AEC DISTRIBUTION FOR PART 50 DOCKET MATERIAL
(TEMPORARY FORM)

CONTROL NO: 7526

FILE:

FROM:	DATE OF DOC	DATE REC'D	LTR	MEMO	RPT	OTHER
Northern States Power Company Minneapolis, Minn. 55401 Mr. L.O. Mayer	10-4-73	10-11-73	X			

TO:	ORIG	CC	OTHER	
J.F. O'Leary	3 signed			SENT AEC PDR XXX SENT LOCAL PDR XXX

CLASS	UNCLASS	PROP INFO	INPUT	NO CYS REC'D	DOCKET NO:
	XXX			37	50-263

DESCRIPTION:
Ltr re their 9-22-73 ltr...furn add'l info concerning suppl #1 to the Request for change in Tech Specs #3.....trans the following.....

PLANT NAME: Monticello

ENCLOSURES:
Exhibit A
Tech Basis for Changes to the Tech Specs.
Figures 1 thru 8

(37 cys ea encl rec'd)

ACKNOWLEDGED
DO NOT REMOVE

FOR ACTION/INFORMATION 10-11-73 JB

BUTLER(L)	SCHWENCER(L)	✓ZIEMANN(L)	REGAN(E)
W/ Copies	W/ Copies	W/ 7 Copies	W/ Copies
CLARK(L)	STOLZ(L)	DICKER(E)	
W/ Copies	W/ Copies	W/ Copies	W/ Copies
GOLLER(L)	VASSALLO(L)	KNIGHTON(E)	
W/ Copies	W/ Copies	W/ Copies	W/ Copies
KNIEL(L)	SCHEMEL(L)	YOUNGBLOOD(E)	
W/ Copies	W/ Copies	W/ Copies	W/ Copies

INTERNAL DISTRIBUTION

✓REG FILE	TECH REVIEW	DENTON		A/T IND
✓AEC PDR	HENDRIE	GRIMES	LIC ASST	BRAITMAN
✓OGC, ROOM P-506A	SCHROEDER	GAMMILL	✓DIGGS (L)	SALTZMAN
MUNTZING/STAFF	MACCARY	KASTNER	GEARIN (L)	
CASE	KNIGHT	BALLARD	GOULBOURNE (L)	PLANS
GIAMBUSSO	PAWLICKI	SPANGLER	LEE (L)	MCDONALD
BOYD	SHAO		MAIGRET (L)	✓DUBE
MOORE (L)(BWR)	STELLO	ENVIRO	SERVICE (L)	
DEYOUNG(L)(PWR)	HOUSTON	MULLER	SHEPPARD (E)	INFO
✓SKOVHOLT (L)	NOVAK	DICKER	SMITH (L)	C. MILES
P. COLLINS	ROSS	KNIGHTON	TEETS (L)	
	IPPOLITO	YOUNGBLOOD	WADE (E)	
✓REG OPR	TEDESCO	REGAN	WILLIAMS (E)	
FILE & REGION(2)	LONG	PROJECT LDR	WILSON (L)	
MORRIS	LAINAS			
STEELE	BENAROYA	HARLESS		
	VOLLMER			

EXTERNAL DISTRIBUTION

✓1 - LOCAL PDR Minneapolis, Minn.
✓1 - DTIE(ABERNATHY)
✓1 - NSIC(BUCHANAN)
1 - ASLB(YORE/SAYRE/WOODARD/"H" ST.
✓16 - CYS ACRS ~~XXXXXXX~~ Sent to Diggs 10-11-73

(1)(2)(10)-NATIONAL LAB'S
1-R.Schoonmaker,OC,GT,D-323
1-W. PENNINGTON, Rm E-201 GT
1-CONSULTANT'S NEWMARK/BLUME/AGBABIAN
1-GERALD ULRIKSON...ORNL

1-PDR-SAN/LA/NY
1-GERALD LELLOUCHE BROOKHAVEN NAT. LAB
1-AGMED(WALTER KOESTER RM-C-427-GT
1-RD..MULLER..F-309 GT

201
8

Regulatory File Cy.

NSP

NORTHERN STATES POWER COMPANY

MINNEAPOLIS, MINNESOTA 55401

October 4, 1973

DOCKETED USAEC OCT 11 1973 REGULATORY MAIL SECTION DOCKET CLERK

RECEIVED OCT 11 1973 U.S. Atomic Energy Commission Regulatory Mail Section

Mr. J F O'Leary, Director
Directorate of Licensing
Office of Regulation
U S Atomic Energy Commission
Washington, D C 20545

Dear Mr. O'Leary:

MONTICELLO NUCLEAR GENERATING PLANT
Docket No. 50-263 License No. DPR-22

Additional Information Concerning Supplement No. 1 to
Technical Specification Change Request No. 3

On September 22, 1972 we submitted proposed Technical Specification changes which, among other items, covered Control Rod Worth and the Rod Worth Minimizer. At our May 17, 1973 meeting in your Bethesda offices, General Electric representatives summarized the nature of the Control Rod Drop Accident and its relationship to the limit on control rod worth stated in Technical Specification 3.3.B.3.(a). It was resolved that a limit on rod worth should be established which applies to the present core loading as well as anticipated reloads of Monticello and similar plants while allowing sufficient flexibility for operation within the constraints of the Specification. Since such a limit is affected by a number of parameters, it was decided that the Technical Specification Bases should discuss bounds for those parameters such that fuel would not exceed 280 calories per gram in the unlikely event of the postulated Rod Drop Accident.

The September 22, 1972 submittal was written for the Monticello core that existed at that time. The attached Exhibit A provides alternative wording for Technical Specification Bases Sections 3.3.B.3 and 4.3.B.3. This basis supports a limiting control withdrawal increment of .013 delta k rather than .015 delta k as previously proposed for Technical Specification 3.3.B.3.(a). Also attached is a document prepared by General Electric entitled "Technical Basis for Changes to Allowable Rod Worth Specified in Technical Specification 3.3.B.3.(a)." This document provides additional information on the subject and is referenced in the alternative wording for the bases.

7526

In this submittal we continue to maintain our earlier position concerning the Rod Worth Minimizer; that is, that the Rod Worth Minimizer may be bypassed and its function performed by an independent individual anytime during a startup or shutdown. Our September 22, 1972 proposal increases the frequency of surveillance required of the second individual to further assure an effective substitute for the Rod Worth Minimizer function. The revised Rod Drop Accident analysis shows that during a startup, more out-of-sequence control rods than previously predicted may exist which, if involved in the unlikely postulated accident, could cause an excess of 280 calories per gram. However, the design basis for the Rod Worth Minimizer remains unchanged from that discussed in the FSAR. The availability of the Monticello Rod Worth Minimizer during reactor startups and shutdowns since commencing commercial operation has been maintained in excess of 95%. There has been no need to excessively bypass the Rod Worth Minimizer and turn to the second individual. To require the Rod Worth Minimizer to be available for reactor startups is inconsistent with its design basis and places unreasonable demands on the reliability of non-redundant equipment.

In the intervening time prior to formal issuance of these changes, we are conforming to the more restrictive Limiting Conditions for Operation stated herein and in our September 22, 1972 submittal.

