR.

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limit is 21.0 kw/ft.

Power peaking is not a directly observable quantity and therefore limits have been established on the basis of the reactor power imbalance produced by the power peaking.

The specified flow rates for curves 1, 2, and 3 of Figure 2.1-2 correspond to the expected minimum flow rates with four pumps, three pumps, and one pump in each loop, respectively.

The curve of Figure 2.1-1 is the most restrictive of all possible reactor coolant pump-maximum thermal power combinations shown in BASES Figure 2.1. The curves of BASES Figure 2.1 represent the conditions at which a minimum DNBR of 1.30 is predicted at the maximum possible thermal power for the number of reactor coolant pumps in operation or the local quality at the point of minimum DNBR is equal to 22%, whichever condition is more restrictive.

Using a local quality limit of 22% at the point of minimum DNBR as a basis for curve 3 of BASES Figure 2.1 is a conservative criterion even though the quality at the exit is higher than the quality at the point of minimum DNBR.

The DNBR as calculated by the BAW-2 DNB correlation continually increases from point of minimum DNBR, so that the exit DNBR is always higher. Extrapolation of the correlation beyond its published quality range of 22% is justified on the basis of experimental data.

For each curve of BASES Figure 2.1, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30 or a local quality at the point of minimum DNBR less than 22% for that particular reactor coolant pump situation. The 1.30 DNBR curve for four pump operation is more restrictive than any other reactor coolant pump situation because any pressure/temperature point above and to the left of the four pump curve will be above and to the left of the other curves.

2.1.3 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Boiler and Pressure Vessel Code which permits a maximum transient pressure of 110%, 2750 psig, of design pressure. The Reactor Coolant System piping, valves and fittings, are designed to ANSI B 31.7, 2/68 Edition. Reactor Coolant System valves are designed to ANSI B 16.5-1963, MSSP-61 and MSSP-66. The maximum transient pressure for the Reactor Coolant System valves is permitted by ASME to be 110%, 2750 psig of design pressure. The Safety Limit of 2750 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3125 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

BASES

The AXIAL 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 AXIAL POWER IMBALANCE reduces the power level trip produced by the flux-to-flow ratio such that the boundaries of Figure 2.2-1 are produced. The flux-to-flow ratio reduces the power level trip and associated reactor power-reactor power-imbalance boundaries by 1.05% for a 1% flow reduction.

RCS Pressure - Low, High and Variable Low

The High and Low trips are provided to limit the pressure range in which reactor operation is permitted.

During a slow reactivity insertion startup accident from low power or a slow reactivity insertion from high power, the RCS Pressure-High setpoint is reached before the Nuclear Overpower Trip Setpoint. The trip setpoint for RCS Pressure-High, 2355 psig, has been established to maintain the system pressure below the safety limit, 2750 psig, for any design transient. The RCS Pressure-High trip is backed up by the pressurizer code safety valves for RCS over pressure protection, and is therefore set lower than the set pressure for these valves, 2500 psig. The RCS Pressure-High trip also backs up the Nuclear Overpower trip.

The RCS Pressure-Low, 1900 psig, and RCS Pressure-Variable Low, (13.00 T_{out} °F-5887) psig, Trip Setpoints have been established to maintain the DNB ratio greater than or equal to 1.30 for those design accidents that result in a pressure reduction. It also prevents reactor operation at pressures below the valid range of DNB correlation limits, protection against DNB.

Due to the calibration and instrumentation errors, the safety analysis used a RCS Pressure-Variable Low Trip Setpointof (13.00 T_{out} °F-5927) psig.

Nuclear Overpower Based on Pump Monitors

In conjuction with the power/imbalance/flow trips the Nuclear Overpower Based On Pump Monitors trip prevents the minimum core DNBR from decreasing belwo 1.30 by triping 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.

