8-8-75

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. 17

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 8th day of August, 1975

Richard I Ruth
Notary Public
RICHARD I. RUTH
Notary Public, Muhlenberg Twp., Berks Co.
My Commission Expires September 23, 1978

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Regulatory Docket File

8-8-75

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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. 17 to Appendix A of the Operating License for Three Mile Island Nuclear Station, Unit 1, dated August 8, 1975, and filed with the U.S. Nuclear Regulatory Commission August 8, 1975, has this 8th day August, 1975, 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 ____________________
Vice President-Generation

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Containment Pressure has been calculated using TMI-1 Specific Parameters. As built containment passive heat sink data has been compiled and is shown in Attachment 2.

B&W has completed a computer calculation utilizing the TMI-1 Specific Parameters with the generic method to determine containment back pressure, in the event of a LOCA. Attachment 3 gives the results of the analysis and shows that the TMI-1 building yields a higher containment pressure than that calculated using the generic containment model described in Section 4.4 of BAW-10103. Therefore the minimum containment back pressure transient used in the ECCS analysis (BAW-10103) is conservative relative to the actual results for the TMI-1 building. The generic back pressure was also used in the development of the proposed Technical Specifications therefore in this regard they are conservative for TMI-1.

In some cases these proposed Technical Specifications are more restrictive and in others less restrictive than the present Technical Specifications. As stated in our submittal of July 9, 1975 we will continue to operate within the most restrictive and conservative limits of proposed and existing Technical Specifications until the NRC Staff has had time to review and act upon this change request.

In summary, this change will assure adequate protection for the health and safety of the public in that it was derived from approved NRC guidelines.

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THREE MILE ISLAND NUCLEAR STATION UNIT I (TMI-I)
Operating License NO. DPR-50
Docket NO. 50-289

8-8-75

Technical Specification Change Request NO. 17

The licensee requests that the attached changed pages replace pages 3-16, 3-34, 3-35, 3-35a, 3-36, Figure 3.5-2A, Figure 3.5-2B, Figure 3.5-2C, Figure 3.5-2D, Figure 3.5-2E, and Figure 3.5-2F of the existing Technical Specifications.

Reasons for Proposed Change

These proposed Technical Specifications are pursuant to 10CFR50 Appendix K, your order for Modification of License dated December 27, 1974, and the concerns of your letter of June 18, 1975 and are necessary in order to ensure compliance with the Final Acceptance Criteria (FAC) of 10 CFR50.46.

Safety Analysis Justifying Change

The proposed Technical Specifications are based on the Babcock & Wilcox (B&W) ECCS evaluation model for B&W 177 fuel assembly class plants with lowered loop arrangement (BAW-10104), that has been determined by the NRC Staff to meet the requirements of Appendix K of 10CFR Part 50, and also based on the evaluation for this class of plants BAW-10103.

Further it should be noted that the analyses were conducted to satisfy NRC concerns as more explicitly expressed in May 1975 regarding partial pump operation and in this regard further information is provided in Attachment 1 to this change request.

Since the proposed Technical Specification rod position limits are based on ECCS power peaking,minimum shutdown margin, and ejected rod worth, the proposed rod position limits are based on the most limiting of the three criteria, which for TMI-1 are ECCS power peaking and ejected rod worth. The current TMI-1 Technical Specifications include restrictions on maximum ejected rod worth at hot zero and rated power in Spec. 3.5.2.3. Inclusion of the ejected rod worth criterion into the rod position limits makes the existing Technical Specification 3.5.2.3 redundant. However, Technical Specification 3.5.2.5 on Rod position limits allows the positions limits to be exceeded for a period of up to 4 hours before a violation of the Technical Specification limiting conditions for operation is considered to exist. This time has been determined on the basis that the LOCA has a very low probability of occurrence that deviations from the limits should be permitted for short periods. We consider the ejected rod accident to be on the same order of probability as the LOCA and the same deviations from limits should also be allowed. (i.e. operation in Restricted Region)

Other ECCS issues (Boron Precipitation, Single Failure Analysis and Submerged Valves) related to these proposed Technical Specifications have been addressed in our submittal of July 9, 1975. Therefore, other than Break Spectrum and Partial-Loop Operation, which is addressed above, the only remaining issue is Containment Pressure.

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3.1.7 MODERATOR TEMPERATURE COEFFICIENT OF REACTIVITY

Applicability

Applies to maximum positive moderator temperature coefficient of reactivity at full power conditions.

Objective

To assure that the moderator temperature coefficient stays within the limits calculated for safe operation of the reactor.

Specification

3.1.7.1 The moderator temperature coefficient shall not be positive at power levels above 95% of rated power.

Bases

A non-positive moderator coefficient at power levels above 95% of rated power is specified such that the maximum clad temperatures will not exceed the Final Acceptance Criteria based on LOCA analyses. Below 95% of rated power the Final Acceptance Criteria will not be exceeded with a positive moderator temperature coefficient of $+0.5 \times 10^{-4}$ $\Delta K/K/F$. All other accident analyses as reported in the FSAR have been performed for a range of moderator temperature coefficients including $+0.5 \times 10^{-4}$ $\Delta K/K/F$.

The experimental value of the moderator coefficient will be corrected to obtain the hot full power moderator coefficient. The correction factor will be verified during startup testing on earlier B&W reactors.

The Final Acceptance Criteria states that post-LOCA clad temperature will not exceed 2200 F.

REFERENCES

(1) FSAR, Section 14
(2) FSAR, Section 3

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f. If a control rod in the regulating or axial power shaping groups is declared inoperable per Specification 4.7.1.2., operation may continue provided the rods in the group are positioned such that the rod that was declared inoperable is maintained within allowable group average position limits of Specification 4.7.1.2.

g. If the inoperable rod in Paragraph "e" above is in groups 5, 6, 7, or 8, the other rods in the group shall be trimmed to the same position. Normal operation of 100 percent of the thermal power allowable for the reactor coolant pump combination may then continue provided that the rod that was declared inoperable is maintained within allowable group average position limits in 3.5.2.5.

3.5.2.3 The worth of single inserted control rods during criticality are limited by the restrictions of Specification 3.1.3.5 and the Control Rod Position Limits defined in Specification 3.5.2.5.

3.5.2.4 Quadrant tilt:

a. Except for physics tests if quadrant tilt exceeds 4 percent, power shall be reduced immediately to below the power level cutoff (see Figures 3.5-2A and 3.5-2B). Moreover, the power level cutoff value shall be reduced 2 percent for each 1 percent tilt in excess of 4 percent tilt. For less than four pump operation, thermal power shall be reduced 2 percent of the thermal power allowable for the reactor coolant pump combination for each 1 percent tilt in excess of 4 percent.

b. Within a period of 4 hours, the quadrant power tilt shall be reduced to less than 4 percent except for physics tests, or the following adjustments in setpoints and limits shall be made:

1. The protection system reactor power/imbalance envelope trip setpoints shall be reduced 2 percent in power for each 1 percent tilt.

2. The control rod group withdrawal limits (Figures 3.5-2A 3.5-2B, and 3.5-2C) shall be reduced 2 percent in power for each 1 percent tilt in excess of 4 percent.

3. The operational imbalance limits (Figure 3.5-2D) shall be reduced 2 percent in power for each 1 percent tilt in excess of 4 percent.