Yours very truly,

L. O. Mayer

L O Mayer, PE
Director of Nuclear Support Services

LOM/MHV/br

cc: J G Keppler
G Charnoff
Minnesota Pollution Control Agency
Attn K Dzugan

EXHIBIT A

Proposed wording to replace Technical Specification Bases Section 3.3.b.3 and 4.3.B.3 (Page 84):

3. Control rod withdrawal and insertion sequences are established to assure that the maximum in-sequence individual control rod or control rod segments which are withdrawn could not be worth enough to cause the core to be more than 0.013 delta k supercritical if they were to drop out of the core in the manner defined for the Rod Drop Accident. (3) These sequences are developed prior to initial operation of the unit following any refueling outage and the requirement that an operator follow these sequences is backed up by the operation of the RWM. This 0.013 delta k limit, together with the integral rod velocity limiters and the action of the control rod drive system, limit potential reactivity insertion such that the results of a control rod drop accident will not exceed a maximum fuel energy content of 280 cal/gm. The peak fuel enthalpy content of 280 cal/gm is below the energy content at which rapid fuel dispersal and primary system damage have been found to occur based on experimental data as is discussed in reference 1.

   Recent improvements in analytical capability have allowed more refined analysis of the control rod drop accident. These techniques have been described in a topical report and two supplements. (1) (2) (3) By using the analytical models described in these reports coupled with conservative or worst-case input parameters, it has been determined that for power levels less than 10% of rated power, the specified limit on in-sequence control rod or control rod segment worths will limit the peak fuel enthalpy content to less than 280 cal/gm. Above 10% power even single operator errors cannot result in out-of-sequence control rod worths which are sufficient to reach a peak fuel enthalpy content of 280 cal/gm should a postulated control rod drop accident occur.

---

(1) Paone, C J, Stirn, R C and Wooley, J A, "Rod Drop Accident Analysis for Large Boiling Water Reactors," NEDO-10527, March 1972.

(2) Stirn, R C, Paone, C J, and Young, R M, "Rod Drop Accident Analysis for Large BWR's," Supplement 1 - NEDO-10527, July 1972.

(3) Stirn, R C, Paone, C J, and Haun, J M, "Rod Drop Accident Analysis for Large Boiling Water Reactors Addendum No. 2 Exposed Cores," Supplement 2-NEDO-10527, January, 1973.

The following conservative or worst-case bounding assumptions have been made in the analysis used to determine the specified 0.013 delta k limit on in-sequence control rod or control rod segment worths. The allowable boundary conditions used in the analysis are quantified in reference 4. Each core reload will be analyzed to show conformance to the limiting parameters.

a. A startup inter-assembly local power peaking factor of 1.30 or less.

b. An end of cycle delayed neutron fraction.

c. A beginning of life Doppler reactivity feedback.

d. The Technical Specification rod scram insertion rate.

e. The maximum possible rod drop velocity (3.11 ft/sec).

f. The design accident and scram reactivity shape function.

g. The moderator temperature at which criticality occurs.

It is recognized that these bounds are conservative with respect to expected operating conditions. If any one of the above conditions is not satisfied, a more detailed calculation will be done to show compliance with the 280 cal/gm design limit.

In most cases the worth of in-sequence rods or rod segments will be substantially less than 0.013 delta k. Further, the addition of 0.013 delta k worth of reactivity as a result of a rod drop and in a conjunction with the actual values of the other important accident analysis parameters described above would most likely result in a peak fuel enthalpy substantially less than the 280 cal/gm design limit. However, the 0.013 delta k limit is applied in order to allow room for future reload changes and ease of verification without repetitive Technical Specification changes.

Should a control rod drop accident result in a peak fuel energy content of 280 cal/gm, less than 660 (7 x 7) fuel rods are conservatively estimated to perforate. This would result in offsite doses twice that previously reported in the FSAR, but still well below the guideline values of 10CFR100. For 8 x 8 fuel, less than 850 rods are conservatively estimated to perforate, which has nearly the same consequences as for the 7 x 7 fuel case because of the operating rod power differences.

---

(4) Report entitled, "Technical Basis for Changes to Allowable Rod Worth Specified in Technical Specification 3.3.B.3.(a)" transmitted by letter from L O Mayer (NSP) to J F O'Leary (USAEC), dated October 4, 1973.

The RWM provides automatic supervision to assure that out of sequence control rods will not be withdrawn or inserted; i.e. it limits operator deviations from planned withdrawal sequences. Reference Section 7-9 FSAR. It serves as an independent backup of the normal withdrawal procedure followed by the operator. In the event that the RWM is out of service, when required, a second independent operator or engineer can manually fulfill the operator-follower control rod pattern conformance function of the RWM. In this case, procedural control is exercised by verifying all control rod positions after the withdrawal of each group, prior to proceeding to the next group. Allowing substitution of a second independent operator or engineer in case of RWM inoperability recognizes the capability to adequately monitor proper rod sequencing in an alternate manner without unduly restricting plant operations. Above 10% power, there is no requirement that the RWM be operable since the control rod drop accident with out-of-sequence rods will result in a peak fuel energy content of less than 280 cal/gm

TECHNICAL BASIS FOR CHANGES TO ALLOWABLE ROD WORTH SPECIFIED

IN

TECHNICAL SPECIFICATION 3.3.B.3 (a)

I INTRODUCTION

A topical report and two supplements (1), (2), (3) have been issued in the last year and a half which document new techniques and models being used to analyze the Rod Drop Accident (RDA). The information in these documents has been used for the development of design approaches on new projects to make the consequences of the RDA acceptable to all concerned. In the case of the operating plants where safety analyses and resulting Technical Specifications were previously established with the old approaches, the new information in the topical reports was not easily applied. The purpose of this document is to bridge that gap by using the information and techniques developed in the referenced reports to provide a technical basis and recommended Technical Specification with the current design basis safety philosophy applied to operating plants in the RDA area.

II SUMMARY & RECOMMENDATIONS

Recommendations have been provided to operating plants previously to establish a Technical Specification for a 1.5%Δk maximum allowable worth of in-sequence control rods based on judgement application of recent RDA work. The 1.5%Δk value could typically be derived from detailed calculations on a plant-by-plant basis. However, in view of the fact that this would not be practical to do on all plants, a "worst case" comprehensive value of 1.3%Δk is recommended for general and immediate application at all operating plants. This recommendation is obtained from a comparison of available specific plant calculations, based on operating data, to those used in deriving a 280 cal/gm peak fuel enthalpy boundary for the RDA with the key parameters affecting the outcome of the RDA. The 1.3%Δk value represents a combination of conservative inputs which are inherently-fixed (e.g. use of the Doppler coefficient corresponding to a Beginning-of-Life (BOL) condition, which will always be conservative) and judgement inputs which could vary significantly in the future but are not expected to be "worse" than those picked (e.g. use of a maximum local peaking factor [P_L] of 1.30 for hot startup conditions).

---

(1)NEDO-10527 "Rod Drop Accident Analysis for Large Boiling Water Reactor", C.J. Paone, et al, 3-72
(2)Suppl. 1 to Ref. 1, 7-72
(3)Suppl. 2 to Ref. 1, 1-73

By using the conservative boundary approach, it has been determined that the offsite doses due to the RDA would increase. However, this increase would be less than double that reported in previous Safety Analysis Reports and still well within the guideline valves of 10 CFR 100.

