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September 14, 1973

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

Subject: Proposed Change of the Maximum Allowable In-Sequence Control Rod Worth Permitted by the Technical Specifications for Dresden Unit 3, AEC Dkt 50-249

Dear Mr. O'Leary:

Pursuant to Section 50.59 of 10 CFR 50 and Paragraph 3.B of facility operating license DPR-25, Commonwealth Edison Company hereby submits proposed changes to the Technical Specifications appended to facility operating license DPR-25. The purpose of the change is to revise the maximum reactivity that could be added by the dropout of any increment of any one control blade to a value which is justified by a rod drop analysis for a wider range of core conditions.

A topical report, NEDO-10527, "Rod Drop Accident Analysis for Large Boiling Water Reactors," and two supplements to that report have been submitted by General Electric which discuss the analysis of BWR rod drop accidents on a generic basis. As requested by D. L. Ziemann, USAEC, on February 2, 1973, Commonwealth Edison submitted on May 2, 1973 information on the rod drop analysis for Dresden Units 2 and 3 and Quad-Cities Units 1 and 2. Attached here as Exhibit A is a summary of the results of a further analysis which justify the Technical Specification change being proposed. This analysis varies from the previous one by the inclusion of a more conservative scram reactivity curve and an increased rod drop velocity.

A limit of 1.3%ΔK is being proposed as the maximum allowable reactivity that could be added by the dropout of any increment of any one control blade withdrawn in an established sequence. The analysis described in Exhibit A indicates that this amount of reactivity, coupled with other conservative inputs, will result in a peak fuel enthalpy less than or equal to 280 cal/gm, the specific energy design limit, and an offsite dose well below the guideline of 10 CFR 100. The range of conditions covered by the analysis includes all those which are expected to occur at Dresden Unit 3.

Also attached to this letter are revised Technical Specification pages which contain the proposed changes to Limiting Condition for Operation 3.3.B.3(a) and the associated bases. Revised Page 57 also contains an expanded surveillance requirement which defines Rod Worth Minimizer operability in the same manner as the existing surveillance requirement for Quad-Cities Units 1 and 2.

The analyses discussed in Exhibit A were based on the new control rod scram time specification in a Proposed Change to Appendix A of DPR-25 submitted August 25, 1973. To provide consistent Technical Specification limits, both the August 1, 1973 scram time change and this proposed change should be issued.

This proposed rod worth limit is applicable to Dresden Unit 2, Docket No. 50-237, DPR-19, and Quad-Cities Units 1 and 2, Docket Nos. 50-254 and 50-265, DPR-29 and DPR-30. The appropriate control rod drive scram time and rod worth limit Technical Specification changes will be submitted for these units by October 5, 1973.

Three signed originals and 37 copies of this submittal are being provided.

Very truly yours,

JS Abel for

Byron Lee, Jr.
Vice-President

SUBSCRIBED and SWORN to
before me this 14th day
of September, 1973.

Genevieve M. Deoner
Notary Public
My Commission Expires April 12, 1975

Exhibit A

TECHNICAL BASIS FOR CHANGES TO ALLOWABLE ROD WORTH SPECIFIED IN TECHNICAL SPECIFICATION 3.3.B.3 (a)

I INTRODUCTION

A topical report [(1)] and two supplements [(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 for recommended Technical Specifications incorporating the current RDA design basis safety philosophy applied to operating plants.

II SUMMARY & RECOMMENDATIONS

Recommendations have been previously provided to operating plants 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 be justified by 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 being recommended for general and immediate application at all operating plants. This recommendation is obtained from a comparison of the key parameters affecting the outcome of the RDA from available specific plant calculations, based on operating data, to those used in deriving a 280 cal/gm peak fuel enthalpy boundary for 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).

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(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 to the RDA analysis, it has been determined that the offsite dose due to the RDA would increase. However, the increased dose would be less than double the offsite dose reported in previous Safety Analysis Reports and still well within the guideline values 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 application of these bases depends upon the definition of the worst single inadvertent operator error or equipment malfunction to cause the RDA. For Dresden Units 2 and 3 and Quad-Cities Units 1 and 2 the Rod Worth Minimizer (RWM) and operator are redundant controls on rod selection so that a single failure cannot cause the drop of an out-of-sequence rod; if the RWM is out of service, a second independent operator is acceptable as a substitute. On new plant designs a different philosophy on redundant protection is provided. In those designs the method used to provide redundant protection changes at the 50% rod density point so the drop of a rod at this operating point must be evaluated. Since the contents of the topical report supplements were developed in conjunction with the new design basis on new plants, it was necessary to review and provide other means for applying the new RDA results to the current Technical Specification application on operating plants. The current Technical Specifications on operating plants are applied on the basis that the maximum reactivity value of any insequence rod withdrawal must be limited in order to maintain the consequences of an 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 was necessary to develop worst case values that would assure coverage of 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) are 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. 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. Doppler Reactivity Feedback

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 $\leq$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 as a function of exposure.