3.5.2.5 Control rod Positions:

a. Operating rod group overlap shall not exceed 25 percent, ± 5 percent, between two sequential groups except for physics tests.

b. Except for physics tests or exercising control rods, the control rod insertion/withdrawal limits are specified on Figure 3.5-2A (from control rod interchange up to 440 full power days of operation), Figure 3.5-2B (for after 440 full power days of operation) for four pump operation, and Figure 3.5-2C for three or two pump operation. If the control rod position limits are exceeded, corrective measures shall be taken immediately to achieve an acceptable control rod position. Acceptable control rod positions shall be attained within four hours.

c. Except for physics tests, power shall not be increased above the power level cutoff (See Figures 3.5-2A and 3.5-2B) unless the xenon reactivity is within 10 percent of the equilibrium value for operation at rated power and asymptotically approaching stability.

d. Core imbalance shall be monitored on a minimum frequency of once every two hours during power operation above 40 percent of rated power. Except for Physics tests, corrective measures (reduction of imbalance by APSR movements and/or reduction in reactor power) shall be taken to maintain operation within the envelope defined by Figure 3.5-2D. If the imbalance is not within the envelope defined by Figure 3.5-2D, corrective measures shall be taken to achieve an acceptable imbalance. If an acceptable imbalance is not achieved within four hours, reactor power shall be reduced until imbalance limits are met.

e. Safety rod limits are given in 3.1.3.5.

3.5.2.6 The control rod drive patch panels shall be locked at all times with limited access to be authorized by the superintendent.

3.5.2.7 A power map shall be taken to verify the expected power distribution at periodic intervals of approximately 10 full power days using the incore instrumentation detection system.

Bases

The power-imbalance envelope defined in Figure 3.5-2D is based on LOCA analyses which have defined the maximum linear heat rate (see Figure 3.5-2E) such that the maximum clad temperature will not exceed the Final Acceptance Criteria (2200F). Operation outside of the power imbalance envelope alone does not constitute a situation that would cause the Final Acceptance Criteria to be exceeded should a LOCA occur. The power imbalance envelope represents the boundary of operation

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limited by the Final Acceptance Criteria only if the control rods are at the withdrawal/insertion limits as defined by Figures 3.5-2A, and 3.5-2B, and if a 4 percent quadrant power tilt exists. Additional conservatism is introducted by application of:

a. Nuclear uncertainty factors.

b. Thermal calibration uncertainty.

c. Fuel densification effects.

d. Hot rod manufacturing tolerance factors.

The Rod Index versus Allowable Power curves of Figures 3.5-2A, 3.5-2B and 3.5-2C describe three regions. These three regions are:

1. Permissible operating Region

2. Restricted Regions

3. Prohibited Region (Operation in this region is not allowed)

Note: Inadvertant operation within the Restricted Region for a period of 4 hours is not considered a violation of a limiting condition for operation. The limiting criteria within the Restricted Region are potential ejected rod worth and ECCS power peaking and since the probability of these accidents is very low especially in a 4 hour time frame, inadvertant operation within the Restricted Region for a period of 4 hours is allowed.

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The 25$\pm$5 percent overlap between successive control rod groups is allowed since the worth of a rod is lower at the upper and lower part of the stroke. Control rods are arranged in groups or banks defined as follows:

Group	Function
1	Safety
2	Safety
3	Safety
4	Safety
5	Regulating
6	Regulating
7	Xenon translent override
8	APSR (axial power shaping bank)

Control rod groups are withdrawn in sequence beginning with group 1. Groups 5, 6, and 7 are overlapped 25 percent. The normal position at power is for groups 6 and 7 to be partially inserted.

The rod position limits are based on the most limiting of the following three criteria: ECCS power peaking, shutdown margin, and potential ejected rod worth. As discussed above, compliance with the ECCS power peaking criterion is ensured by the rod position limits. The minimum available rod worth, consistent with the rod position limits, provides for achieving hot shutdown by reactor trip at any time, assuming the highest worth control rod that is withdrawn remains in the full out position (1). The rod position limits also ensure that inserted rod groups will not contain single rod worths greater than: 0.65% $\Delta k/k$ at rated power. These values have been shown to be safe by the safety analysis (2) of the hypothetical rod ejection accident. A maximum single inserted control rod worth of 1.0% $\Delta k/k$ is allowed by the rod position limits at hot zero power. A single inserted control rod worth of 1.0% $\Delta k/k$ at beginning of life, hot, zero power would result in a lower transient peak thermal power and, therefore, less severe environmental consequences than a 0.65% $\Delta k/k$ ejected rod worth at rated power.

The plant computer will scan for tilt and imbalance and will satisfy the technical specification requirements. If the computer is out of service, then manual calculation for tilt above 15 percent power and imbalance above 40 percent power must be performed at least every two hours until the computer is returned to service.

The quadrant power tilt limits set forth in Specification 3.5.2.4 have been established within the thermal analysis design base using the definition of quadrant power tilt given in Technical Specifications, Section 1.6.

During the physics testing program, the high flux trip setpoints are administratively set as follows to assure an additional safety margin is provided:

Test Power	Trip Setpoint
0	<5%
15	50%
40	50%
50	60%
75	85%
>75	105.5%

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REFERENCES

(1) FSAR, Section 3.2.2.1.2

(2) FSAR, Section 14.2.2.2

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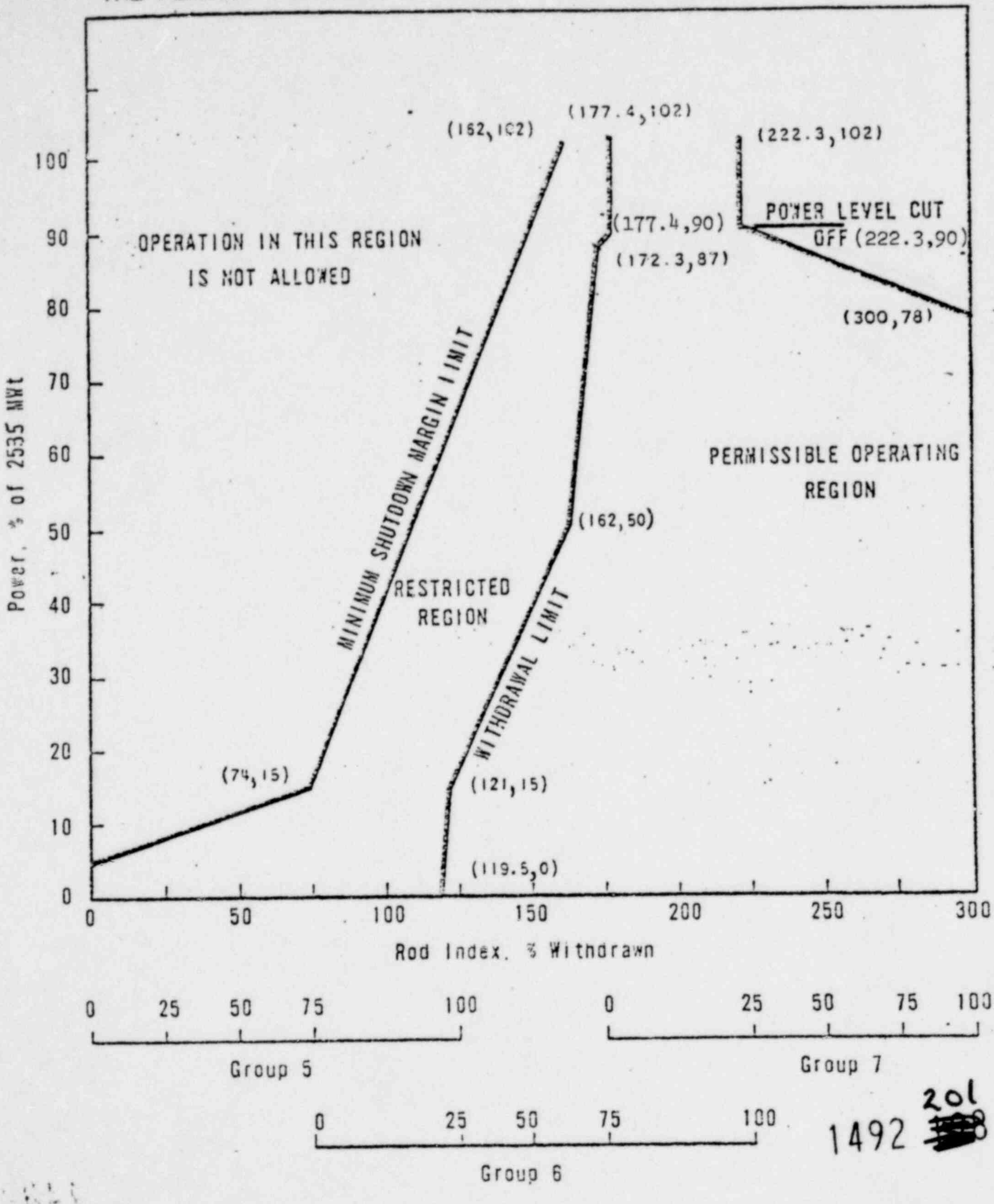
ROD POSITION LIMITS FOR 4 PUMP OPERATION APPLICABLE DURING THE PERIOD FROM 255±10 EFPD TO 440± EFPD.