III DISCUSSION

A. Design Basis

The design basis for evaluating the consequences of the RDA are described in the topical reports (pgs. 3/4 of ref. 3). The difference in the application of these bases between the new projects and the operating plants is in the definition of the worst single inadvertent operator error or equipment malfunction to cause the RDA. Previously for new projects and currently for the operating plants, the Rod Worth Minimizer (RWM) and operator were the redundant controls on rod selection so that a single failure could not cause the drop of an out-of-sequence rod; if the RWM were out of service, a second independent operator was acceptable as a substitute. This has not been accepted on new projects and a third system, the Rod Sequence Control System (RSCS), has been applied. Since this new system changes its mode of operation beyond the 50% rod density point, the design basis for new projects has shifted so that the drop of an out-of-sequence rod at this point is analyzed. If it cannot be assumed that the RWM or operator will prevent the selection of an out-of-sequence rod, then the worst case accident for new projects becomes the drop of an out-of-sequence rod at the point where the RSCS changes its mode of operation.

Since the contents of the topical report supplements were developed in conjunction with the new design basis on new projects, it became necessary to review and provide other means for applying the new RDA results to the current Technical Specification application on operating plants. i.e. The current Technical Specifications on operating plants are applied on the basis that the maximum reactivity value of any insequence rod must be limited in order to maintain the consequences of a RDA within those analyzed and accepted. The topical reports also covered only particular plants at particular reactivity/ exposure conditions, and since this added more variable parameters to an analysis that already contained many variables, it became necessary to develop worst case values that would assuredly cover a wide range of conditions.

In this case, available data from calculations performed for particular operating plants and conditions was compared with the same parameters used in calculating RDA consequences for the topical reports. These parameters and comparisons are described in detail below.

B. Parameters Considered & Design Assumptions Used

Although there are many input parameters to the rod drop accident analysis, the resultant peak fuel enthalpy is most sensitive to the following input parameters:

1. Steady state accident reactivity shape function
2. Total control rod reactivity worth
3. Maximum inter-assembly local power peaking factor [P_L-normalized over four bundles]
4. Delayed neutron fraction
5. Scram reactivity shape function
6. Doppler reactivity feedback
7. Moderator temperature

For fixed control rod drop velocity and scram insertion rate, these parameters can be varied and combined to yield a peak fuel enthalpy of 280 cal/gm.

Rod drop velocity was assumed to be that justified by the statistical evaluation in the appendix of Ref. (1) i.e., the maximum velocity of 3.11 ft./sec. was used. Also, the current standard Technical Specification scram times tabulated below were used in developing the scram reactivity curves for the 280 cal/gm design limit boundary corresponding to the third basic condition specified below:

%of Rod Insertion	Time from De-Energization of Scram Solenoid Valve (sec.)
5	0.475
20	1.10
50	2.0
90	5.0

In order to meet the RDA design limit of 280 cal/gm the above parameters are combined to meet three basic conditions. These are (A) the accident reactivity characteristics, (B) the Doppler reactivity feedback, and (C) the scram reactivity feedback. If any one of these conditions are not satisfied, then a more detailed analysis would have to be performed to establish compliance with the 280 cal/gm design limit.

C. Three Basic Conditions

1. Accident Reactivity Characteristics - Accident reactivity shape function, total control rod reactivity worth, inter-assembly local power peaking factor, and the delayed neutron fraction

The sensitivity of the rod drop accident to the first three parameters at cold startup and hot startup are shown by Figures 1 and 2 and the effect of the delayed neutron fraction (beta) can be seen by comparing Figures 1 and 2 with Figures 3 and 4 respectively. To determine whether or not a specific condition will meet the 280 cal./gm design limit at cold startup or hot startup, the accident reactivity characteristics (i.e., accident shape function, local peaking, etc.) for the plant being analyzed should be matched to those presented in Figures 1 through 5. If the accident reactivity characteristic curves are equal to or less than those shown as solid lines in Figures 1 through 4, then one of the three conditions needed to conservatively ensure RDA peak fuel enthalpy equal to or less than 280 cal/gm is satisfied. If the actual plant accident reactivity characteristics are greater, a more detailed analysis would have to be performed.

When applying these functions a linear interpolation can be employed to determine intermediate points with regards to the local peaking factor and beta variables.

Some example curves resulting from calculations with operating plant data are also plotted as dotted lines on Figures 3 and 4 to demonstrate compliance with the condition, including the

one with the highest $k_{eff.}$ Other data (not plotted to avoid confusion) is shown in Table 1. Comparisons have been made on Figures 3 and 4 because the betas most closely coincide. The beta for Figures 1 and 2 correspond to Beginning-of-Life (BOL) conditions which no longer exist for operating plants. Although the betas associated with the operating plant curves are not precisely the same as the value used for the 280 cal/gm boundary curves, the differences are in the conservative direction, i.e., as shown in Table I, betas for operating plant conditions are generally higher than those used in Figures 3 and 4 for the 280 cal/gm boundary curves, thus allowing higher P_{L}'s or rod worths within the boundary.

A typical plant local peaking factor map is shown in Figure 8. As can be seen the maximum value on this map is 1.217. While this is not the maximum that could be expected for a hot startup condition, values above 1.30 would not be expected to occur at any plant. Actual maximum local peaking factors (P_L) would be expected to be slightly higher in the cold startup condition than in the hot startup condition; however, as can be seen by comparison of Figures 3 and 4, a higher P_L can be tolerated for cold startup conditions at the 280 cal/gm boundary, other conditions being equal. Thus, in reviewing the compensating factors involved, it is apparent that the "worst case", or lowest rod K_{eff} allowable at the 280 cal/gm boundary would be represented by the solid curves in Figure 4, which are for the hot startup condition with the minimum beta.

2. <u>Doppler Reactivity Feedback</u>

The Doppler reactivity coefficients used for these analyses to identify a 280 cal/gm boundary were held fixed at the beginning of life (BOL) condition. The Doppler reactivity coefficients for the cold and hot startup conditions are presented in Figure 5.

If the Doppler reactivity coefficients are equal to or more negative than those given as solid lines in Figure 5, then another one of the three conditions needed to conservatively ensure RDA peak fuel enthalpy 280 cal/gm is satisfied.

Using the BOL Doppler reactivity coefficient will be conservative since the Doppler coefficient always becomes more negative with increasing exposure. This effect is typically demonstrated by the exposed core data shown as dotted lines on Figure 5, and is due primarily to the Pu-240 buildup and contribution as a function of exposure.

3. <u>Scram Reactivity Feedback</u>

The scram reactivity feedback function is unique in that the total scram feedback is not required to terminate the accident and limit peak fuel enthalpy in the time scale of interest. The combined Doppler and .01Δk scram will be more than sufficient to terminate the accident and bring the reactor core subcritical for control rod worths of interest. This is not meant to imply that total scram is not required for complete shutdown but rather to emphasize the fact that partial scram bank insertion would be sufficient to limit the resultant RDA peak fuel enthalpy to 280 cal/gm in the time scale of interest. Therefore, up to .01Δk, the actual plant scram reactivity feedback function must be equal to or greater than the data presented in Figures 6 and 7 for the cold and hot startup operating states respectively in order to satisfy the third of the three conditions needed to conservatively ensure RDA peak fuel enthalpy $\leq$ 280 cal/gm.

A typical example derived from operating plant data is also plotted on these figures as dotted lines to demonstrate that the condition is met in actual scram performance. Additional available data was not plotted to avoid graphic confusion, but is summarized with total scram worths in Table I.