TABLE 3.2-2

DNB MARGIN

Parameter	LIMITS		
	Four Reactor Coolant Pumps Operating	Three Reactor Coolant Pumps Operating	One Reactor Coolant Pump Operating in Each Loop
Reactor Coolant Hot Leg Temperature, T_H °F	≤ 609.3	≤ 609.3 (1)	≤ 609.3
Reactor Coolant Pressure, psig (2)	≥ 2060.4	≥ 2056.4 (1)	≥ 2091.4
Reactor Coolant Flow Rate, gpm	≥ 377,000	≥ 280,400	≥ 182,800

(1) Applicable to the loop with 2 Reactor Coolant Pumps Operating.

(2) Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step increase of greater than 10% of RATED THERMAL POWER.

TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the control rod drive trip breakers in the closed position and the control rod drive system capable of rod withdrawal.

**When Shutdown Bypass is actuated.

#The provisions of Specification 3.0.4 are not applicable.

##High voltage to detector may be de-energized above 10^{-10} amps on both Intermediate Range channels.

(a) Trip may be manually bypassed when RCS pressure $\leq$ 1820 psig by actuating Shutdown Bypass provided that:

(1) The Nuclear Overpower Trip Setpoint is $\leq$ 5% of RATED THERMAL POWER.

(2) The Shutdown Bypass RCS Pressure--High Trip Setpoint of $\leq$ 1820 psig is imposed.

(3) The Shutdown Bypass is removed when RCS pressure > 1900 psig.

(b) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and/or open the control rod drive trip breakers.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels STARTUP and/or POWER OPERATION may proceed provided all of the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within one hour.

b. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.1,

TABLE 3.3-3 (Continued)

TABLE NOTATION

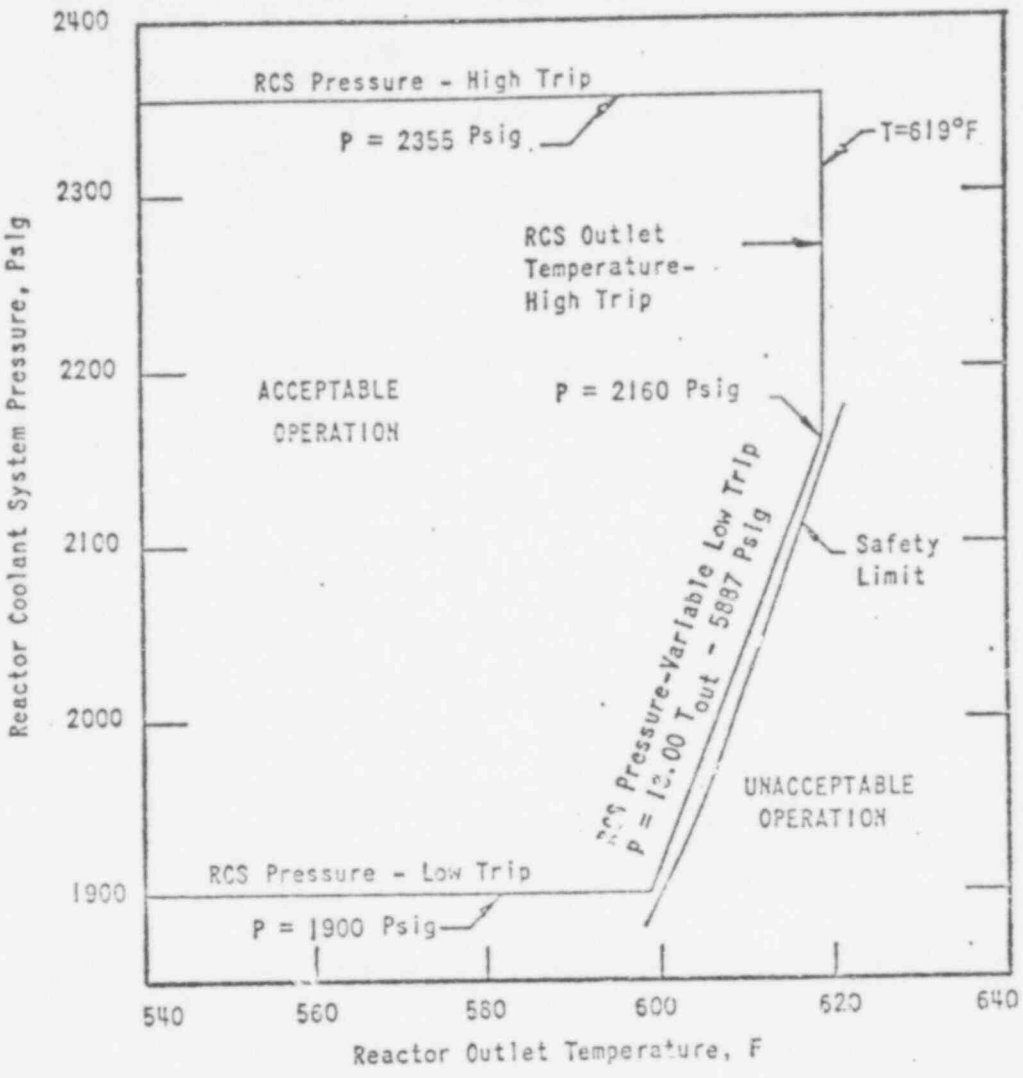
* Trip function may be bypassed in this MODE with RCS pressure below 1920 psig. Bypass shall be automatically removed when RCS pressure exceeds 1950 psig.

