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JUSTIFICATION FOR REMOVAL OF ORIFICE ROD ASSEMBLIES IN THREE MILE ISLAND UNIT 2, CYCLE 1

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1. INTRODUCTION

This report provides justification for continued operation of the first cycle of Three Mile Island Unit 2 (TMI-2) at the rated core power of 2772 MWt following the removal of orifice rod assemblies (ORAs) from the core. The ORAs are used to limit bypass flow through fuel assemblies with empty guide tubes. A system flow of 102% of design flow has been used in these analyses which offsets the increased core bypass flow due to removal of ORAs.

An evaluation of thermal-hydraulic performance has been made based on the increase in system flow and removal of ORAs and has been compared to the analyses presented in the TMI-2 FSAR[1] and Fuel Densification Report.[2] This evaluation shows that the effects of the removal of forty ORAs and the increase in reactor coolant flow rate provide improved safety margins relative to those reported in the TMI-2 FSAR[1] and Fuel Densification Report.[2]

The use of retainers[3] to provide positive holddown of burnable poison rod assemblies (BPRAs) in the remainder of cycle 1 has also been considered.

2. THERMAL-HYDRAULIC DESIGN

The thermal-hydraulic design evaluation supporting continued cycle 1 operation used the methods and models described in reference 2 with the following exceptions:

1. An increase in core bypass flow due to ORA removal.
2. An increase in system flow.
3. The inclusion of retainers to provide positive holddown of BPRAs.

During the initial portion of cycle 1 operation, fuel assemblies which did not contain control rods, BPRAs, or neutron sources had ORAs installed in the guide tubes to minimize core bypass flow. The maximum core bypass flow, with ORAs installed in forty fuel assembly locations, was 6.04% of system flow. Thirty-eight ORAs will be removed for the remainder of cycle 1. Two fuel assemblies will contain primary neutron sources and modified ORAs. The thermal-hydraulic analysis assumed a total of forty vacant fuel assemblies and resulted in a maximum core bypass flow of 7.6%.

As previously noted, a system flow of 102% of design flow was used in the analysis (see Table 2-1) which offsets the affect of the increased bypass flow. This system flow rate is conservatively based on a predicted four-pump flow rate of 105% of design flow as verified during startup testing.

Retainers will be installed on all fuel assemblies containing BPRAs and primary neutron sources with modified ORAs. This retainer design is described in reference 3. The additional form loss due to retainer installation has been included in the calculation of core flow distribution. The limiting fuel assembly does not contain a BPRA during cycle 1 operation.

Maximum design conditions and significant parameters are shown in Table 2-1 for cycle 1 operation with and without the ORAs.

The potential affect of fuel rod bow on DNBR was considered by incorporating suitable margins into DNB limited core safety limits and RPS setpoints (pressure temperature limits and flux/flow setpoint). The maximum rod bow penalty was calculated from the equation:

$$\frac{\Delta C}{C_o} = 0.065 + 0.001449 \sqrt{BU}$$

where

ΔC = rod bow magnitude, mils,

C_o = initial gap (138 mils),

BU = maximum assembly burnup, MWd/mtU.

The pressure-temperature limit curves shown in Figure 2-1 (section 5 of this report) provide the basis for the variable low-pressure trip setpoint. These curves have been changed from those of bases Figure 2-1 of the TMI-2 Technical Specifications.[4] The revised pressure-temperature limits cover an 11.2% fuel rod bow penalty, based on an assumed maximum assembly burnup of 33,000 MWd/mtU, while incorporating the core flow changes discussed above.

The flux/flow trip setpoint was determined by analyzing an assumed one-pump coastdown starting from an initial indicated power level of 102%. The Technical Specification flux/flow setpoint of 1.05 was re-evaluated based on the initial conditions determined with the ORAs out. The 1.05 setpoint provides coverage of a 9.1% rod bow penalty in the analysis. The maximum cycle 1 burnup is 19,422 MWd/mtU. Using this burnup, a rod bow penalty of 9.1% is calculated. A thermal margin credit equivalent to 1% DNBR to offset the rod bow penalty has been used as a result of the flow area (pitch) reduction factor included in all thermal hydraulic analysis. Applying the 1% credit against the 9.1% calculated penalty results in an 8.1% penalty to be applied to the analysis. Therefore, the present flux/flow setpoint provides more than adequate rod bow penalty coverage for cycle 1 operation.