D. <u>Application of the 280 cal/gm Boundary</u>

In summary, all three conditions 1, 2, and 3, as stated above, just be satisfied in order to conservatively stay within the 280 cal/gm design limit boundary. If any of the conditions are not met then a more detailed evaluation would have to be performed to demonstrate compliance with the design limit.

Likewise, given a particular set of conditions, a maximum rod worth could be determined which could show compliance with a Technical Specification based on keeping RDA consequences below the peak fuel enthalpy design limit of 280 cal/gm.

It is important to recognize that there is no practical way to analyze all possible conditions or parametric values as they may occur during the cycle at a particular plant or plants. However, some evaluations have been performed to obtain typical values as shown in this document and judgement can be exercised to obtain worst cases or perceive the effects of variations. On this basis, it would be reasonable to pick some worst case values of the key parameters in the RDA based on the approaches used in this document and derive a rod worth for Technical specification application that could be widely used without recourse to lengthy repetitive analyses for each reactor and each fuel cycle.

Such a process was conducted in the course of preparing this document, with the following results:

1. Scram reactivity condition: While there could be significant variation in the shape and total worth of the scram reactivity curve, actual operation in the future is not likely to degrade down to the point where the net effect on a RDA would be any less than that represented by the 280 cal./gm curves of Figures 6 and 7.

2. Doppler reactivity condition: The least effective (BOL) Doppler feedback has been assumed in the 280 cal/gm boundary cases adopted for this document and it would be simplest to maintain this assumption in deriving a comprehensive Technical Specification application. This conservatism would also serve to compensate for any concern in other areas where variations beyond the 280 cal/gm boundary might be postulated in extreme situations.

3. Accident reactivity characteristic condition: If it is assumed that the 280 cal/gm boundary conditions established in 1. & 2. above represent worst case values that no operating plants are likely to exceed, then selection of a recommended comprehensive Technical Specification on maximum allowable rod worth reduces to a consideration of the parameters associated with the accident reactivity characteristics discussed in C.1. above. There are four parameters considered for this 280 cal/gm boundary condition and it was established in C.1 that the closest approach of actual plant operating parameters to this 280 cal/gm boundary was represented by Figure 4. It was also established that two of the parameters, the accident reactivity shape function and beta, derived from any actual plant operating data, generally could not reach those used in calculating the 280 cal/gm boundary shown in Figure 4. Thus, the maximum allowable rod worth can be derived by determining the maximum P_L in the hot startup condition and using the corresponding solid curve. As stated in C.1, a P_L above 1.30 would not be expected at any plant and a maximum allowable rod worth would, therefore, be 1.3%Δk. This value is recommended for comprehensive Technical Specification application on a "worst case" basis in the absence of specific detailed analysis on each operating plant.

E. Effect on Accident Evaluation

In order to establish an conservative upper bound on the number of fuel rods that could fail as a result of a postulated control rod drop accident, it was assumed that a peak fuel energy content of 280 cal/gm was attained. From this analysis it was determined that conservative upper limit of 660 fuel rods would reach a fuel energy content of 170 cal/gm for 7 x 7 fuel. For 8 x 8 fuel, this number is 850 rods, however, the total quantity of fission products released by this event is about the same since the 8 x 8 fuel rods operate at a lower power. The limit of 170 cal/gm for eventual fuel cladding perforation is based upon a survey of experimental data and has been traditionally used in Safety Analysis Reports.

Safety Analysis Reports written prior to the reanalysis of the control rod drop accident based on the new models and technique reported that less than 330 fuel rod (7 x 7) would experience a fuel energy content of 170 cal/gm. Based on the difference between the boundary approach and the previous analyses, the previously reported offsite doses are doubled when the boundary approach is used. However, even with this increase, the offsite doses are still well below the guideline values set forth in 10 CFR 100.

***

TABLE I

TYPICAL RELOAD OPERATING CORES NUCLEAR DATA

A. In-Sequence Control Rod Worth

PLANT	CONDITION	POINT IN CYCLE	MAX. Δk_{eff}
A	Cold SU	BOC	0.007
B	Cold SU	BOC	0.011
B	Cold SU	EOC	0.003
C	Cold SU	BOC	0.005
B	Hot SU	BOC	0.003
C	Hot SU	BOC	0.005

B. Scram Bank Worth*

PLANT	CONDITION	POINT IN CYCLE	TOTAL NEG. Δk_{eff}
A	Cold SU	BOC	0.071
B	Cold SU	BOC	0.049
B	Cold SU	EOC	0.051
A	Hot SU	BOC	0.131
B	Hot SU	BOC	0.125
B	Hot SU	EOC	0.121
D	Hot SU	BOC	0.147
D	Hot SU	MOC	0.143
D	Hot SU	EOC	0.141

*Minus the dropping rod in the RDA

(Continued)

TABLE I

TYPICAL RELOAD OPERATING CORES NUCLEAR DATA

C. Delayed Neutron Fraction (β)

PLANT	CONDITION	POINT IN CYCLE	BETA
A	Hot SU	BOC	0.0059
A	Hot SU	EOC	0.0054
B	Hot SU	BOC	0.0059
B	Hot SU	EOC	0.0054
C	Hot SU	BOC	0.0060
C	Hot SU	EOC	0.0056