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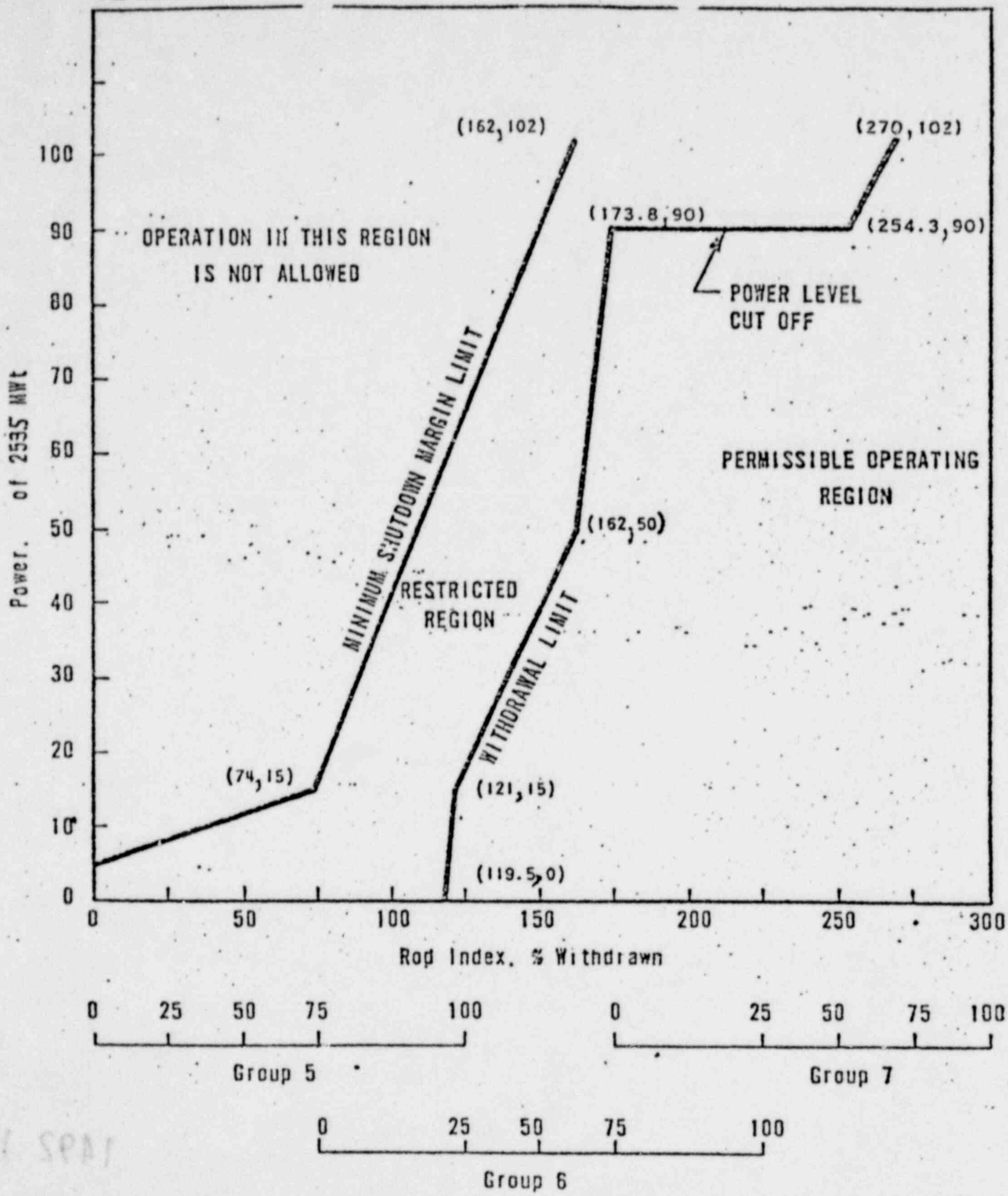
Rod index is the percentage sum of the withdrawl of Groups 5.6 and 7

UNIT 1
ROD POSITION LIMITS
Figure 3.5-2A

ROD POSITION LIMITS FOR 4 PUMP OPERATION APPLICABLE DURING THE PERIOD AFTER 4??±10 EFPD.



Rod index is the percentage sum of the withdrawl of Groups 5,6 and 7.

