D 04/27/78

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)
DISTRIBUTION FOR INCOMING MATERIAL 50-289

REC: REID R W
NRC

ORG: HERBEIN J G
METROPOL EDISON

DOCDATE: 04/20/78
DATE RCVD: 04/26/78

DOCTYPE: LETTER NOTARIZED: NO

COPIES RECEIVED
LTR 1 ENCL 1

SUBJECT:
FURNISHING ADDL INFO TO APPLICANT"S PROPOSED TECH SPEC CHANGE REQUEST NO 70A, CONSISTING OF TECH SPEC PAGES SPECIFIC TO CYCLE 3, THAT WERE INADVERTENTLY OMITTED FROM APPLICANT"S CYCLE 4 SUBMITTAL, AND FURNISHING JUSTIFICATION FOR THESE CHANGES ... W/ATT I

PLANT NAME: THREE MILE ISLAND - UNIT 1

REVIEWER INITIAL: XJM
DISTRIBUTOR INITIAL: DL

***************** DISTRIBUTION OF THIS MATERIAL IS AS FOLLOWS *****************

GENERAL DISTRIBUTION FOR AFTER ISSUANCE OF OPERATING LICENSE.
(DISTRIBUTION CODE A001)

FOR ACTION: BR CHIEF REID**W/7 ENCL

INTERNAL:
REG FILE**W/ENCL
I & E**W/2 ENCL
HANAUER**W/ENCL
EISENHUT**W/ENCL
BAER**W/ENCL
EEB**W/ENCL
J. MCGOUGH**W/ENCL
NRC PDR**W/ENCL
OELD**LTR ONLY
CHECK**W/ENCL
SHAO**W/ENCL
BUTLER**W/ENCL
J COLLINS**W/ENCL

EXTERNAL:
LPDR'S
HARRISBURG, PA**W/ENCL
TIC**W/ENCL
NSIC**W/ENCL
ACRS CAT B**W/16 ENCL

1505 151

DISTRIBUTION: LTR 40 ENCL 39
SIZE: 2P+5P

CONTROL NBR: 781160186

*********************************** THE END ***********************************

7910310589 P

METROPOLITAN EDISON COMPANY SUBSIDIARY OF GENERAL PUBLIC UTILITIES CORPORATION

POST OFFICE BOX 542 READING, PENNSYLVANIA 19603 TELEPHONE 215 – 929-3601

April 20, 1978
GQL 0748

1978 APR 25 AM 10 54

Director of Nuclear Reactor Regulation
Attn: R. W. Reid, Chief
Operating Reactors Branch No. 4
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Sir:

Three Mile Island Nuclear Station, Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289
Technical Specification Change Request No. 70A
(Additional Information)

Your Mr. G. B. Zwetzig identified several TMI-1 Technical Specification pages, specific to Cycle 3, that were inadvertently omitted from our Cycle 4 submittal. These pages and justification for their changes are as follows:

REGULATORY DOCKET FILE COPY

Figure 2.1-3

The power levels for Curves 2 and 3 of Figure 2.1-3 of the TMI-1 Cycle 4 Technical Specifications should be consistent. These power levels differ from those reported for Cycle 3 by the setpoint adjustment error which is a necessary component of the calibration and instrumentation errors. The correct Cycle 4 values for the power levels of the aforementioned Curves 2 and 3 are 87.1% and 59.6% respectively. Figure 2.1-3 has been revised to reflect these valves. There is no reduction in the safety margin as a result of these changes.

Figure 2.1-1

Section 6 of the Cycle 4 Reload Report submitted in support of Tech. Spec. Change Request No. 70, states that the only difference between cycles 3 and 4 is the core configuration. For Cycle 4, the addition of the higher flow resistance Mark B2 and B3 assemblies provides additional DNBR margin over Cycle 3 due to increased flow through the limiting Mark B4 assembly.

