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

Technical Specification Change Request No. 36

The Licensee requests that the attached pages replace the corresponding existing Technical Specifications pages.

Reasons for Proposed Change

On June 4, 1976, it was discovered that:

a. The fuel densification penalty was not properly incorporated into technical specifications prepared for cycle 2.

b. Proper incorporation of this penalty would affect DNB based pressure-temperature limit curves such that they would be more restrictive.

c. Babcock & Wilcox calculations confirmed that elimination of the internal vent valve bypass flow penalty, as authorized by Nuclear Regulatory Commission letter of March 10, 1976, would more than compensate for this error.

Thus, elimination of the internal vent valve flow penalty will allow continued use of present pressure temperature curves until revised curves included are authorized.

As a prerequisite for eliminating the vent valve flow penalty, the Commission required in its letter of March 10, 1976, "...testing to be conducted each refueling outage to confirm that no vent valve is stuck in an open position and that each valve continues to exhibit complete freedom of movement." This surveillance requirement was performed during the last refueling outage. This proposed change incorporates this surveillance requirement into technical specifications, as well as revised figures 2.1-1, 2.1-3, and 2.3-1 which include credit for elimination of vent valve bypass flow.

Note: The proposed technical specification 4.16 included in Technical Specification Change Request No. 13 (still under review) is no longer needed, due to equipment modification. Therefore, Technical Specification Change Request No. 13 will be retracted.

Safety Analysis Justifying Proposed Change 2380 151

Elimination of the vent valve flow penalty has been authorized by the Commission.

Revised densification analysis indicates that the correct penalties are 5.93% DNBR (versus 1.88% in the Reload Report) and 3.47% power peaking relative to DNBR (versus 1.06% quoted in the Reload Report).

The variable low pressure trip setpoint for cycle 2 operation is based on the four pump open vent valve pressure-temperature limit curve presented in figure 8.3 of the Reload Report. The corresponding limit curve, based

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on closed vent valves and incorporating the revised densification penalty, will be approximately 3F less restrictive (max. allowable T_o will be 3F higher at a given pressure).

The flux/flow trip setpoint for cycle 2 (1.08) is based on the one pump coastdown analysis. When the revised densification penalty is incorporated and the vent valve penalty is eliminated, the thermal-hydraulic limiting flux/flow setpoint is greater than 1.12 (this limit must be at least 1.11 to justify the tech spec setpoint of 1.08). It can also be shown that a thermal-hydraulic limit of 1.11 on the flux/flow setpoint can be justified by taking credit for 1/2 of the vent valve penalty.

Based upon the above, it is determined that this change does not constitute a threat to the health and safety of the public, nor does it involve an unreviewed safety question.

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 49.2 percent or flow rate is 45.4 percent and the power level is 49 percent.

The flux/flow ratios account for the maximum calibration and instrumentation errors and the maximum variation from the average value of the RC flow signal in such a manner that the reactor protective system receives a conservative indication of the RC flow.

No penalty in reactor coolant flow through the core was taken for an open core vent valve because of the core vent valve surveillance program during each refueling outage.

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.

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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.

The low pressure (1800 psig) and variable low pressure (11.379Tout - 4914) trip setpoint shown in Figure 2.3-1 have been established to maintain the DNB ratio greater than or equal to 1.3 for those design accidents that result in a pressure reduction (3, 4).

Due to the calibration and instrumentation errors, the safety analysis used a variable low reactor coolant system pressure trip value of (11.379Tout - 4954).

d. Coolant outlet temperature

The high reactor coolant outlet temperature trip setting limit (619 F) shown in Figure 2.3-1 has been established to prevent excessive core coolant temperatures in the operating range.

The calibrated range of the temperature channels of the RPS is 520 to 620 F. The trip setpoint of the channel is 619 F. Under the worst case environment, power supply perturbations, and drift, the accuracy of the trip string is ±1F. This accuracy was arrived at by summing the worst case accuracies of each module. This is a conservative method of error analysis since the normal procedure is to use the root mean square method.

Therefore, it is assured that a trip will occur at a value no higher than 620F even under worst case conditions. The safety analysis used a high temperature trip set point of 620F.

The calibrated range of the channel is that portion of the span of indication which has been qualified with regard to drift, linearity, repeatability, etc. This does not imply that the equipment is restricted to operation within the calibrated range. Additional testing has demonstrated that in fact, the temperature channel is fully operational approximately 10% above the calibrated range.

Since it has been established that the channel will trip at a value of RC outlet temperature no higher than 620F even in the worst case, and since the channel is fully operational approximately 10% above the calibrated range and exhibits no hysteresis or foldover characteristics, it is concluded that the instrument design is acceptable.

e. Reactor building pressure

The high reactor building pressure trip setting limit (4 psig) p provides positive assurance that a reactor trip will occur in the unlikely event of a steam line failure in the reactor building or a loss-of-coolant accident, even in the absence of a low reactor coolant system pressure trip.