3. Scram Reactivity Feedback

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. Application of the 280 cal/gm Boundary

In summary, all three conditions 1, 2, and 3, as stated above, must 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% k. 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. <u>Effect on Accident Evaluation</u>

In order to establish a 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 at most 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 lower power. The limit of 170 cal/gm for eventual fuel cladding perforation is based upon a survey of experimental data and has been used in past 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 rods (7 x 7) would attain a fuel energy content of 170 cal/gm. Based on the difference between the boundary approach and the previous analyses, the previously estimated offsite dose is doubled when the boundary approach is used. However, even with this increase the offsite dose is still well below the guideline values set forth in 10 CFR 100.

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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

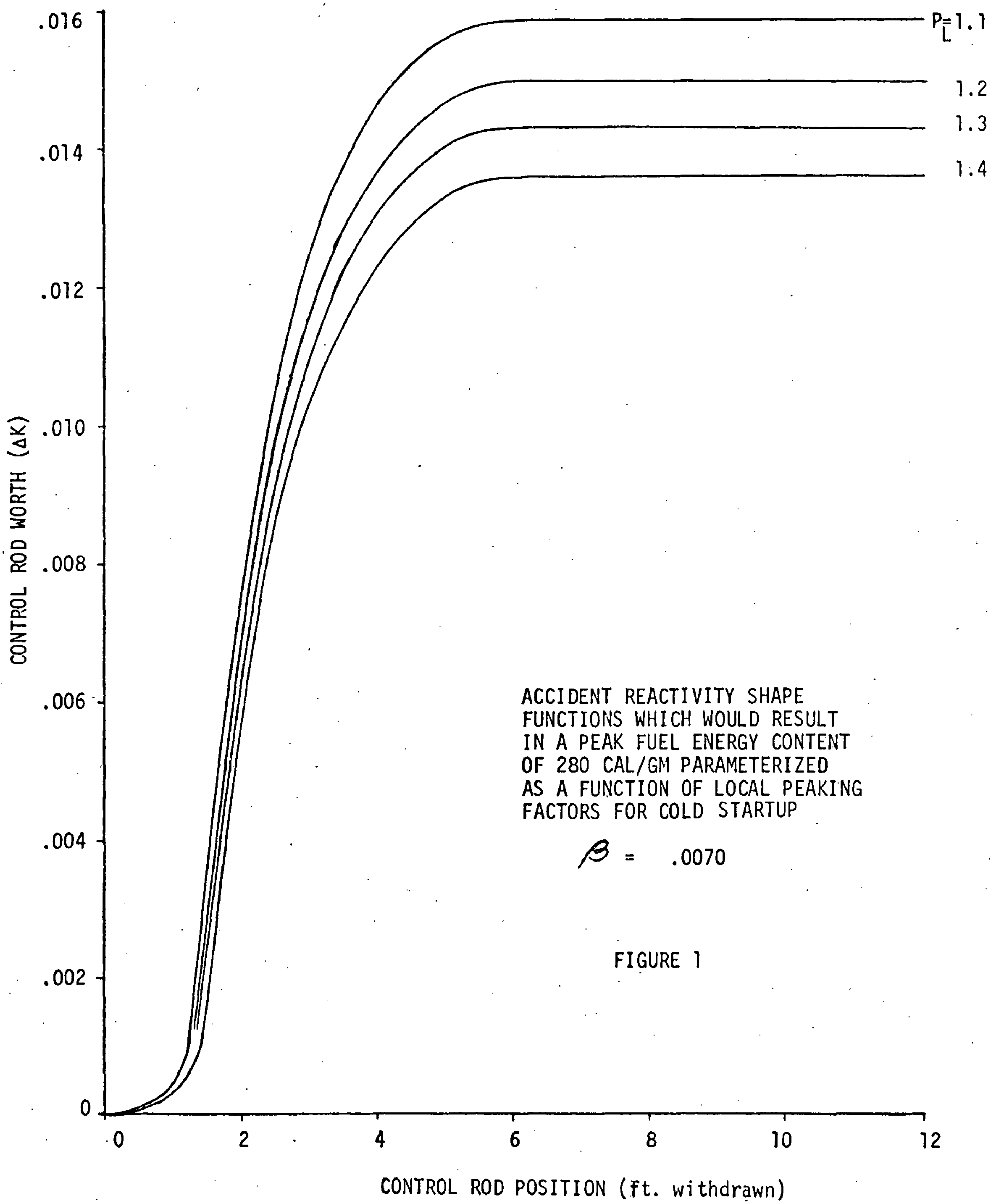
(Continued)