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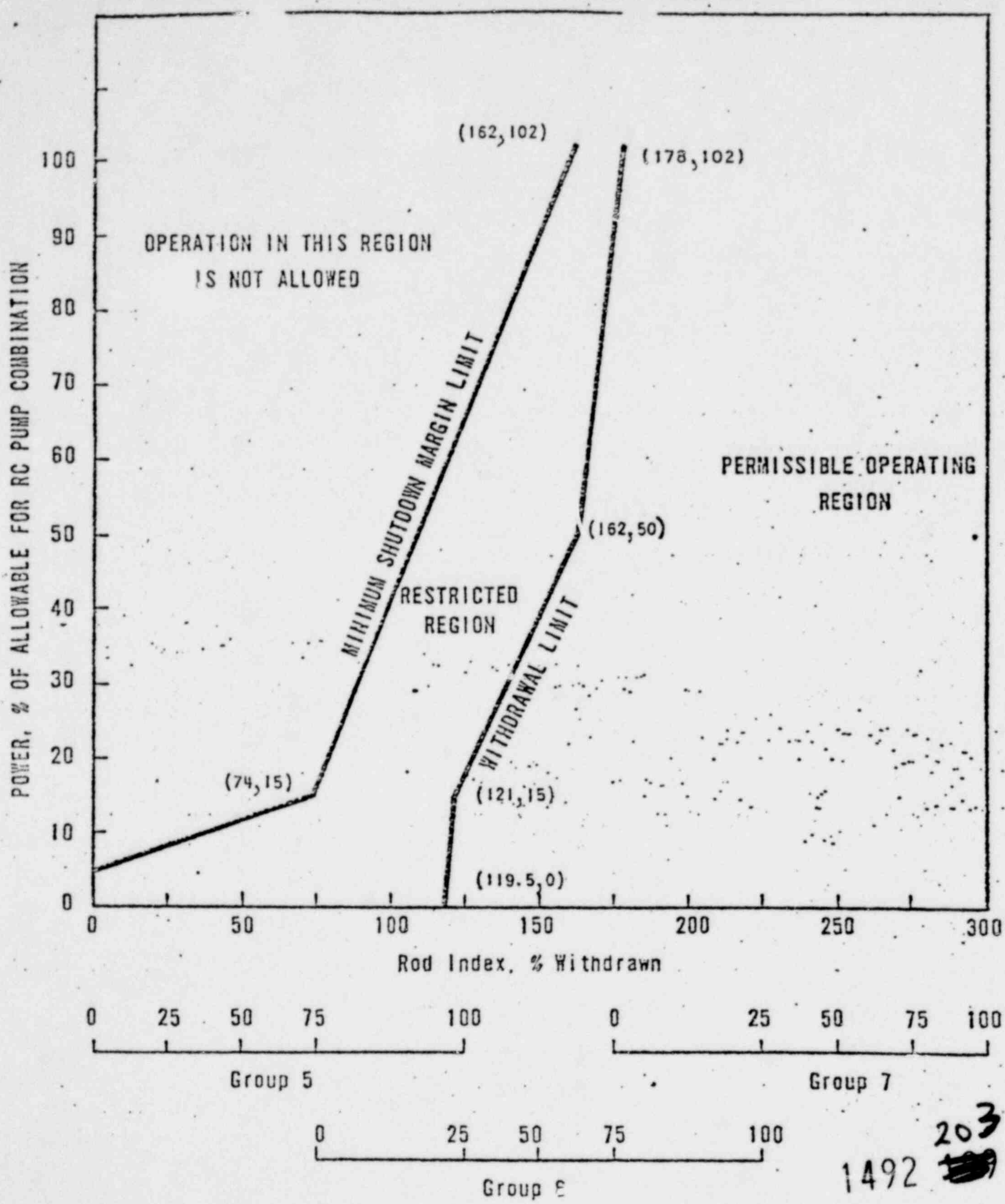
UNIT 1

ROD POSITION LIMITS

Figure 3.5-28

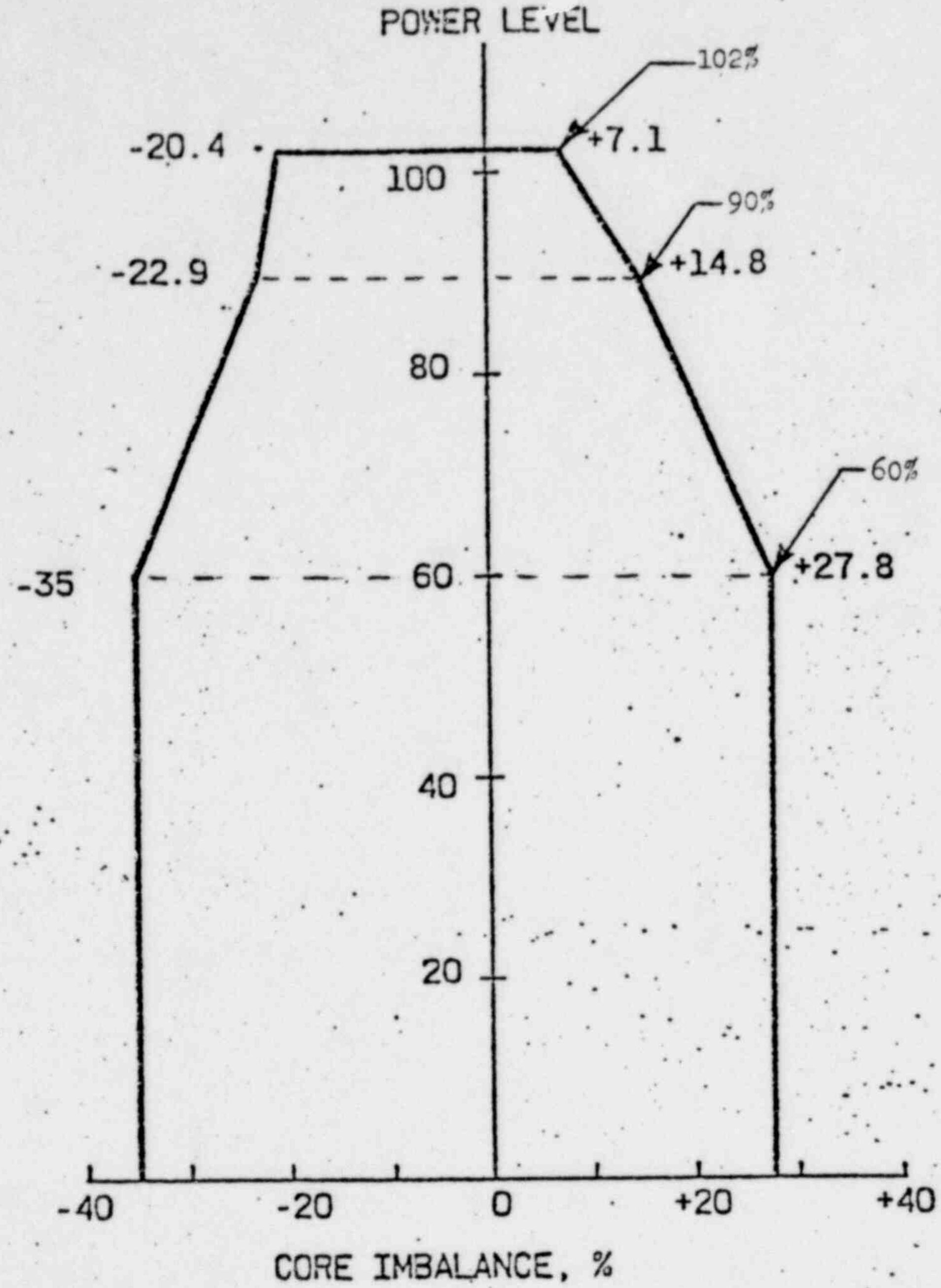


Rod index is the percentage sum of the withdrawl of Groups 5,6 and 7.