** 3 channels per Automatic Actuation Logic, Each R. B. Pressure High Channel trips one Safety Injection Channel and one R. B. Cooling & Isolation Channel.

*** 3 channels per Automatic Actuation Logic, R. B. Spray Valves are actuated by R. B. Cooling and Isolation.

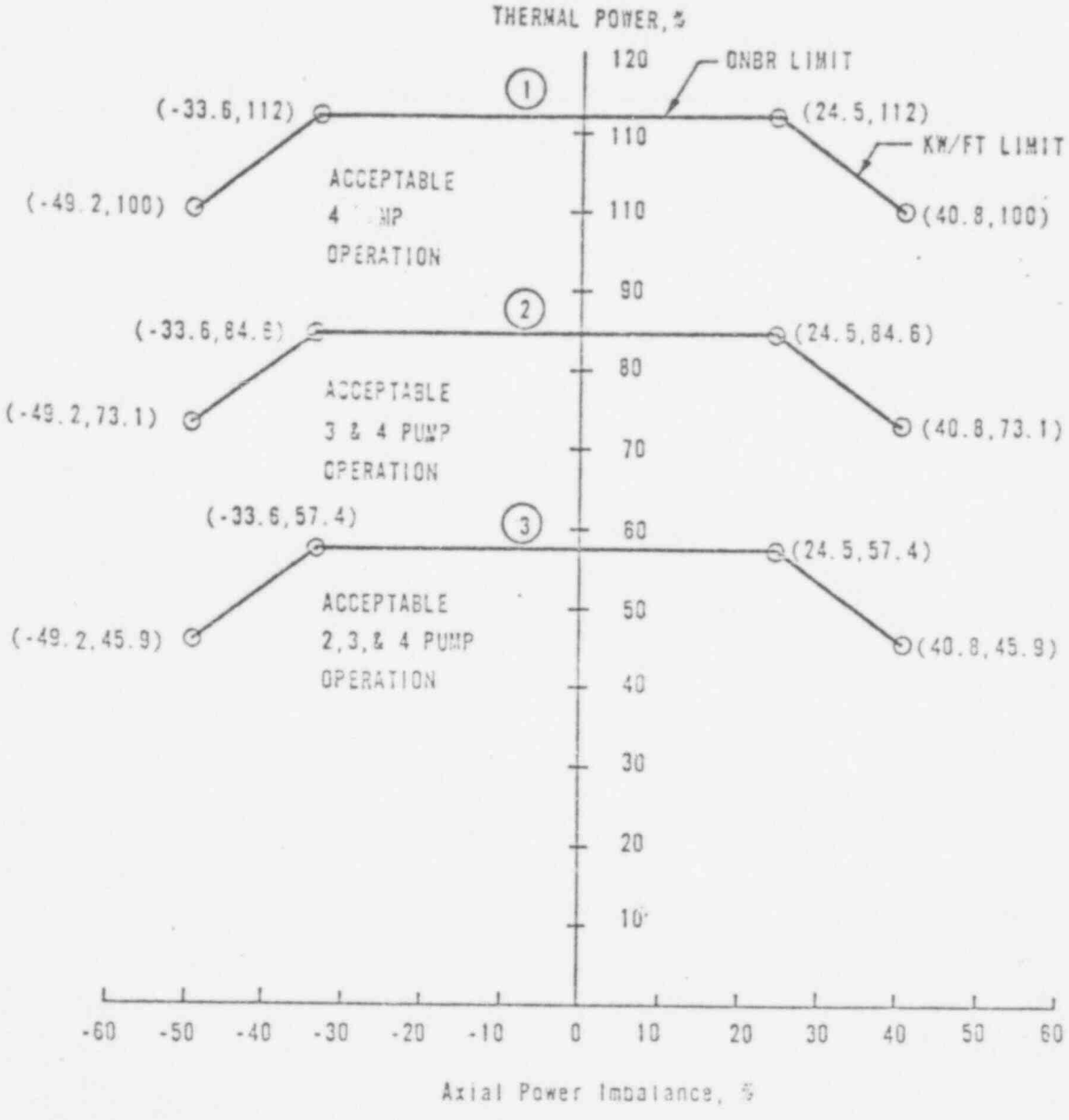
**** Trip function may be bypassed in this mode with steam generator pressure $\leq$ 800 psig. Bypass shall be removed when steam generator pressure $\geq$ 800 psig.