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)	108 times flow minue 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.	(11.379Tout-4914) (1)	(11.379Tout-4914) (1)	11.379Tout-4914) (1)	Bypassed
7. Reactor coolant temp. F., Max.	619	619	619	619
8. High Reactor Building pressure, psig, max.	4	4	4	4

(1) Tout is in degrees Fahrenheit (F)
(2) Reactor coolant system flow, %
(3) Administratively controlled reduction set only 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.

4.16 REACTOR INTERNALS VENT VALVES SURVEILLANCE

Applicability

Applies to Reactor Internals Vent Valves.

Objective

To verify that no reactor internals vent valve is stuck in the open position and that each valve continues to exhibit freedom of movement.

Specification

4.16.1 At intervals not exceeding the refueling interval, each reactor internals vent valve will be tested to verify that no valve is stuck in the open position and that each valve continues to exhibit freedom of movement.

Bases

Verifying vent valve freedom of movement insures that coolant flow does not bypass the core through reactor internals vent valves during operation and therefore insures the conservatism of Core Protection Safety limits as delineated in figures 2.1-1 and 2.1-3, and the flux/flow trip setpoint.

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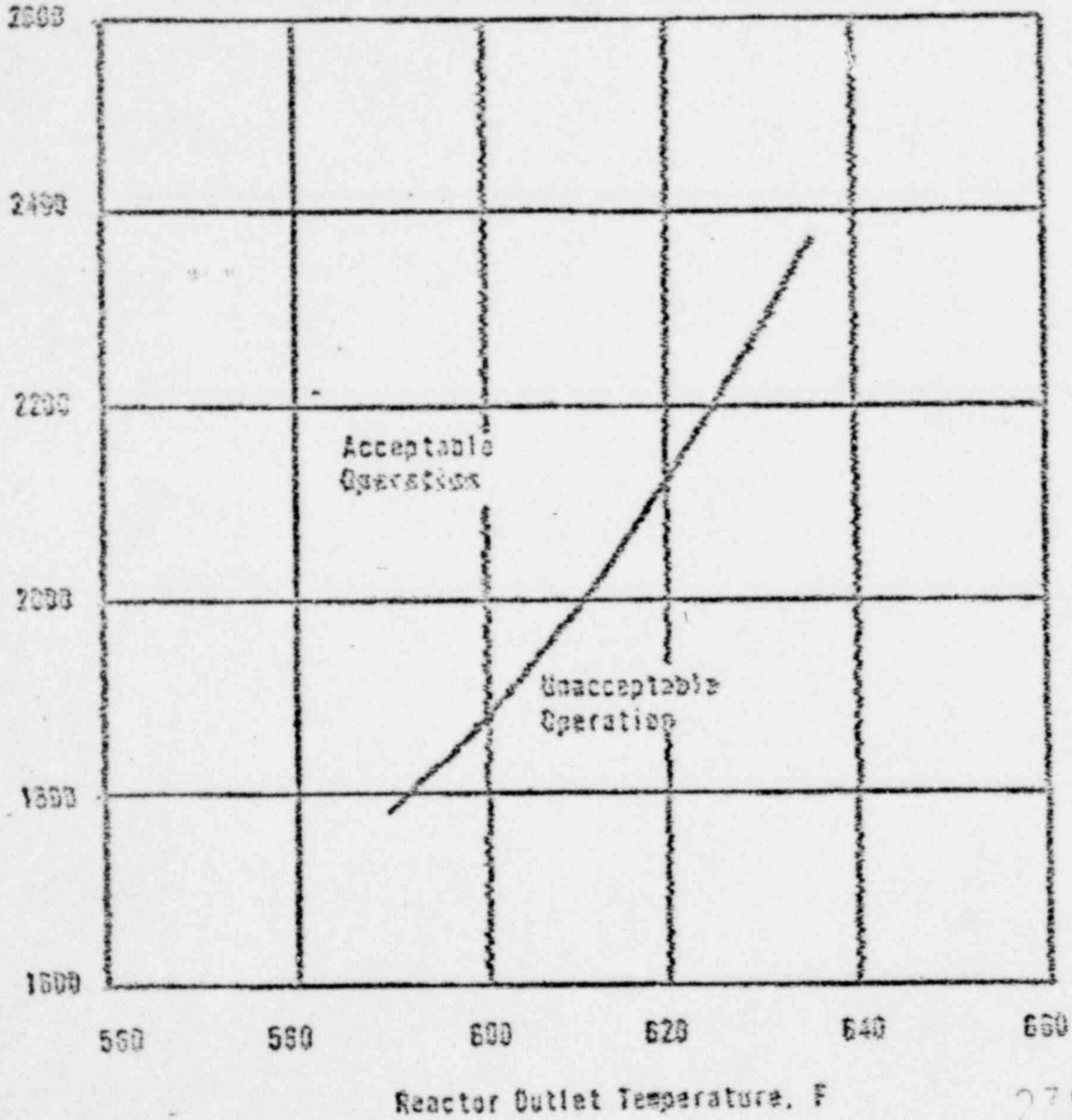