Additional conservatism for Cycle 4 operation is provided by the reduced peaking factors. The minimum DNBR calculated from the Cycle 2 analysis (the Cycle 2 DNBR analysis was used for reference Cycle 3) is based

1505 152

A001 S/1

on a radial-local peak of 1.783. The maximum radial-local peak calculated for Cycle 4 operation (287.1 EFPD Cycle 3 and Non-cross core shuffle), including 8% nuclear uncertainty, is 1.547 at BOC. This decreases to 1.403 at the end of Cycle 4. This provides a 13.2% margin to the design peak at BOC and a 21.3% margin at EOC.

Figure 2.1-1, unchanged for Cycle 4, represents a more conservative limit and does not result in a reduced safety margin.

Page 2-4, Page 2-9 (Table 2.3-1), and Figure 2.3-1

A bounding analysis, showing peak RC system pressure vs. moderator coefficients and a conservative doppler coefficient for the feedwater line break accident, has been performed to justify the high pressure trip setpoint of 2405 psig and the corresponding 2500 psig pressurizer code safety valve settings. Results of this analysis were submitted to NRC on April 17, 1978, (GQL 0669), and supplemented on April 20, 1978 (GQL 0743). Since the Cycle 4 parameters are bounded by the analysis, and the 2750 psig peak pressure limit is not exceeded, there is no reduction in safety margin and no resulting threat to the health and safety of the public.