UNIT 1

ROD POSITION LIMITS

Figure 3.5-2C



OPERATIONAL POWER
IMBALANCE ENVELOPE
Figure 3.5-2D

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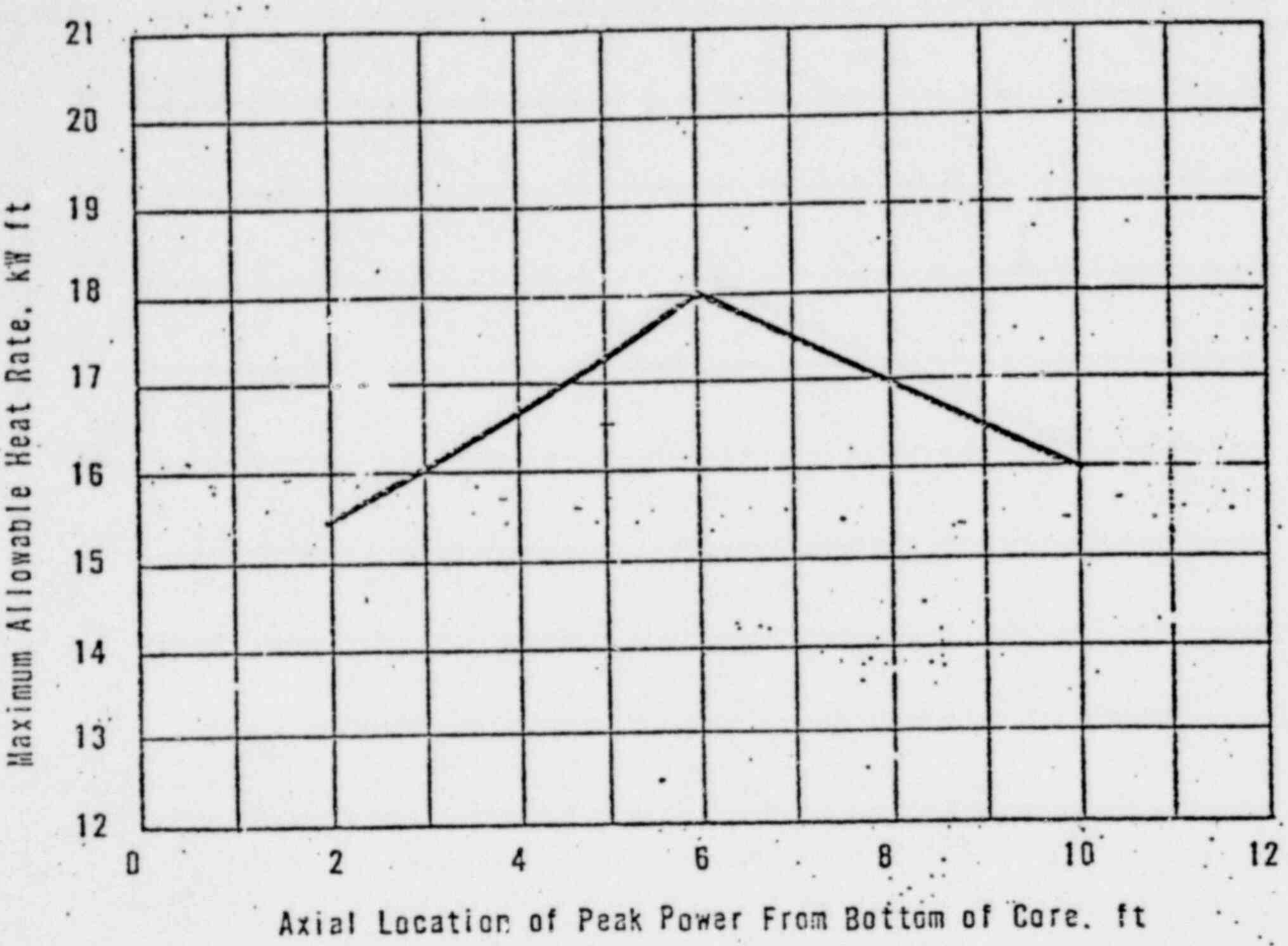

LOCA LIMITED MAXIMUM ALLOWABLE
LINEAR HEAT RATE

Figure 3.5-2E

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Attachment 1

Partial Loop LOCA Analysis

This study shows that in the event of a LOCA during partial loop operation, peak cladding temperature and Metal Water reactions are significantly lower than during four pump operation. The partial loop analysis was performed assuming the worst case break ($8.55ft^2DE$, $C_D=1$) reported in BAW-10103 at the maximum kW/ft limits allowed under Technical Specifications. The maximum cladding temperature for the partial loop LOCA analysis, is 1766F, which is 313 F less than the same break at full power and flow conditions.

There are 5 possible break configurations at the pump discharge for partial loop operation:

1. 3 - pump operation
   a. break in down loop (loop with idle pump), down cold leg (cold leg with idle pump)
   b. break in down loop, up cold leg
   c. break in up loop, up cold leg
2. 2 - pump operation, one pump up in each loop
   a. break in down cold leg
   b. break in up cold leg

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Analysis of the 3-pump operation instead of 2-pump operation was chosen for the following reasons. First, 3-pump operation is the more probable partial loop operational mode. Second, the rated power level for 3-pump operating is 77% of full power rating compared to 51% of full power rating for 2-pumps operating. The reflooding rate will be lower for higher core power, thus a greater cladding temperature rise after the End of Blowdown (EOB) is expected for 3 pumps operating.

Due to the nature of core flow which results during a cold leg break, two break locations for 3-pump operation were examined. Typically, core flow remains highly positive during the initial phase of the blowdown transient. As the head of the RC pumps degrades, due to 2-phase effects, of core flow, the magnitude of the positive core flow diminishes. Core flow then becomes negative for the remainder of the blowdown transient. The two phases of core flow, positive and negative are effected by the choice of break location. Placement of the break at the pump discharge of the

idle pump (down leg) will induce a greater driving force from the intact cold legs. This will yield high positive flows and low negative flows.

A break at the pump discharge of the down loop, up cold leg will cause a loss in positive flow during the first half of the transient. Analyzing both break locations will ensure that the most conservative assumptions effecting core flow during the blowdown transient have been considered.

The parameters used in the partial loop CRAFT and THETA models are consistent with the spectrum analysis reported in section 6 of BAW-10103, except for the following:

1. The total plant power for both cases analyzed is reduced to 77% of rated power for 3-pump operation. The peak linear heat rate for the hot bundle is the maximum kW/ft LOCA limit allowed under Tech Spec at the 6 ft elevation for this mode of operation.
2. Since there is a power imbalance between the loop with 2 RC pumps and the loop with 1 RC pump, the load ratio between the steam generators is changed to 2.27:1 by control in the feedwater flow to each steam generator.
3. The flow and pressure distribution was modeled to reflect the imbalance caused by the idle pump and the reduction in the RC flow to 75% of normal 4 pump operation. At steady state conditions the idle pump is locked in position because flow is reversed in that cold leg. The flow proceeds from the idle pump to the lower plenum of the steam generator where it mixes and proceeds back to the reactor vessel through the RC pump in the down loop, up cold leg. About 14% of the RC flow, from the downcomer plenum is directed back in the cold leg. If the flow reverses to the positive direction during the transient the idle pump would act as a free spinning rotor with no power.