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Table 2-1. Thermal-Hydraulic Design Conditions

	TMI-2 FSAR	Densif'n Report	Revised Cycle 1
Design power level, MWt	2772	2772	2772
System pressure, psia	2200	2200	2200
RC flow, gpm	369,600	369,600	377,000
Vessel inlet coolant temperature, 100% power, F	557	557	557.2
Reference design radial-local power peaking factor	1.783	1.783	1.783
Reference design axial flux shape	1.5 cos	1.5 cos	1.5 cos
Hot channel factors			
Enthalpy rise	1.011	1.011	1.011
Heat flux	1.014	1.014	1.014
Flow area	0.98	0.98	0.98
Active fuel length, in.	144.0	141.7	141.7
Average heat flux, 100% power, Btu/h-ft^2	185,000[a]	188,000[b]	188,000[b]
CHF correlation	W-3	BAW-2	BAW-2
Minimum DNBR, 112% power	1.39	1.62	1.65

(a)Based on the active fuel length and cold fuel pin diameter.

(b)Based on the densified active fuel length and hot fuel pin diameter.

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These analysis results demonstrate that the removal of forty ORAs from TMI-2, when combined with the increased reactor coolant system flow rate, result in improved core safety margins relative to those defined in references 1 and 2.

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3. TRANSIENT ANALYSIS

The DNBR related transients presented in reference 2 have been reviewed for applicability to operation with the ORAs removed. The four pump coastdown is the loss-of-coolant-flow (LOCF) transient analyzed in the Densification Report. The minimum DNBR during this transient was 1.65 (BAW-2). The initial conditions for these transients are at 102% power. Re-analysis at 102% power with ORAs removed shows an increase of 1% in the initial DNBR. The higher initial minimum DNBR makes the results of the transients analyzed for the Densification Report applicable and conservative for the revised cycle 1.

All loss-of-coolant flow transients, with the exception of the loss of one pump from four pump operation, will result in a reactor trip initiated by the pump monitors. The most limiting LOCF transient for which the pump monitors provide DNBR protection is the four pump coastdown which has been shown to be acceptable.

The one pump coastdown from four pump operation is the most limiting flow transient by virtue of its use in determining the flux/flow trip setpoint. The flux/flow trip is based on preventing the minimum DNBR from going below the design value plus the rod bow penalty. Therefore, a one pump coastdown with the resulting flux/flow reactor trip will result in the most limiting DNBR during normal operation.

The TMI-2 FSAR[1] has been reviewed for the most limiting DNBR transients of moderate frequency since the one pump coastdown does not appear directly as an accident. The most limiting FSAR transient is the excessive heat removal accident (feedwater temperature decrease). This transient has been re-analyzed for revised cycle 1 operation with the same input as used in the FSAR. The results of the re-analysis are shown on Figure 3-1. The minimum DNBR is 1.58 (BAW-2) versus a 1.43 (W-3) reported in the FSAR.