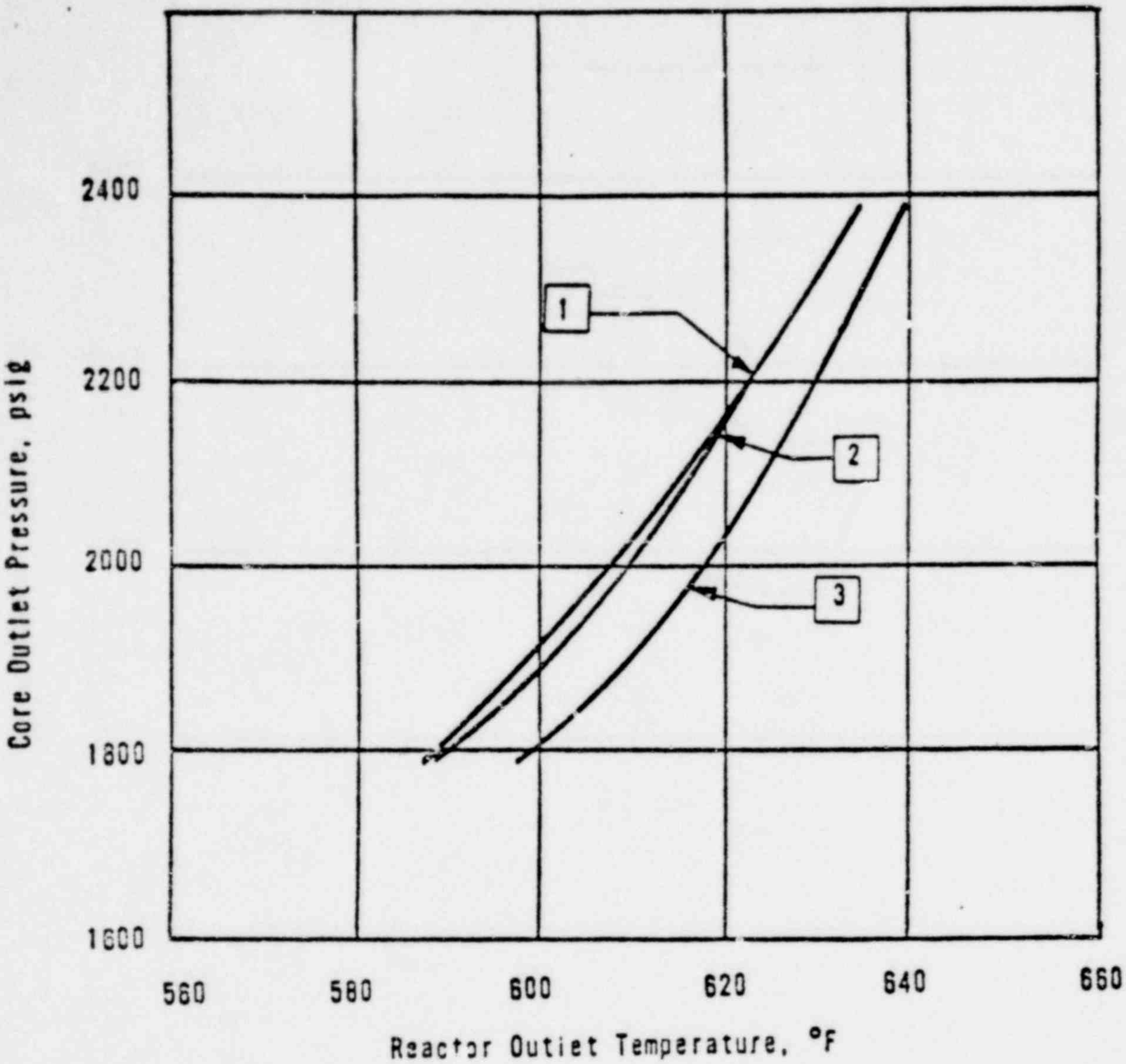
Sincerely,

JG Herbein

J. G. Herbein
Vice President - Generation

JGH:RJS:jmr
Attachments

1505 153



REACTOR COOLANT FLOW

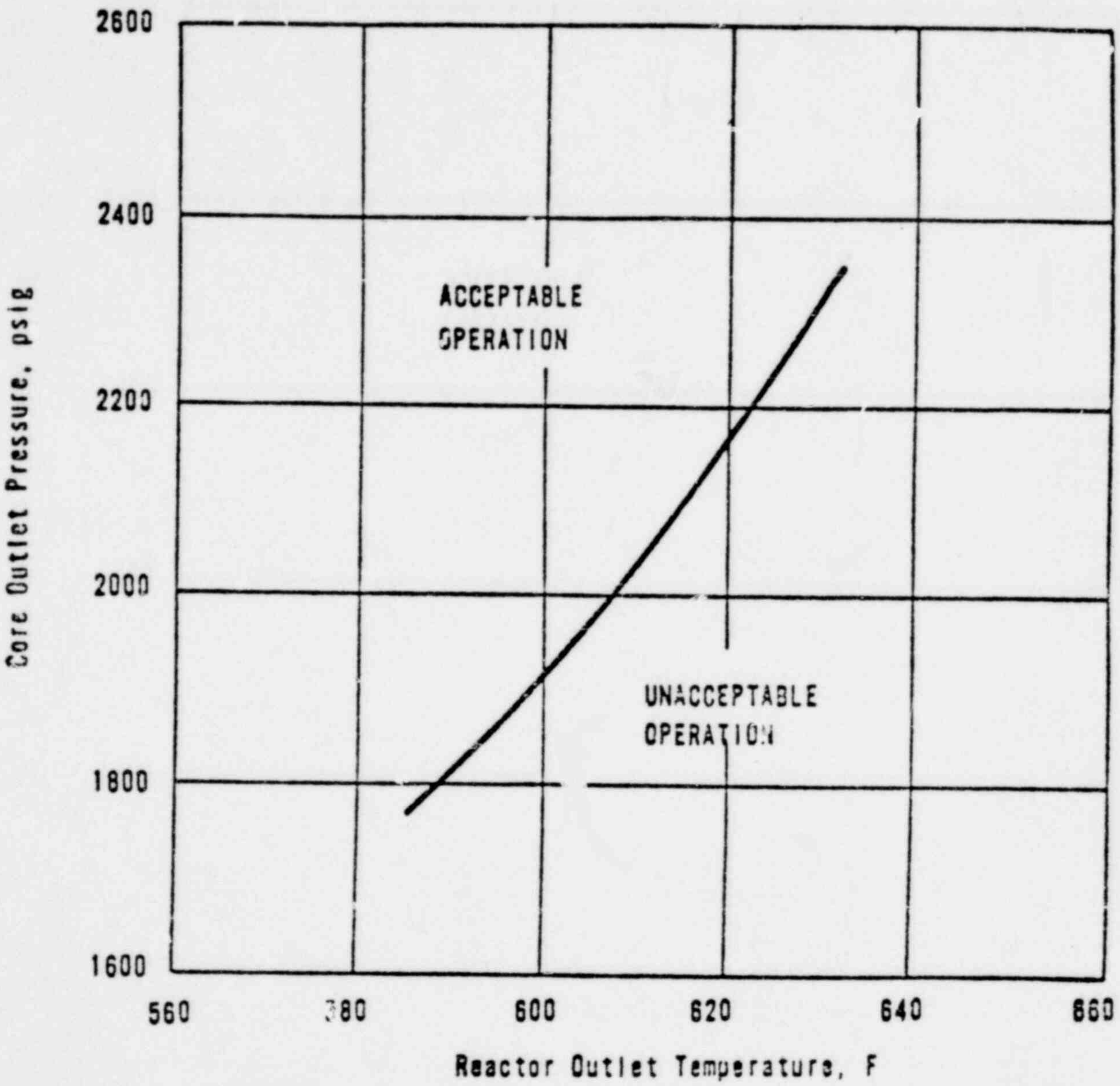
CURVE	(LBS/HR)	POWER	PUMPS OPERATING (TYPE OF LIMIT)
1	139.8×10^6 (100%)*	112%	Four Pumps (DNBR Limit)
2	104.5×10^6 (74.7%)	87.1%	Three Pumps (DNBR Limit)
3	68.8×10^6 (49.2%)	59.6%	One Pump in Each Loop (Quality Limit)