\# The provisions of Specification 3.0.4 are not applicable.



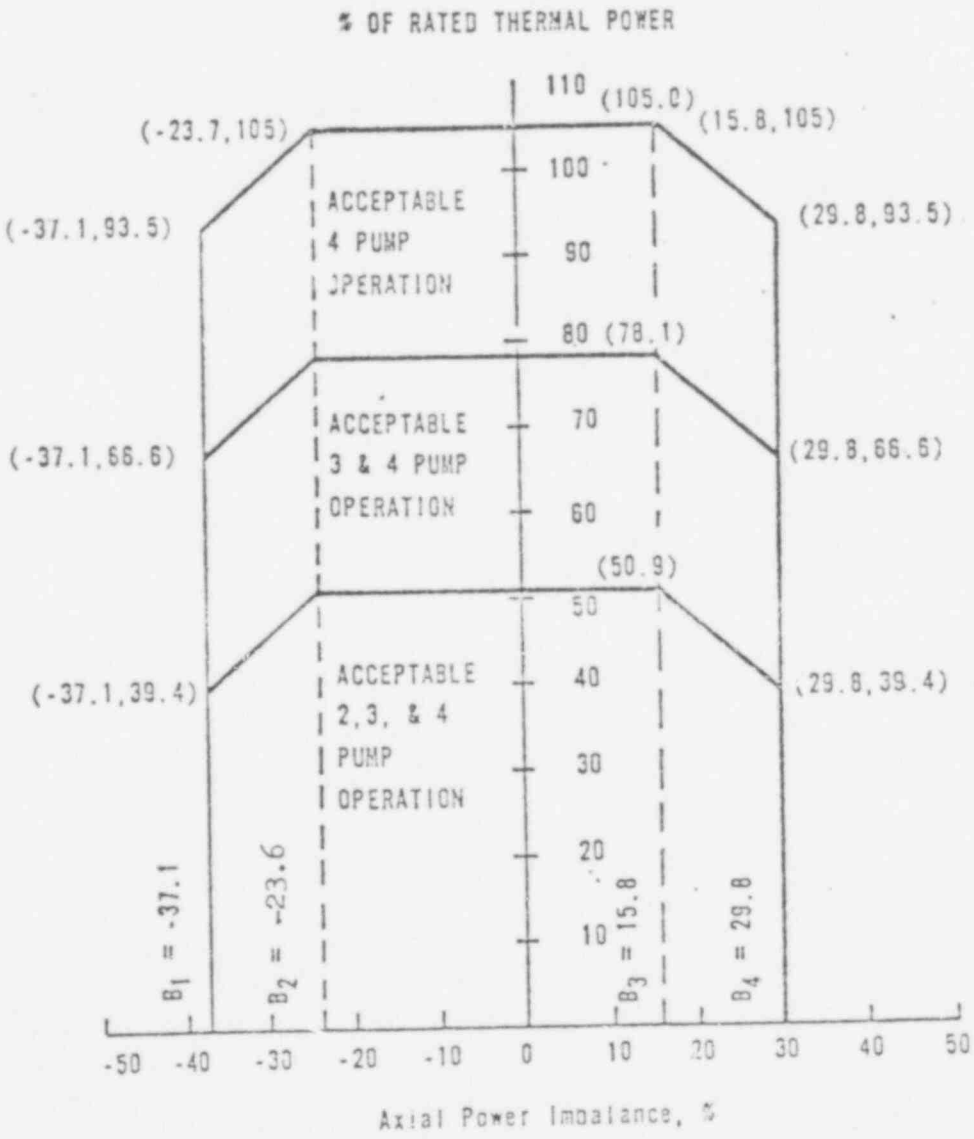
TMI - UNIT 2
REACTOR CORE SAFETY LIMIT