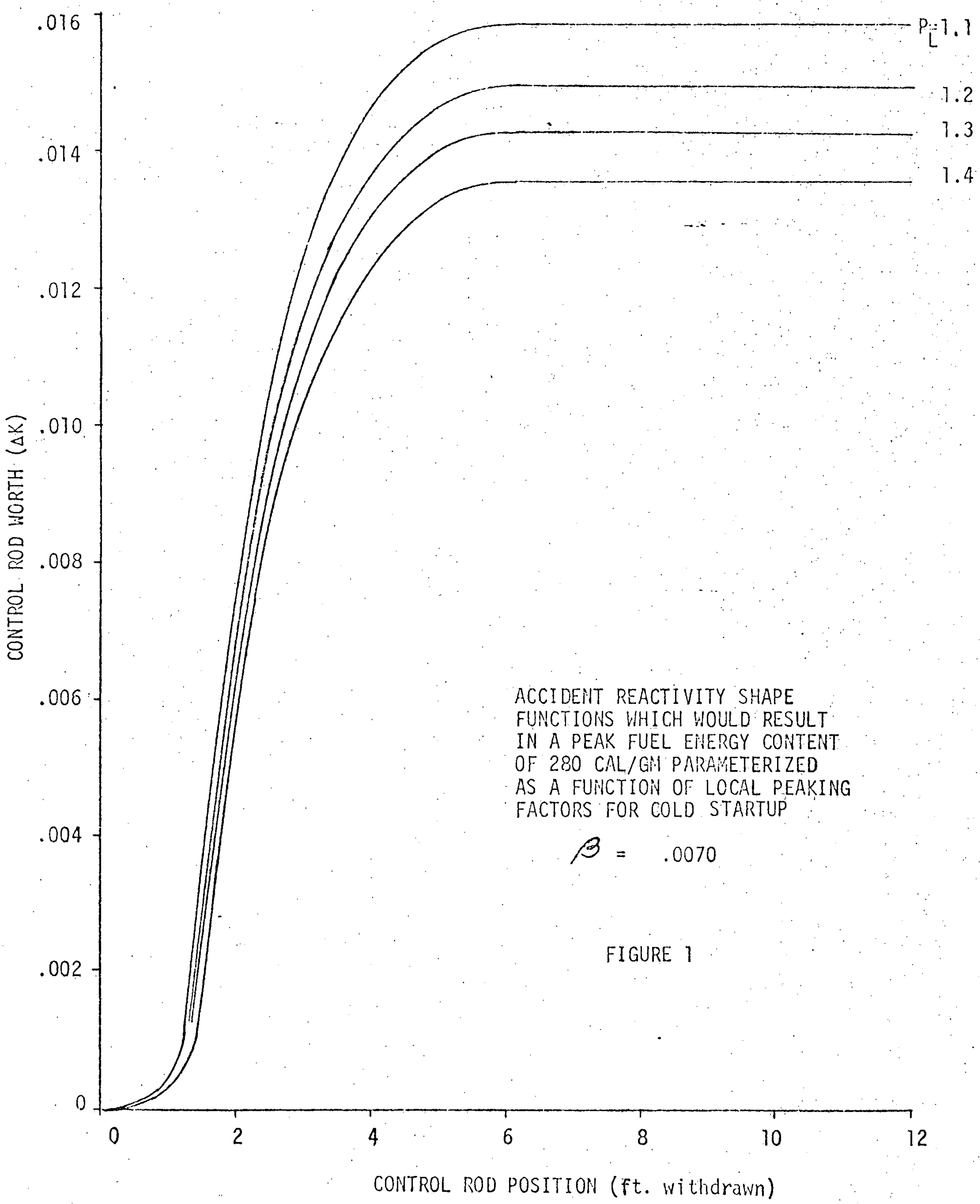
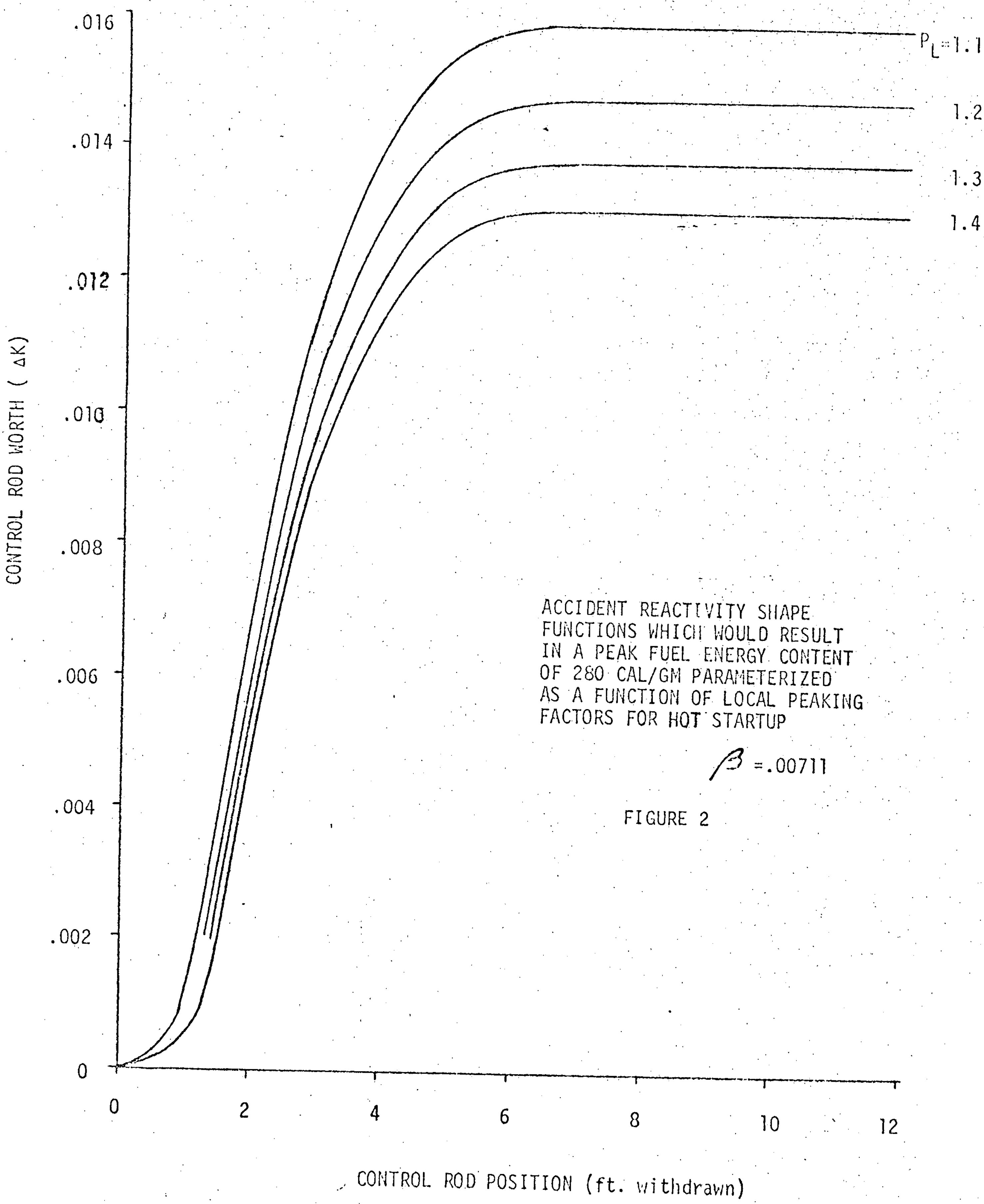

.016
.014
.012
.010
.008
.006
.004
.002
0
CONTROL ROD WORTH (ΔK)
0
2
4
6
8
10
12
CONTROL ROD POSITION (ft. withdrawn)
P_L=1.1
1.2
1.3
1.4
ACCIDENT REACTIVITY SHAPE FUNCTIONS WHICH WOULD RESULT IN A PEAK FUEL ENERGY CONTENT OF 280 CAL/GM PARAMETERIZED AS A FUNCTION OF LOCAL PEAKING FACTORS FOR COLD STARTUP
β = .0070

FIGURE 1



.016
.014
.012
.010
.008
.006
.004
.002
0
CONTROL ROD WORTH (ΔK)
0
2
4
6
8
10
12
CONTROL ROD POSITION (ft. withdrawn)
P_L=1.1
1.2
1.3
1.4
ACCIDENT REACTIVITY SHAPE FUNCTIONS WHICH WOULD RESULT IN A PEAK FUEL ENERGY CONTENT OF 280 CAL/GM PARAMETERIZED AS A FUNCTION OF LOCAL PEAKING FACTORS FOR HOT STARTUP
β =.00711