*106.5% of Cycle 1 Design Flow

THREE MILE ISLAND, UNIT I
CORE PROTECTION SAFETY BASES

Figure 2.1-3

1505 154



THREE MILE ISLAND, UNIT I
CORE PROTECTION SAFETY LIMIT
Figure 2.1-1

1505 155

2.2 SAFETY LIMITS - REACTOR SYSTEM PRESSURE

Applicability

Applies to the limit on reactor coolant system pressure.

Objective

To maintain the integrity of the reactor coolant system and to prevent the release of significant amounts of fission product activity.

Specification

2.2.1 The reactor coolant system pressure shall not exceed 2750 psig when there are fuel assemblies in the reactor vessel.

Bases

The reactor coolant system (1) serves as a barrier to prevent radionuclides in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure, the reactor coolant system is a barrier against the release of fission products. Establishing a system pressure limit helps to assure the integrity of the reactor coolant system. The maximum transient pressure allowable in the reactor coolant system pressure vessel under the ASME Code, Section III, is 110% of design pressure. Thus, the safety limit of 2750 psig (110% of the 2500 psig design pressure) has been established. (2) The maximum settings for the reactor high pressure trip (2405 psig) and the pressurizer code safety valves (2500 psig) (3) have been established in accordance with ASME Boiler and Pressure Vessel Code, Section III, Article 9, Winter, 1968 to assure that the reactor coolant system pressure safety limit is not exceeded. The initial hydrostatic test was conducted at 3125 psig (125% of design pressure) to verify the integrity of the reactor coolant system. Additional assurance that the reactor coolant system pressure does not exceed the safety limit is provided by setting the pressurizer electromatic relief valve at 2255 psig. (4)