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UNIT 1, CYCLE 2
CORE PROTECTION SAFETY LIMIT
Figure 2.1-1

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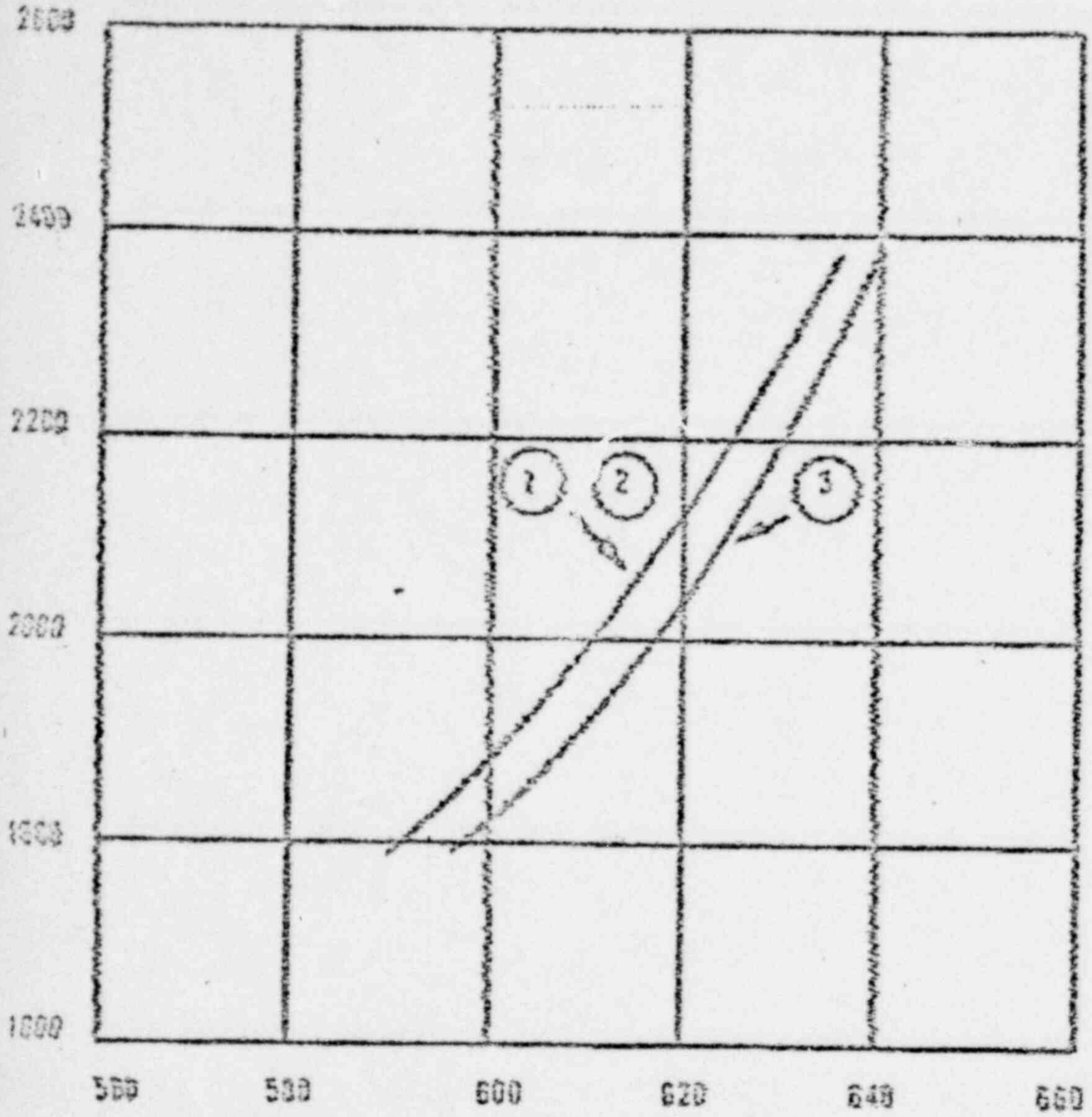