FIGURE 2

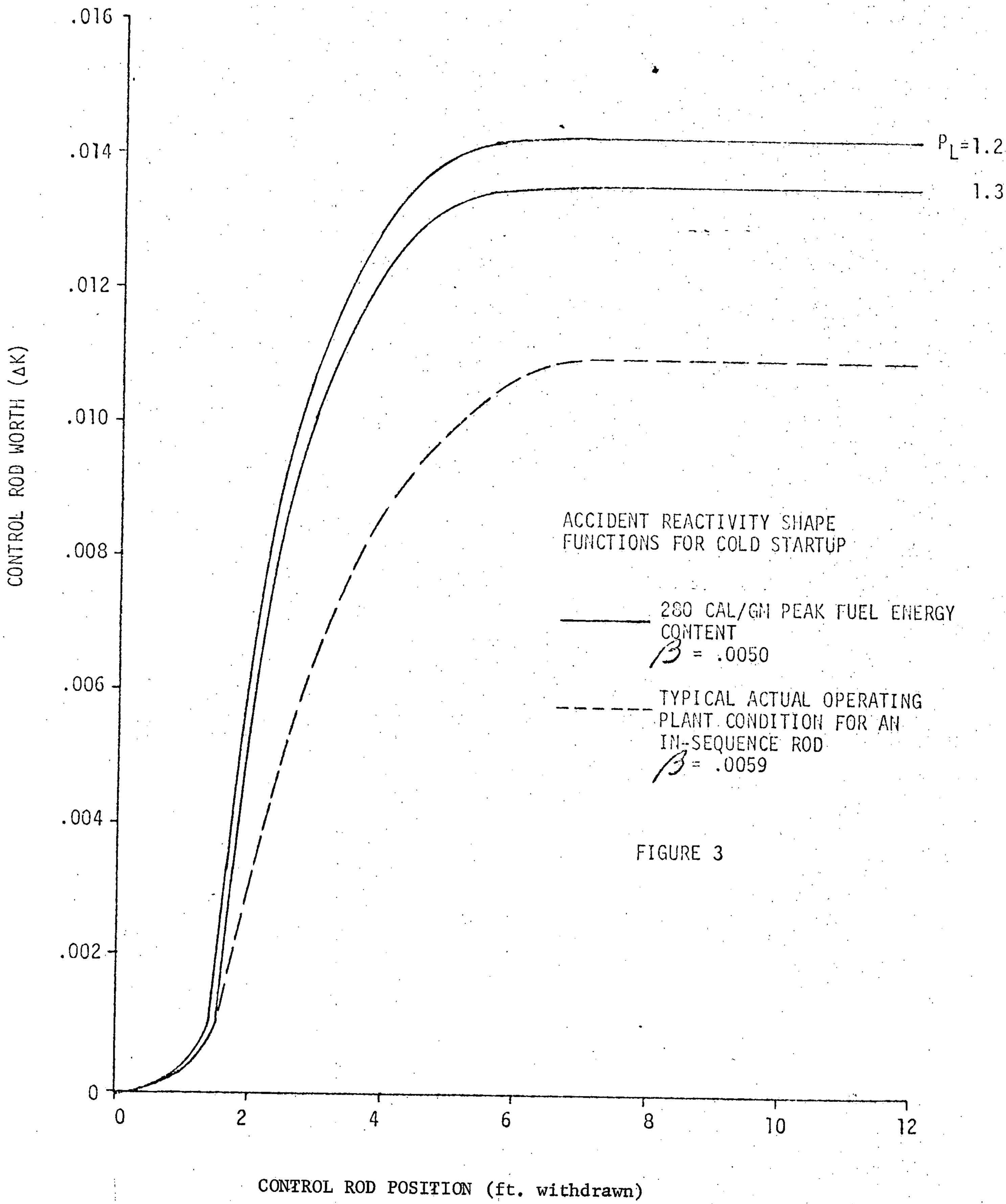

.016
.014
.012
.010
.008
.006
.004
.002
0
CONTROL ROD WORTH (ΔK)
P_L=1.2
1.3
ACCIDENT REACTIVITY SHAPE FUNCTIONS FOR COLD STARTUP
280 CAL/GM PEAK FUEL ENERGY CONTENT
β = .0050
TYPICAL ACTUAL OPERATING PLANT CONDITION FOR AN IN-SEQUENCE ROD
β = .0059
0
2
4
6
8
10
12
CONTROL ROD POSITION (ft. withdrawn)