Figure 2.1-1



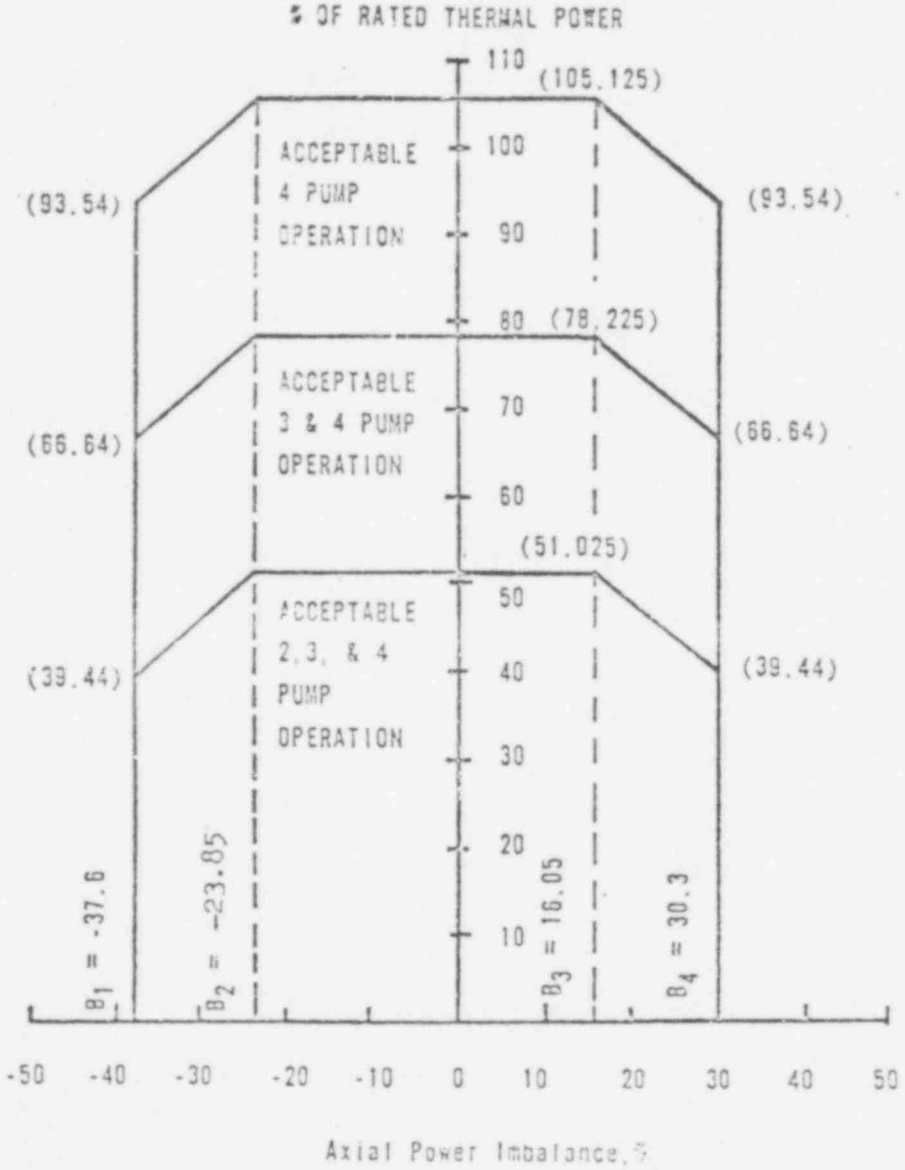
CURVE	REACTOR COOLANT FLOW (GPM)
1	377,000
2	280,400
3	182,800