Table 1 summaries the results of the partial loop analysis and compares those results to the worst break reported in BAW-10103. Figures 1 and 2 show respectively the hotspot and rupture node cladding temperature and the core flow for 3 pumps running with the break located at the up leg of the down loop. The maximum cladding temperature is 1766F at 98.5 seconds. Figures 3and 4 show respectively the hot spot and ruptured

node cladding temperature and the core flow for 3 pump running with the break located in the down leg of the down loop. The maximum cladding temperature is 1751 F at 91 seconds. Examination of the core flow for both cases reveals a distinct difference in the flow transient. With the break at the idle pump, core flow is similar to the 4-pump operation shown in figure 6-2 of BAW-10103. When the break is placed at the pump discharge of the up leg-down loop, the positive phase of the core flow is sharply reduced and the transition from positive to negative flow occurs earlier, approximately 11 seconds compared to approximately 14 seconds for the 4-pump case. The negative flow is increased due to the decrease from 3 to 2 active pumps trying to force the flow into the vessel. The flooding rates calculated using the REFLOOD code are slightly higher than those predicted for the 4 pump operation case because of the lower average core power. The hot pin cladding temperature response calculated with the THETA code are shown in Figures 1 and 3 for the two cases examined. Rupture for both cases occurs just after the EOB. The ruptured node cladding temperature decreases rapidly after rupture because of the reduced gap heat transfer from the fuel to the cladding and the increase in the surface area for cooling. The reflooding heat transfer coefficients are high enough to prevent a rise in the ruptured cladding temperature after rupture. The containment building pressure calculated by the CONTEMPT code is similar to the worst case shown in Figure 6-10 of BAW-10103.

The low temperatures experienced for the partial loop cases analysed are considerably lower than those for 4-pump operation reported in BAW-10103. The maximum cladding temperature for the partial loop LOCA analysis is only 1766F compared to 2079F for the worst 4 pump operation break as reported in Section 6 of BAW-10103. The proposed Technical Specifications for partial pump operation are calculated in a manner consistent with these results, ejected rod worth, and minimum shutdown margin.

TABLE 1

Comparison of 8.55-ft^2 DE break at pump discharge, C_D = 1.0, with 4 and 3 pumps operating.

	4-pumps(BAW-10103)	3-pumps, break in down loop, up leg	3-pumps, break in down loop, down leg
Case Number	FC 112(IL)	PP102(Y1)	PP101(1B)
Per Cent Power (100% Power =2772)	102	77	77
Peak Cladding Temp unrupt/time, F/s	2079/61.5	1766/98.5	1751/91.0
Peak Cladding Temp rupt/time, F/s	1916/43.5	1674.4/11.5	1569/42.0
Cont Pressure at Peak Cladding Temp, psia	36.4	35.37	35.48
Rupture Time/blockage S/%	13.8/63.14	25.39/65.04	25.9/64.78
CFT actuation time, s	16.7	16.6	17.2
End of bypass, s	24.4	24.8	25.2
End of Blowdown, s	24.4	24.8	25.2
End of adiabatic heatup, s	35.4	35.8	36.4
Water mass in reactor at end of blowdown, lbm	1532.0	1824	1623.0
Local metal-water reaction, %	4.2923	2.86	2.738
Full-power seconds at end of blowdown	1.949	1.874	1.959

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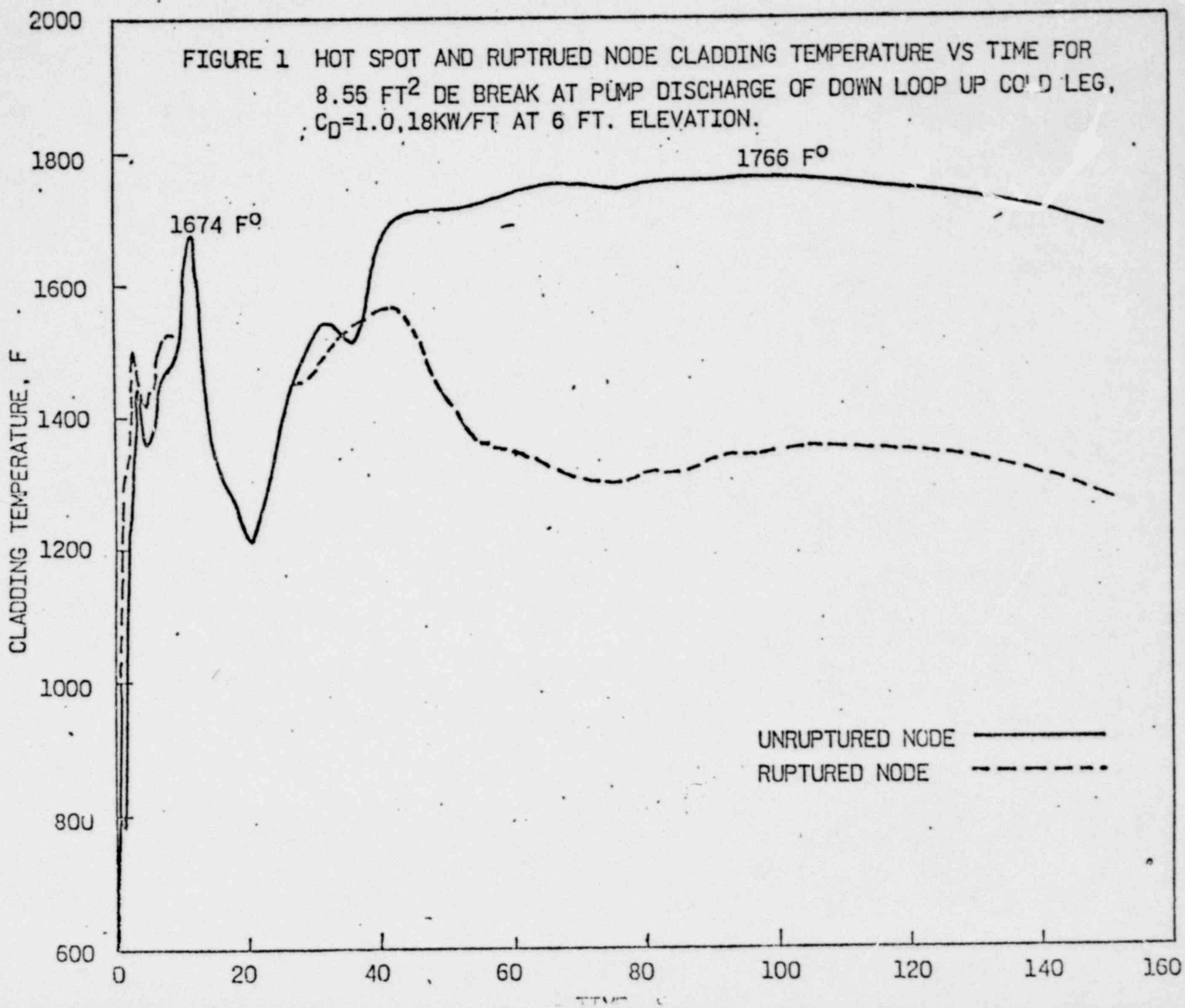

FIGURE 1 HOT SPOT AND RUPTRUED NODE CLADDING TEMPERATURE VS TIME FOR 8.55 FT^2 DE BREAK AT PUMP DISCHARGE OF DOWN LOOP UP CO'D LEG, C_D=1.0,18KW/FT AT 6 FT. ELEVATION.