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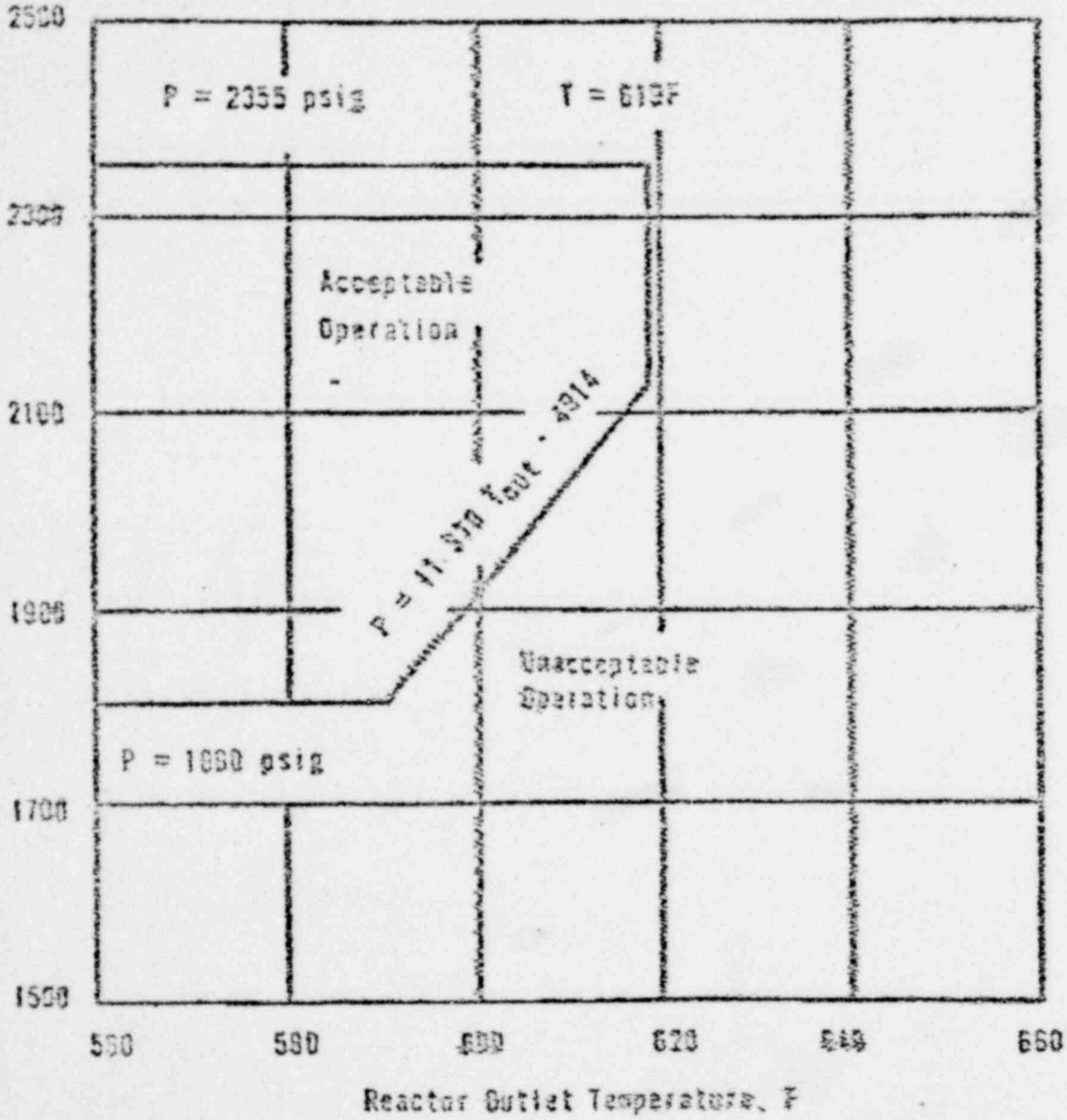
CURVE	REACTOR COOLANT FLOW (LB/HR)	POWER	PUMPS OPERATING (TYPE OF LIMIT)
1	139.8×10^6 (100%)*	112%	Four Pumps (DNBR Limit)
2	104.5×10^6 (74.7%)	86.3%	Three Pumps (DNBR Limit)
3	68.8×10^5 (49.2%)	59.1%	One Pump in Each Loop (Quality Limit)

*105.5% of Cycle 1 Design Flow

UNIT 1, CYCLE 2

CORE PROTECTION SAFETY BASES

Figure 2.1-3



UNIT 3, CYCLE 2
PROTECTION SYSTEM MAXIMUM
ALLOWABLE SET POINTS
Figure 2.3-1

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