FIGURE 3

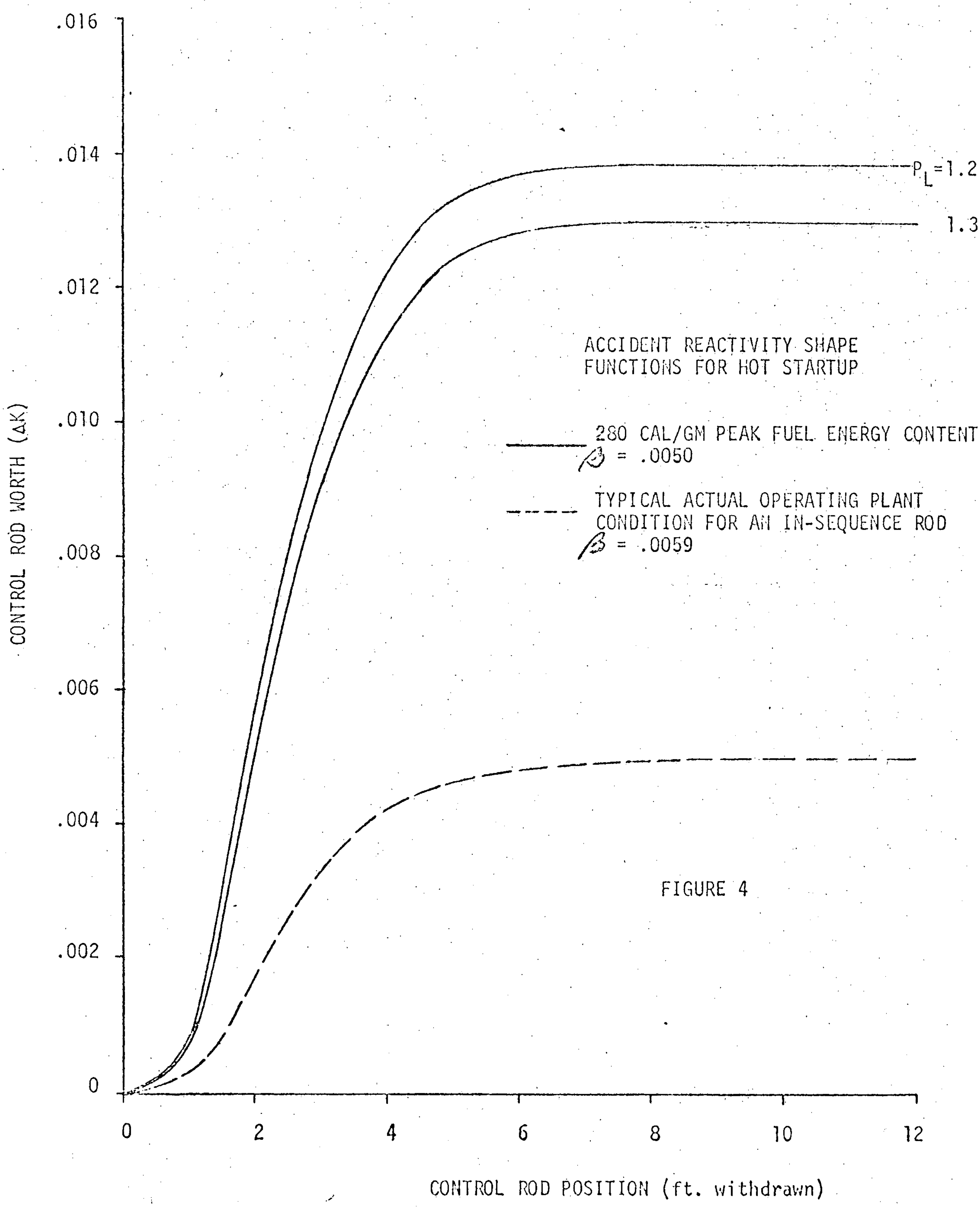
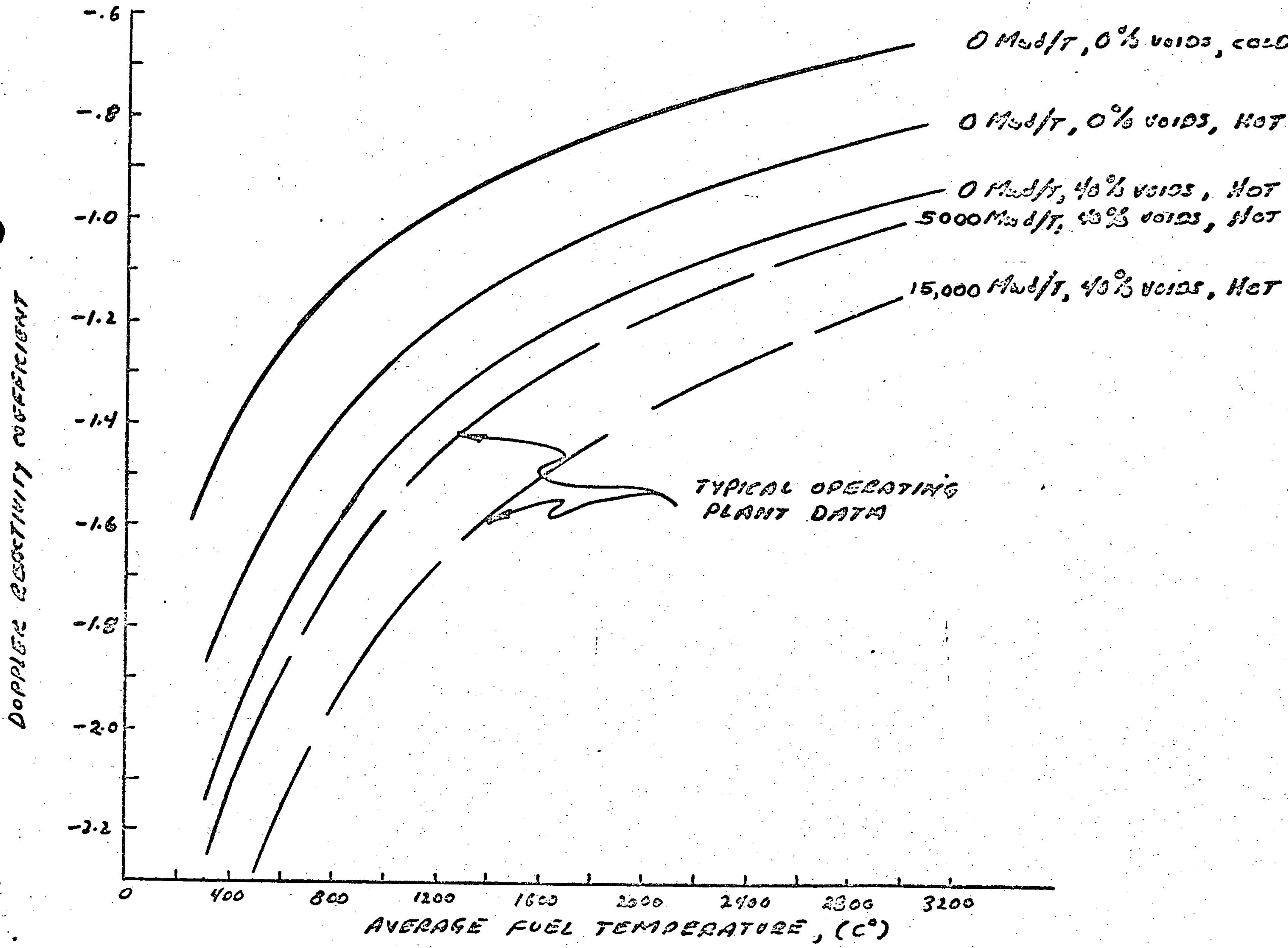

.016
.014
.012
.010
.008
.006
.004
.002
0
CONTROL ROD WORTH (ΔK)
0
2
4
6
8
10
12
CONTROL ROD POSITION (ft. withdrawn)
P_L=1.2
1.3
ACCIDENT REACTIVITY SHAPE
FUNCTIONS FOR HOT STARTUP
280 CAL/GM PEAK FUEL ENERGY CONTENT
β = .0050
TYPICAL ACTUAL OPERATING PLANT
CONDITION FOR AN IN-SEQUENCE ROD
β = .0059