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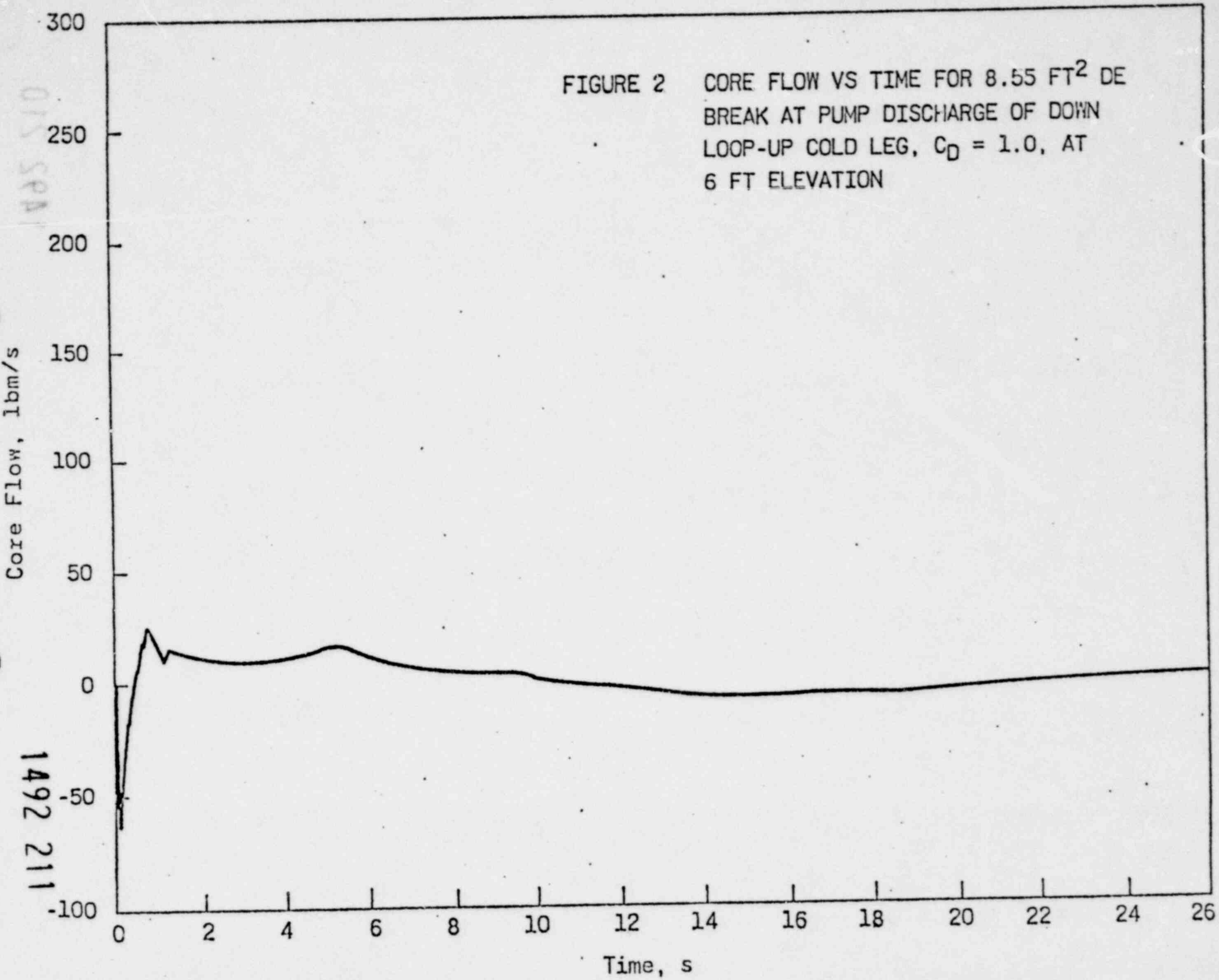

FIGURE 2 CORE FLOW VS TIME FOR 8.55 FT^2 DE BREAK AT PUMP DISCHARGE OF DOWN LOOP-UP COLD LEG, C_D = 1.0, AT 6 FT ELEVATION

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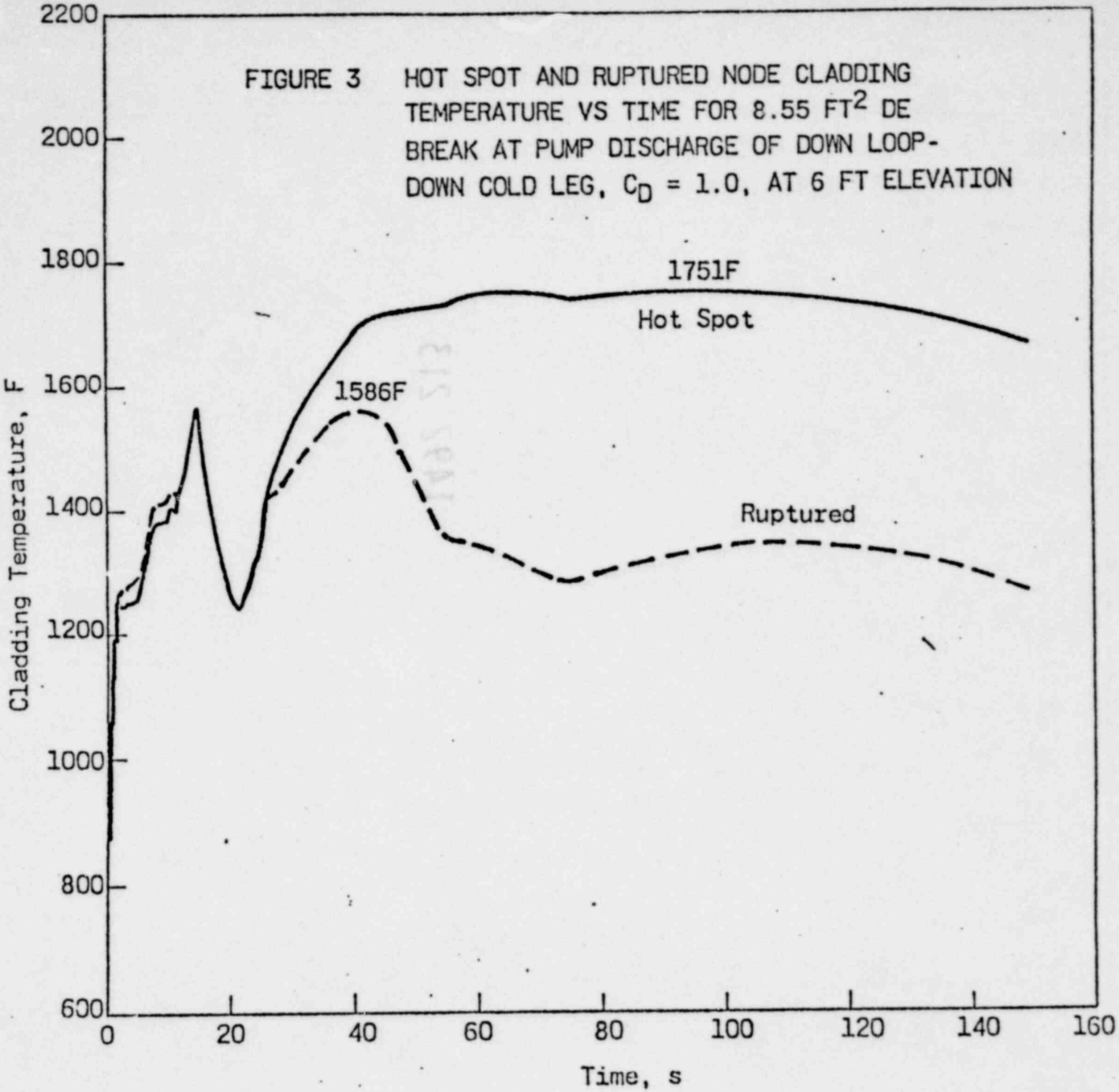