REACTOR CORE SAFETY LIMITS
Figure 2.1-2



TRIP SETPOINT FOR NUCLEAR OVERPOWER
BASED ON RCS FLOW AND AXIAL POWER
IMBALANCE

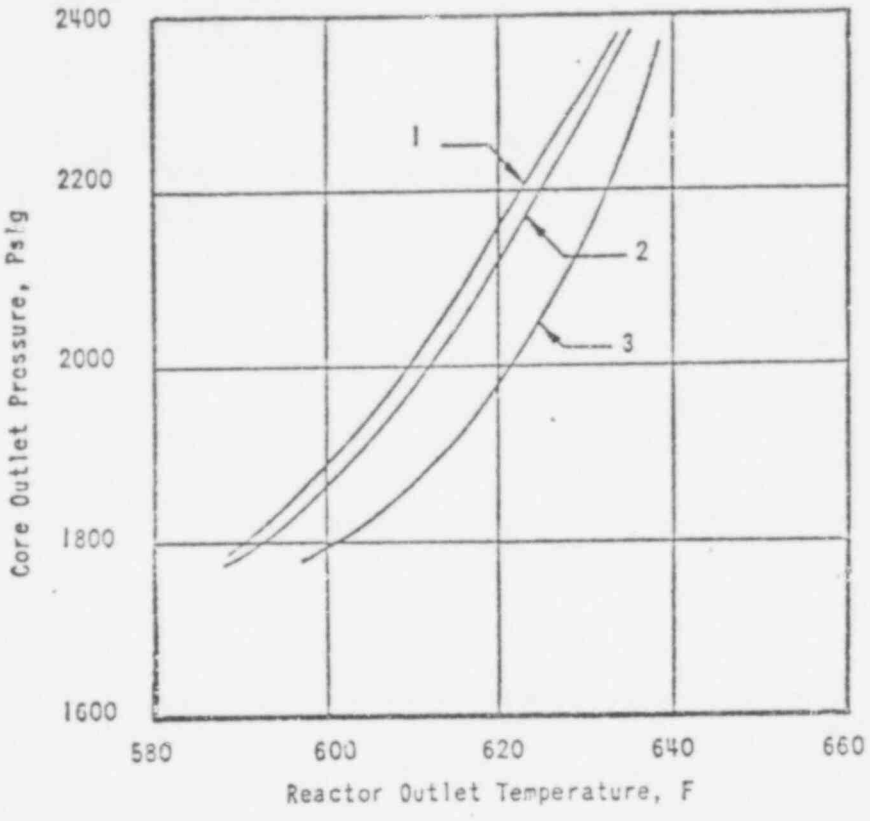
Figure 2.2-1



ALLOWABLE VALUE FOR NUCLEAR OVERPOWER BASED ON RCS FLOW AND AXIAL POWER IMBALANCE

Figure 2.2-2

57-218



CURVE	GPM	POWER	PUMPS OPERATING (TYPE OF LIMIT)
1	377,000 (100%)	112%	FOUR PUMPS (DNBR)
2	280,400 (74.4%)	84.6%	THREE PUMPS (DNBR)
3	182,800 (48.5%)	57.4%	ONE PUMP PER LOOP (QUALITY)

TMI - UNIT 2

PRESSURE/TEMPERATURE LIMITS AT MAXIMUM ALLOWABLE POWER FOR MINIMUM DNBR

Figure 2.1