Figure 3-1. Feedwater Temperature Decrease

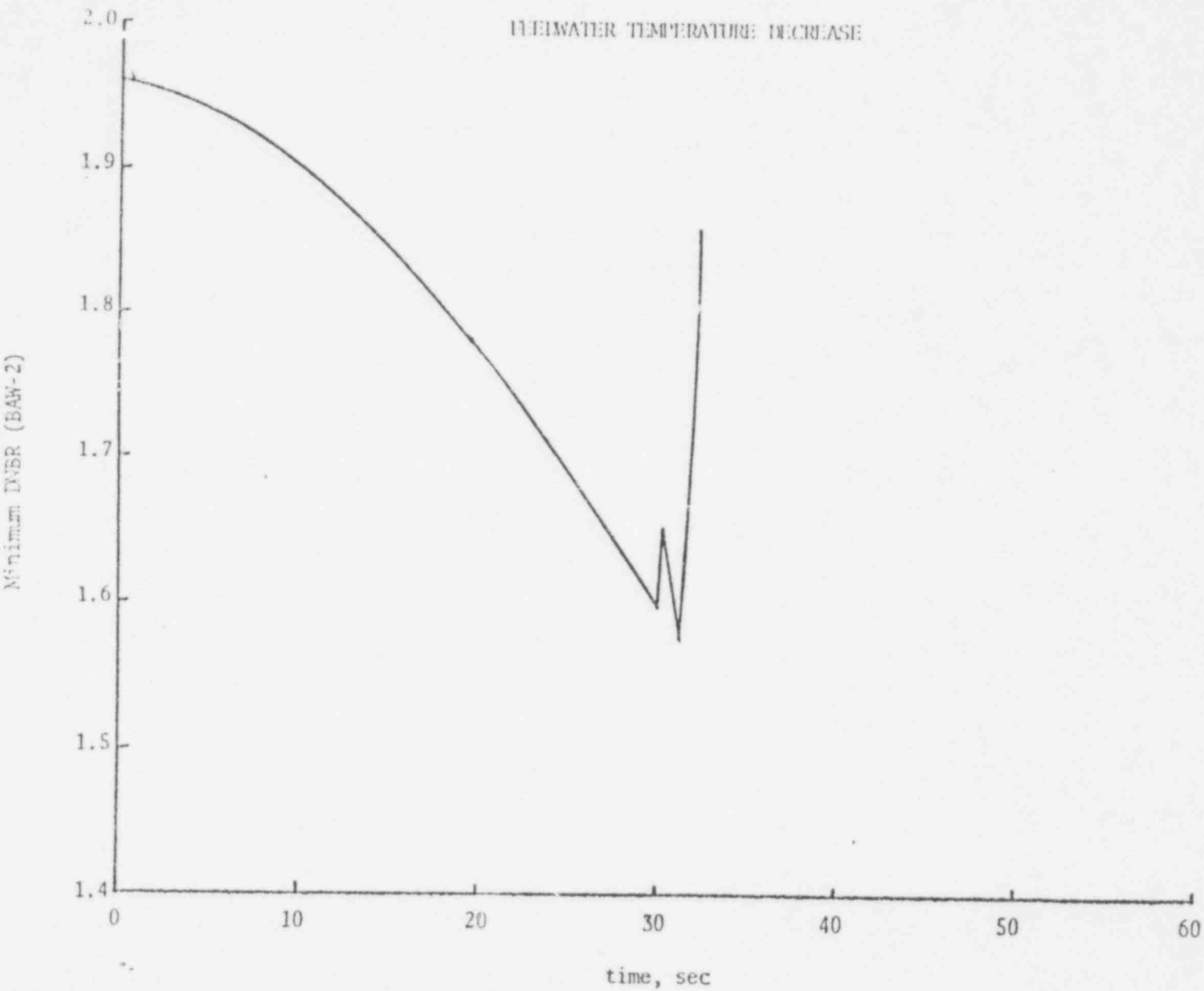

Figure 4-1. TMI-2 Initial Core Loading Plan

FUEL TRANSFER CANAL →

X

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						TP	TY	UC	TC	TW					
B				UT	UX	UG C139	RL B153 L007	UB C124	RF B158 L008	UA C132	U1	U4 03AT & L004 P			
C			TS	T7 B164 L009	PW C163	SN B174 L010	Q8 C179	SY B177 L011	QB C172	SC B173 L012	PY C156	TB B163 L013	TF		
D		TR	TA B168 L014	PV C155	T5 B197 L015	Q4 A024	SW B160 L016	RC C144	S1 B183 L017	Q2 A017	T4 B200 L018	PT C152	T8 B169 L019	TD	
E		U2	QS C162	PN B198 L020	QN C131	RX B195 L021	QG C171	SZ B187 L022	QJ C164	TO B204 L023	R8 C128	PF B206 L024	PZ C157	U5	
F	TN	UR C138	RW B171 L025	Q5 A023	[illegible] B202 L026	R6 C151	RS B146 L027	QZ C140	S4 B155 L028	QK C148	S7 B201 L029	QQ A018	RH B182 L030	UQ C133	TG
G	TM	SQ B147 L031	QV C178	SO B185 L032	R2 C170	RQ B145 L033	QX C183	RP B149 L034	QE C180	SM B139 L035	QH C165	SS B176 L036	QA C173	RK B157 L037	US
H	UU	UF C127	S2 B186 L038	RD C147	T1 B189 L039	QY C143	SH B143 L040	RR C123	SJ B142 L041	QF C141	T2 B188 L042	RA C145	SX B184 L043	UE C125	UY
K	UW	RM B152 L044	Q7 C177	RU B161 L045	R4 C169	T3 B141 L046	QD C182	RT B144 L047	QM C181	SL B154 L048	QC C166	SB B179 L049	Q9 C174	RJ B151 L050	UV
L	TQ	UK C137	RG B162 L051	QP A022	SR B192 L052	QL C150	SK B156 L053	RO C142	S3 B150 L054	QW C149	RV B199 L055	Q3 A019	SA B165 L056	UL C134	TJ
M		UO	QR C161	PM B193 L057	Q6 C130	S8 B194 L058	R3 C168	S6 B159 L059	R1 C167	RY B191 L060	R9 C129	T6 B205 L061	QT C158	TU	
N		TZ	U9 B175 L062	PU C154	SG B203 L063	Q1 A021	SV B180 L064	RB C146	RZ B181 L065	QU A020	PQ B196 L066	PS C153	UH B190 L067	TE	
O			U8	UJ B166 L068	QO C160	S9 B167 L069	R7 C176	ST B172 L070	R5 C175	SU B178 L071	PX C159	T9 B170 L072	TT		
P				TV L005 & 03AU P	U6	UP C136	RE B148 L073	UM C126	SP B140 L074	UN C135	U7	TX			
R						TH	TL	UD	U3	TK					