References

(1) FSAR, Section 4

(2) FSAR, Section 4.3.10.1

(3) FSAR, Section 4.2.4

(4) FSAR, Table 41

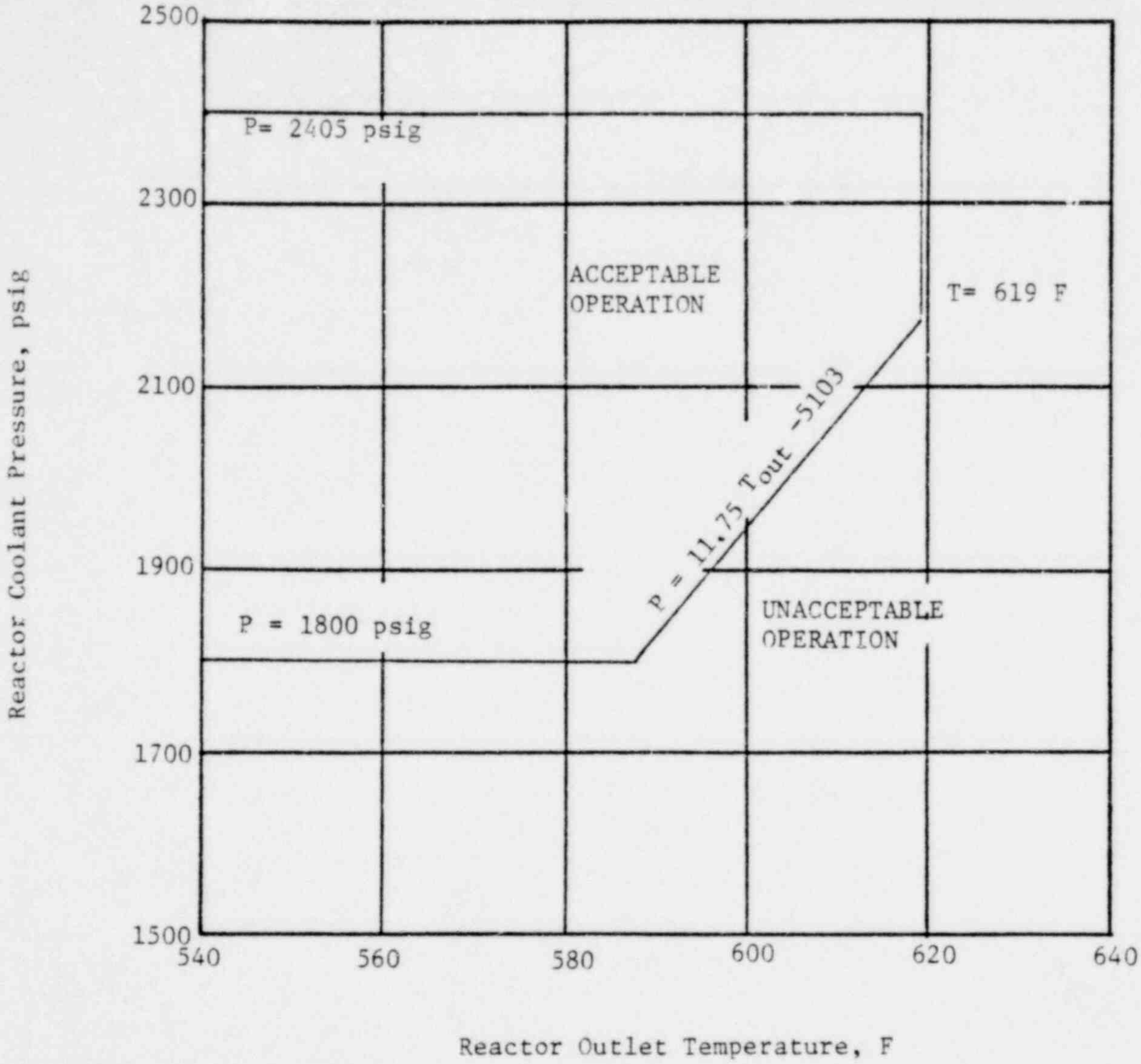
TABLE 2.3-1

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 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.	2405	2405	2405	1720(4)
5.	Low reactor coolant system pressure, psig min.	1800	1800	1800	Bypassed
6.	Variable low reactor coolant system pressure psig, min.	(11.75 Tout-5103) (1)	(11.75 Tout-5103)(1)	(11.75 Tout-5103)(1)	Bypassed
7.	Reactor coolant temp. F., max.	619	619	619	619
8.	High Reactor Building pressure, psig, max.	4	4	4	4

1505 157

(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
(6) Trip settings limits are setting limits on the setpoint side of the protection system bistable comparators.



Three Mile Island, Unit 1
Protection System Maximum
Allowable Set Points

Figure 2.3-1

1505 158