FIGURE 3 HOT SPOT AND RUPTURED NODE CLADDING TEMPERATURE VS TIME FOR 8.55 FT^2 DE BREAK AT PUMP DISCHARGE OF DOWN LOOP-DOWN COLD LEG, C_D = 1.0, AT 6 FT ELEVATION

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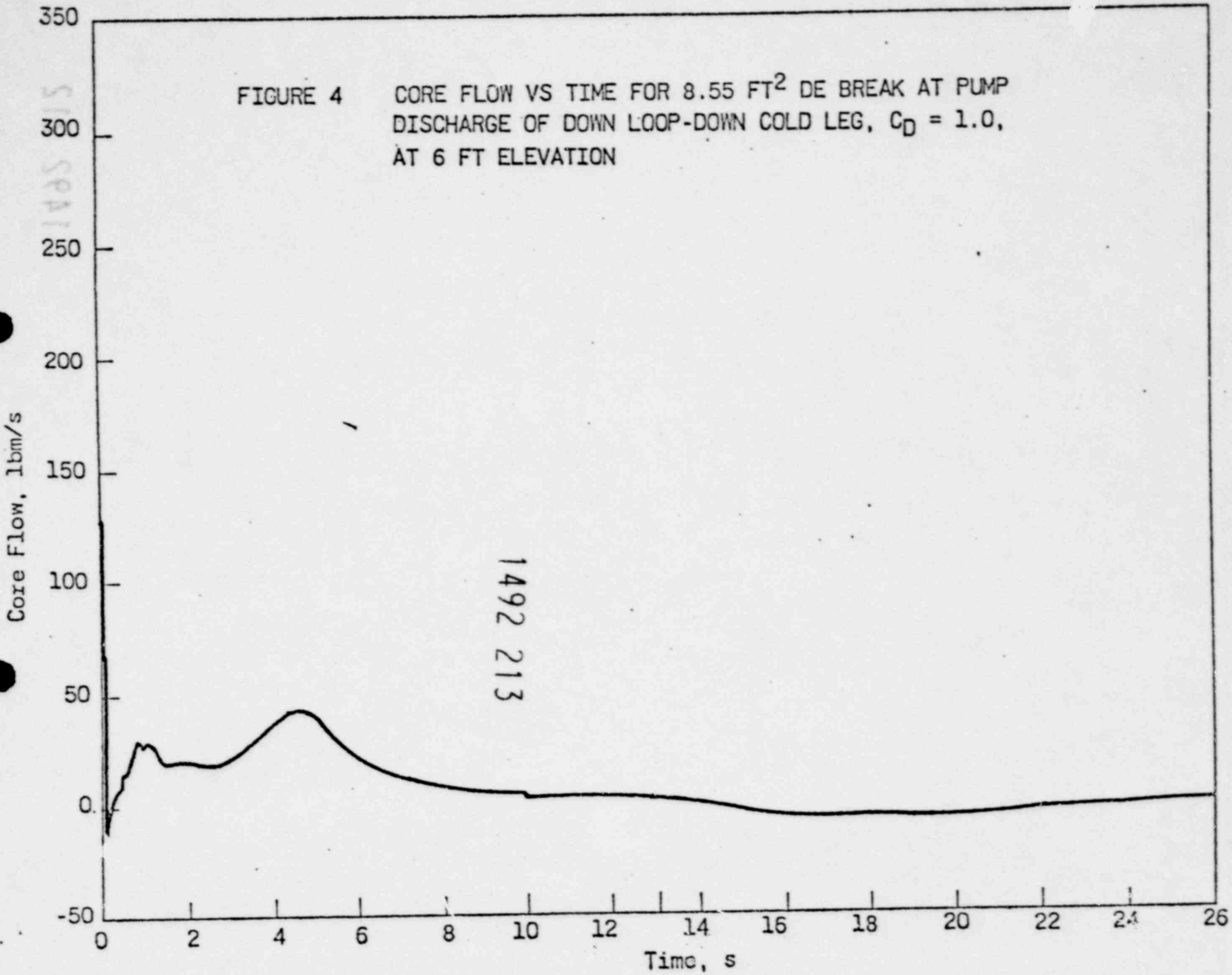

FIGURE 4 CORE FLOW VS TIME FOR 8.55 FT^2 DE BREAK AT PUMP DISCHARGE OF DOWN LOOP-DOWN COLD LEG, C_D = 1.0, AT 6 FT ELEVATION

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A Comparison of Key Parameters of the Generic Evaluation Model to Individual Plant Parameters

Parameter	Generic Model	TMI-1
Reactor Building Free Volume ft^3	2.205×10^6	2.126×10^6

a. The reactor building walls including the concrete wall, steel liner, and anchors:

Exposed area, ft^2	67,410	63,300
Paint thickness, ft	0.00083	.00083
Steel thickness, ft	0.05504	.02946
Concrete thickness, ft	4.0	3.5

The surface area is 5% larger than the largest values reported for Category 1 plants.

b. The reactor buildingdome including concrete, steel liner, and anchors:

Exposed area, ft^2	18,375	18,400
Paint thickness, ft	0.00083	.00083
Steel thickness, ft	0.06546	.02946
Concrete, ft	3.0	3.0

The surface area is 5% larger than the largest value reported for Category 1 plants.

c. Painted internal steel:

Exposed area, ft^2	249,000	311,599
Paint thickness, ft	0.00083	.00083
Steel thickness, ft	0.03125	.0227

d. Unpainted internal steel:

Exposed area, ft^2	36,000	94
Steel thickness, ft	0.03125	.0064

e. Unpainted stainless steel:

Exposed area, ft^2	10,000	42,151
Steel thickness, ft	0.03125	.0108

f. Internal concrete:

Exposed area, ft^2	160,000	118,000
Paint thickness, ft	0.00083	.00083
Concrete thickness, ft	1.0	1.54

ATTACHMENT 3

CONTAINMENT - PRESSURE COMPARISON: TMI-1 & GENERIC

Time (sec)	Pressure (psig) TMI-1	Pressure (psig) Generic
2.0	13.32	12.29
4.7	18.11	17.52
6.7	22.19	21.46
8.9	25.67	24.75
9.9	27.00	26.00
12	29.24	28.11
13	30.10	28.59
15	31.34	29.87
17	32.0[illegible]	30.41
19	32.46	30.68
21	32.49	30.59
23	31.87	29.54
25	30.67	28.71
27	29.43	27.69
29	28.38	26.61
31	27.53	25.77
33	26.74	25.11
35	26.09	24.53
37	25.53	24.08
39	25.04	23.69
42	24.[illegible]2	23.22
46	23.70	22.77
48	23.32	22.50
52	22.81	22.10
56	22.38	21.84

(sec)	Pressure (psig) TMI-1	Pressure (psig) Generic
72	21.55	21.41
76	21.42	21.31
78	21.38	21.28
82	21.27	21.21
88	21.18	21.14
92	21.10	21.07
94	21.07	21.06
96	21.03	21.03
98	21.00	20.99
105	20.91	20.30
115	20.82	20.79
125	20.69	20.69
135	20.60	20.52
145	20.47	20.38
155	20.36	20.23
165	20.25	20.09
175	20.14	19.24
185	20.01	19.79
195	19.89	19.63