Z

Fuel Assembly ID (PS-RD:Batch 1; RE-T6 & PM-PQ:Batch 2; T7-UY:Batch 3)
Control Component ID (CXXX - CRA; AXXX - APSRA; BXXX - BPRA; OXXX-MOD.ORA; LOXX -RETAINER)
Sources (P - Primary)

NOTE: NJ00 precedes all fuel assembly ID's.

4. CORE LOADING PLAN

Figure 4-1 shows the revised core loading plan for the remainder of cycle 1. All fuel assemblies are remaining in their original core locations, i.e., no fuel shuffle will take place. The changes occurring are:

1. Retainers will be installed on all BPRAs.
2. Thirty-eight ORAs will be removed.
3. Two ORAs will be modified and installed in the primary neutron source locations (B-12 and P-4).

5. PROPOSED MODIFICATIONS TO TECHNICAL SPECIFICATIONS

The Technical Specifications have been revised for the remainder of cycle 1 operation. Changes were the result of the following:

1. The pressure-temperature limits have been revised to incorporate the affects of ORA removal, retainer installation, and rod bow penalty.
2. System flow of 102% of design flow was used.
3. The low pressure setpoint has been raised to account for the LOCA small break analysis (backup function only).
4. Instrument drift numbers have been included for calibration drift in accordance with item 2.C.(3)f. of the operating license.

Figures 2.1-1, 2.1-2, 2.2-1, 2.2-2, and 2.1 (Tech Spec numbering) illustrate the revisions to previous Technical Specification limits.

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Figure 2.1-1. TMI Unit 2 Reactor Core Safety Limit