FIGURE 4

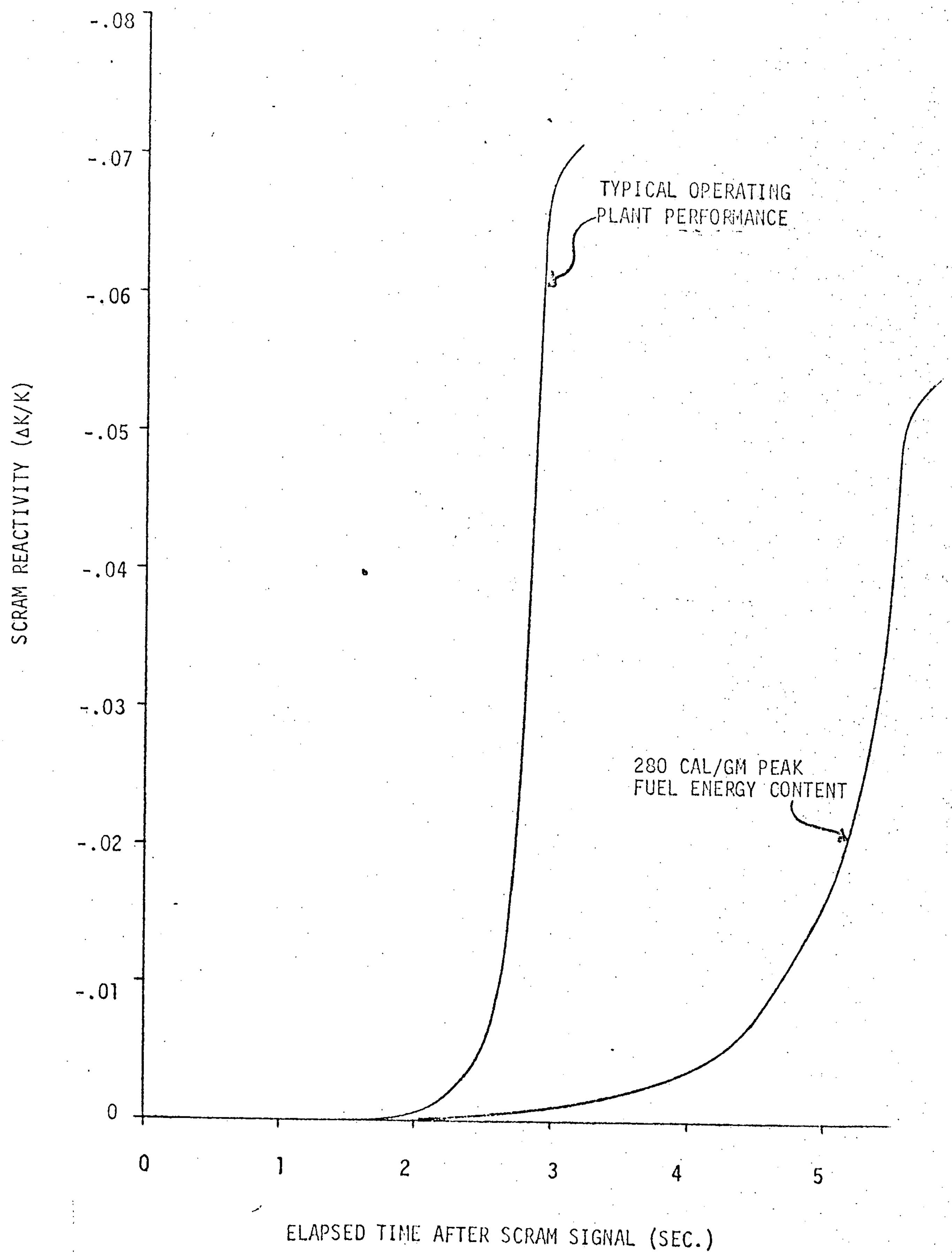
FIGURE 5

DOPPLER REACTIVITY COEFFICIENT vs AVERAGE FUEL TEMPERATURE AS A FUNCTION OF EXPOSURE AND MODERATOR CONDITION



SCRAM REACTIVITY FUNCTION FOR COLD STARTUP

FIGURE 6



SCRAM REACTIVITY FUNCTION FOR HOT STARTUP

FIGURE 7

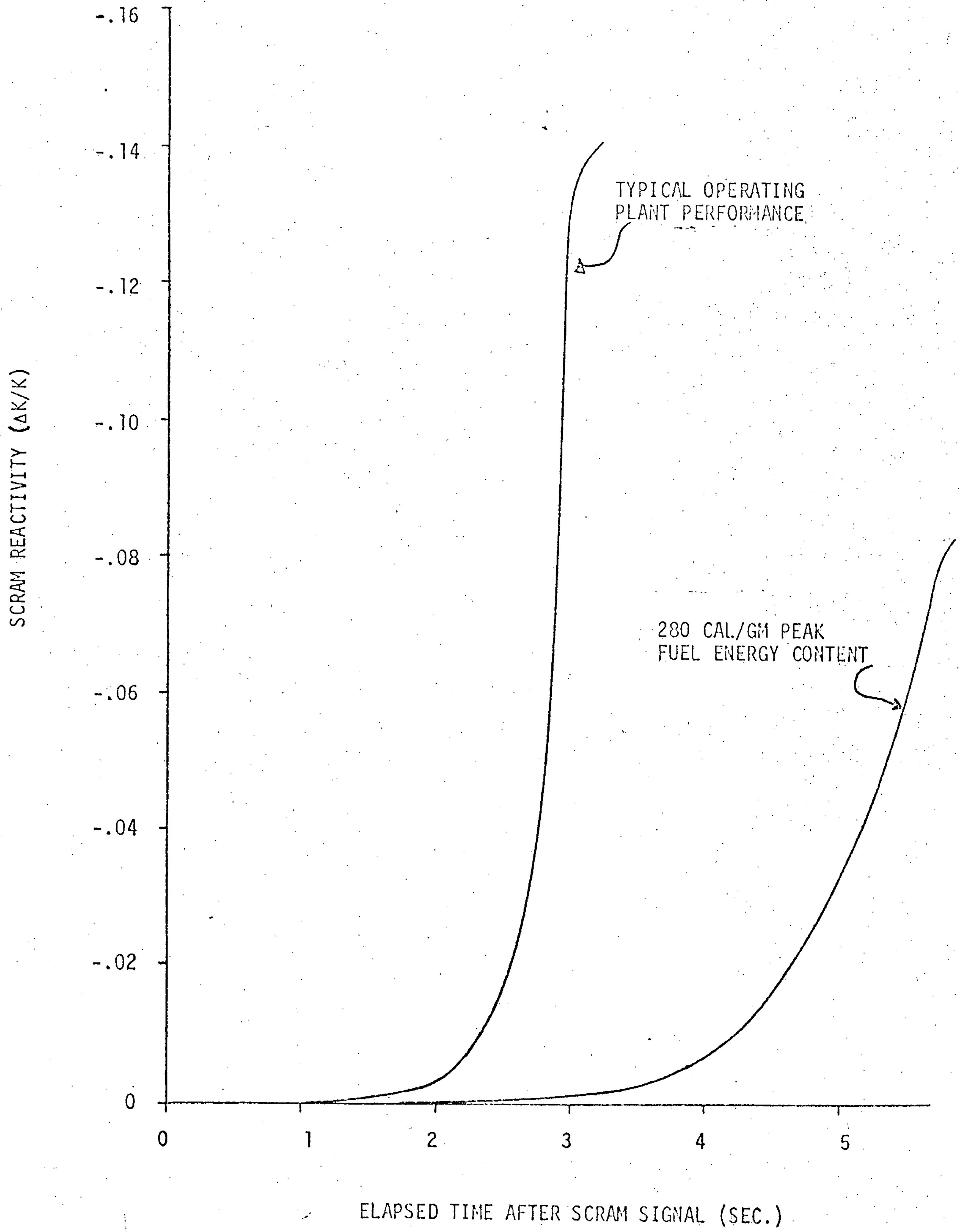


FIGURE 8

TYPICAL FOUR BUNDLE LOCAL PEAKING FACTOR MAP

HOT STARTUP - NORMALIZED TO TOTAL POWER

1.139	1.174	1.063	1.146	1.149	1.202	1.210	1.161	1.027	0.986	0.958	1.009	0.913	1.019
1.174	1.033	1.098	1.072	1.099	1.171	1.090	0.986	1.064	1.004	1.017	0.818	0.930	0.913
[illegible]063	1.098	0.285	0.860	0.907	0.953	1.181	1.140	0.969	0.905	0.892	0.961	0.818	1.009
1.146	1.072	0.860	0.830	0.806	0.279	1.071	1.075	0.915	0.854	0.855	0.892	1.017	0.958
1.149	1.099	0.907	0.806	0.828	0.888	1.114	1.080	0.918	0.855	0.854	0.905	1.004	0.986
1.202	1.171	0.953	0.279	0.888	1.014	1.217	1.120	0.982	0.918	0.915	0.969	1.064	1.027
1.210	1.090	1.181	1.071	1.114	1.217	1.138	1.001	1.120	1.080	1.075	1.140	0.986	1.161

[illegible]161	0.986	1.140	1.075	1.080	1.120	1.001	1.138	1.217	1.114	1.071	1.181	1.090	1.210
1.027	1.064	0.969	0.915	0.918	0.982	1.120	1.217	1.014	0.888	0.279	0.953	1.171	1.202
0.986	1.004	0.905	0.854	0.855	0.918	1.080	1.114	0.888	0.828	0.806	0.907	1.099	1.149
0.958	1.017	0.892	0.855	0.854	0.915	1.075	1.071	0.279	0.806	0.830	0.860	1.072	1.146
1.009	0.818	0.961	0.892	0.905	0.969	1.140	1.181	0.953	0.907	0.860	0.285	1.098	1.063
0.913	0.930	0.818	1.017	1.004	1.064	0.986	1.090	1.171	1.099	1.072	1.098	1.033	1.174
1.019	0.913	1.009	0.958	0.986	1.027	1.161	1.210	1.202	1.149	1.146	1.063	1.174	1.139