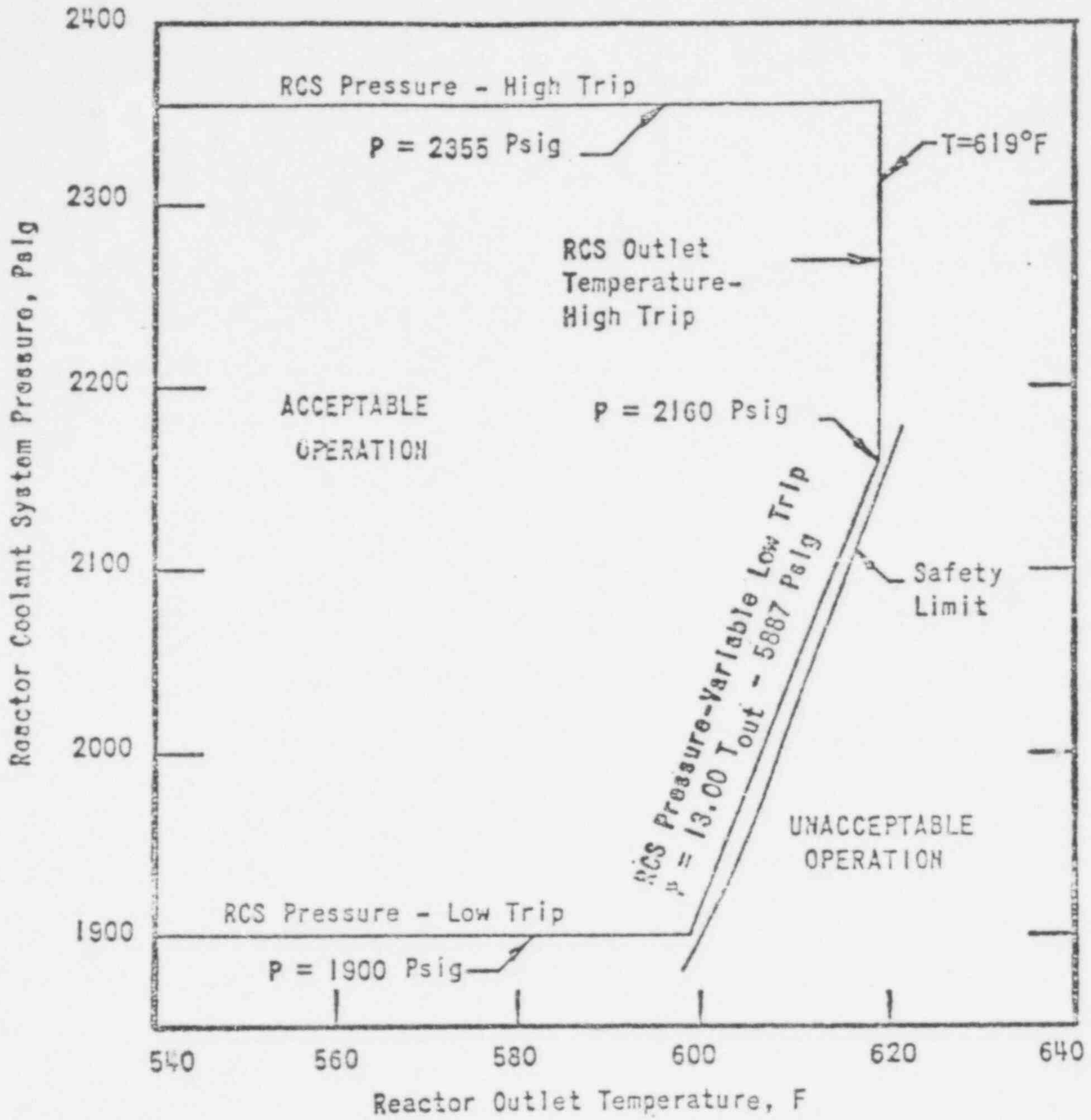
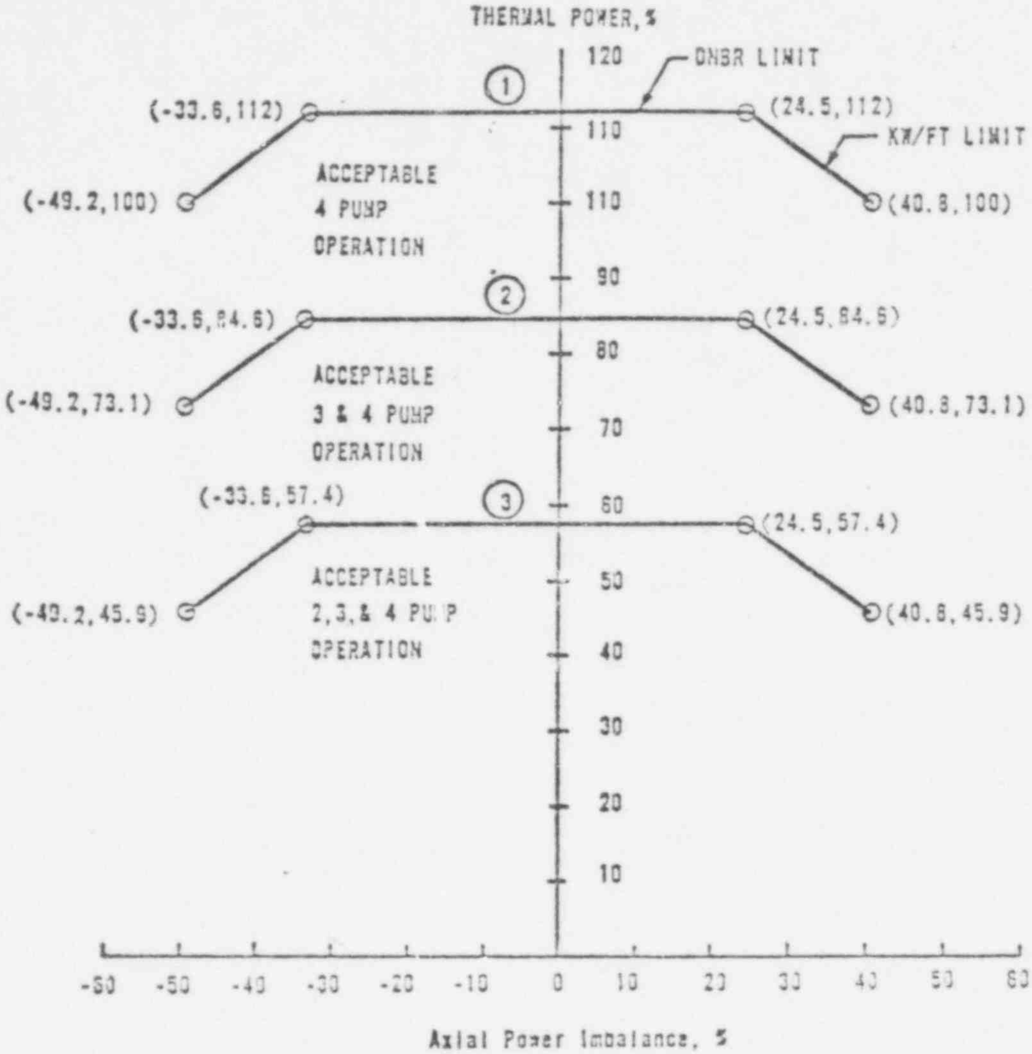


Figure 2.1-2. Reactor Core Safety Limits



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

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Figure 2.2-1. Trip Setpoint for Nuclear Overpower Based on RCS Flow and Axial Power Imbalance

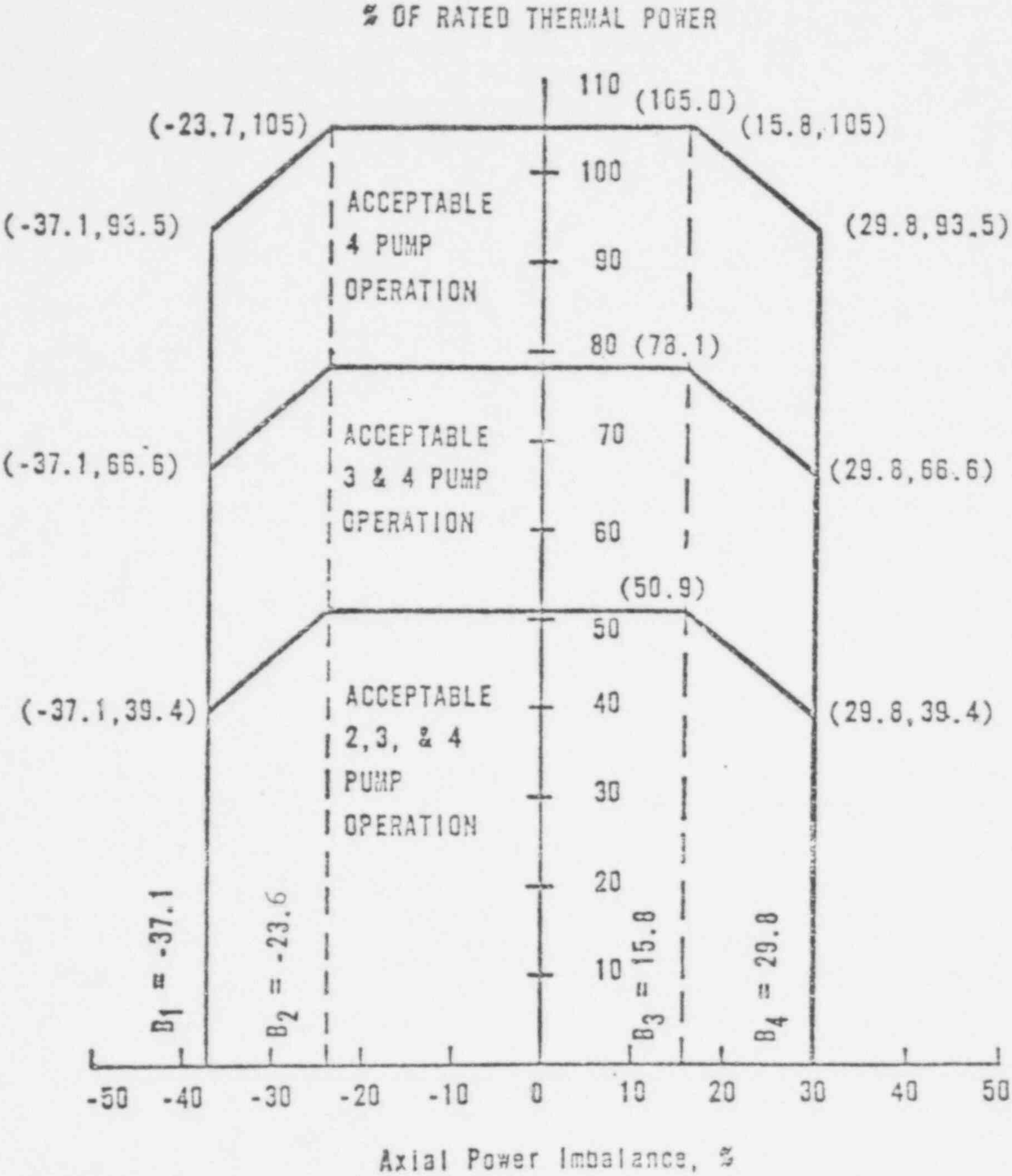

Figure 2.2-2. Allowable Value for Nuclear Overpower Based on RCS Flow and Axial Power Imbalance

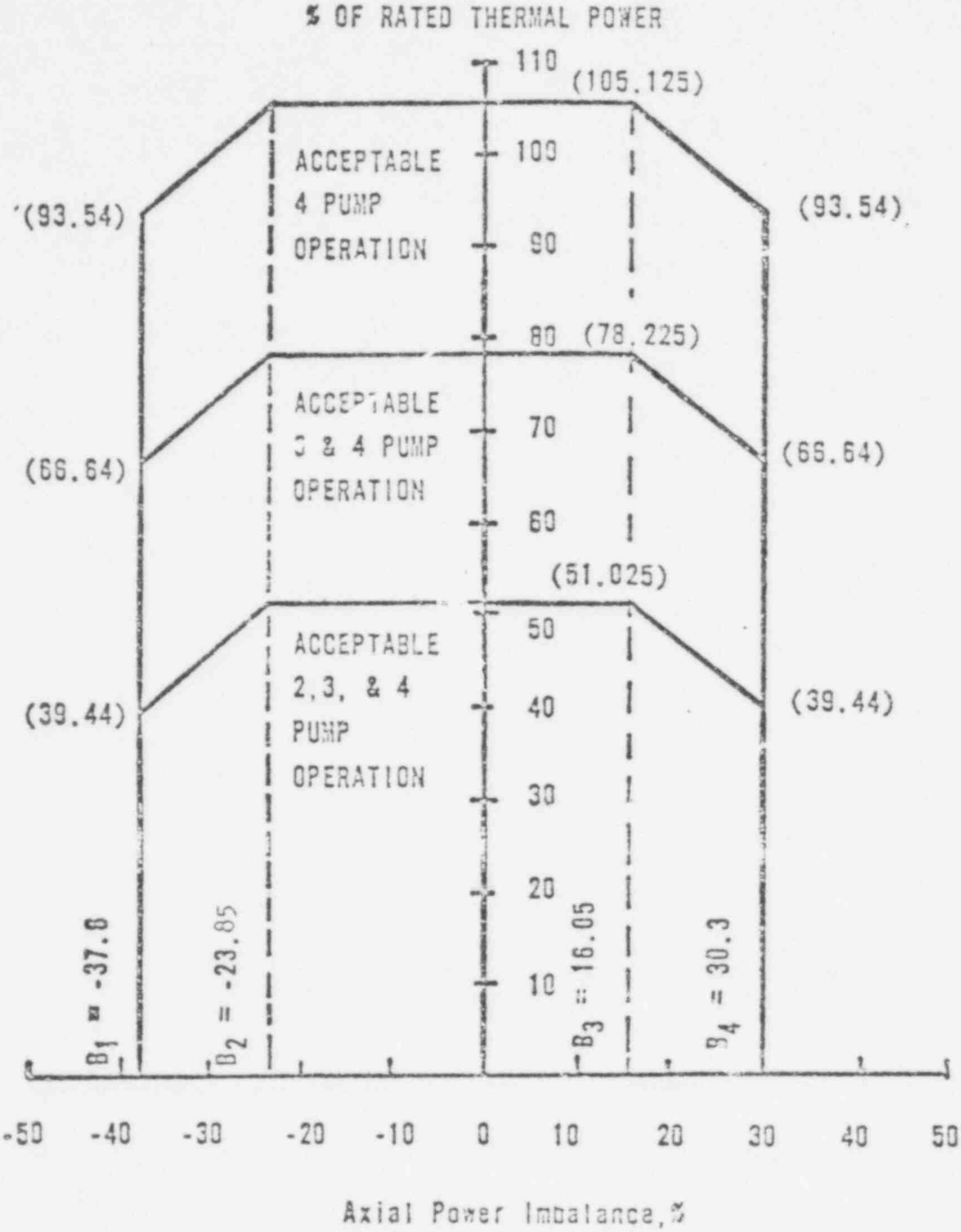

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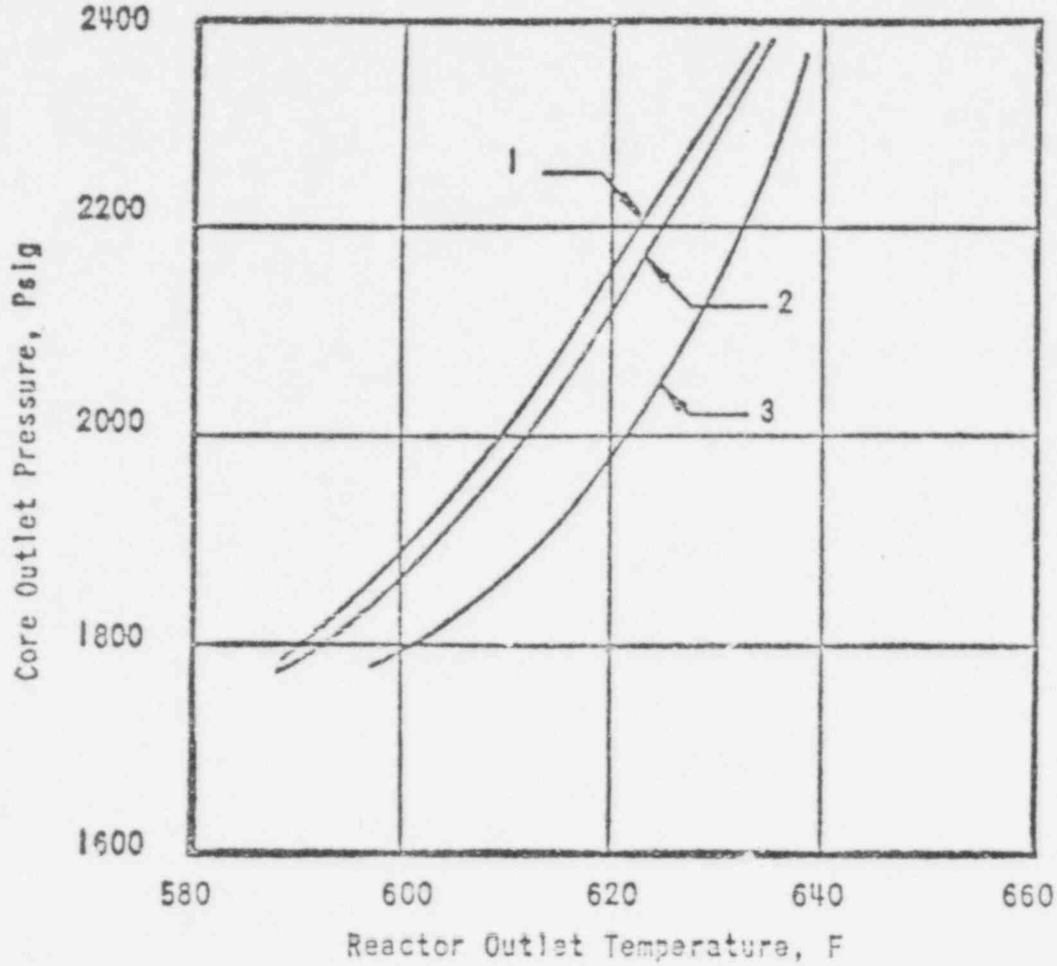
Figure 2.1. TMI Unit 2 Pressure/Temperature Limits at Maximum Allowable Power for Minimum DNBR



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)

REFERENCES

1 Three Mile Island Nuclear Station, Unit 2 — Final Safety Analysis Report, Docket No. 50-320.

2 Three Mile Island, Unit 2 Fuel Densification Report, BAW-1455, Babcock & Wilcox, Lynchburg, Virginia, July 1977.

3 BPRA Retainer Design Report, BAW-1496, Babcock & Wilcox, Lynchburg, Virginia, May, 1978.

4 NUREG-0432, Three Mile Island Nuclear Station Unit 2 Technical Specification, Appendix A to License No. DPR-73, February 8, 1978.