TABLE I

TYPICAL RELOAD OPERATING CORES NUCLEAR DATA

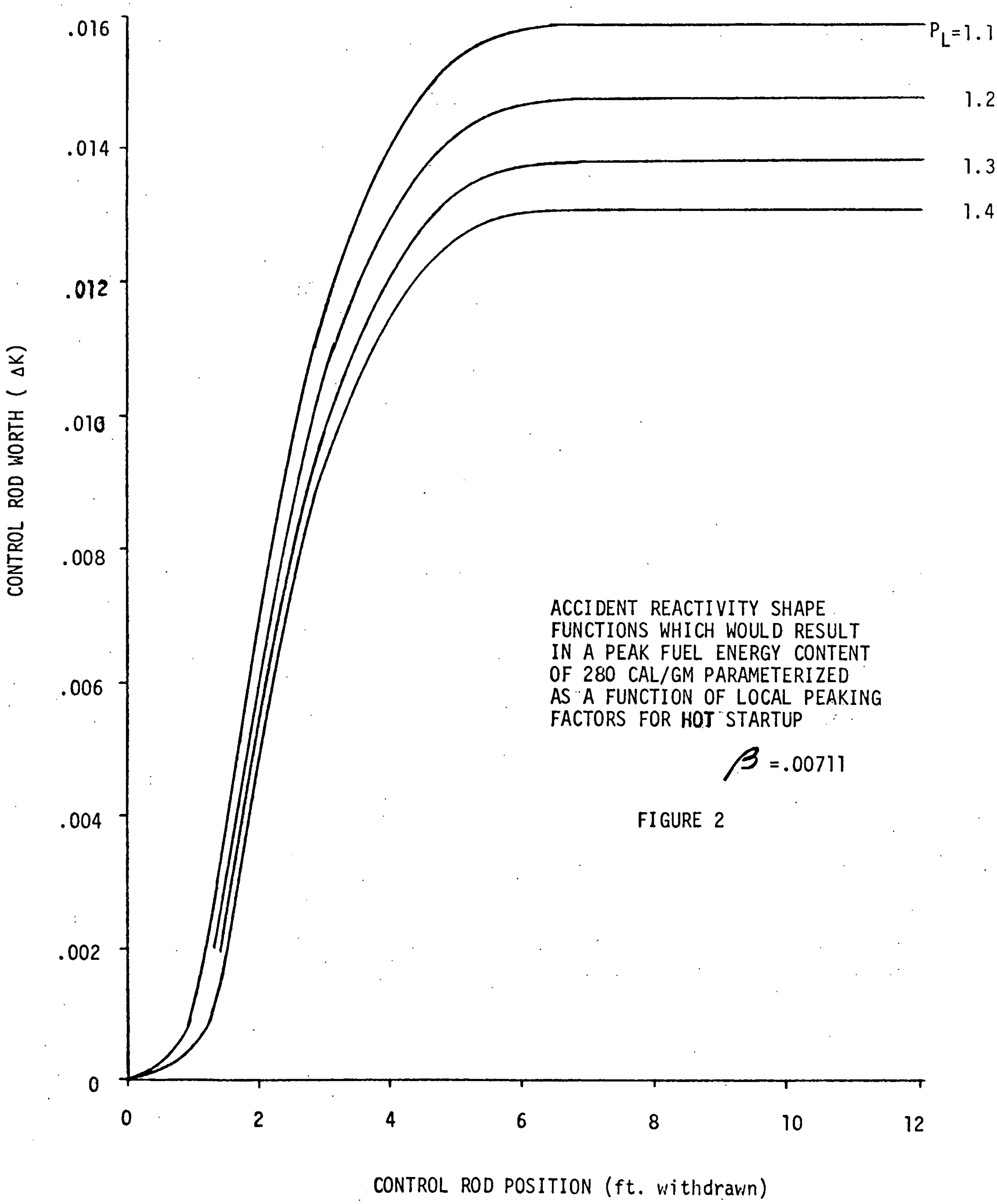
C. Delayed Neutron Fraction (β)

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



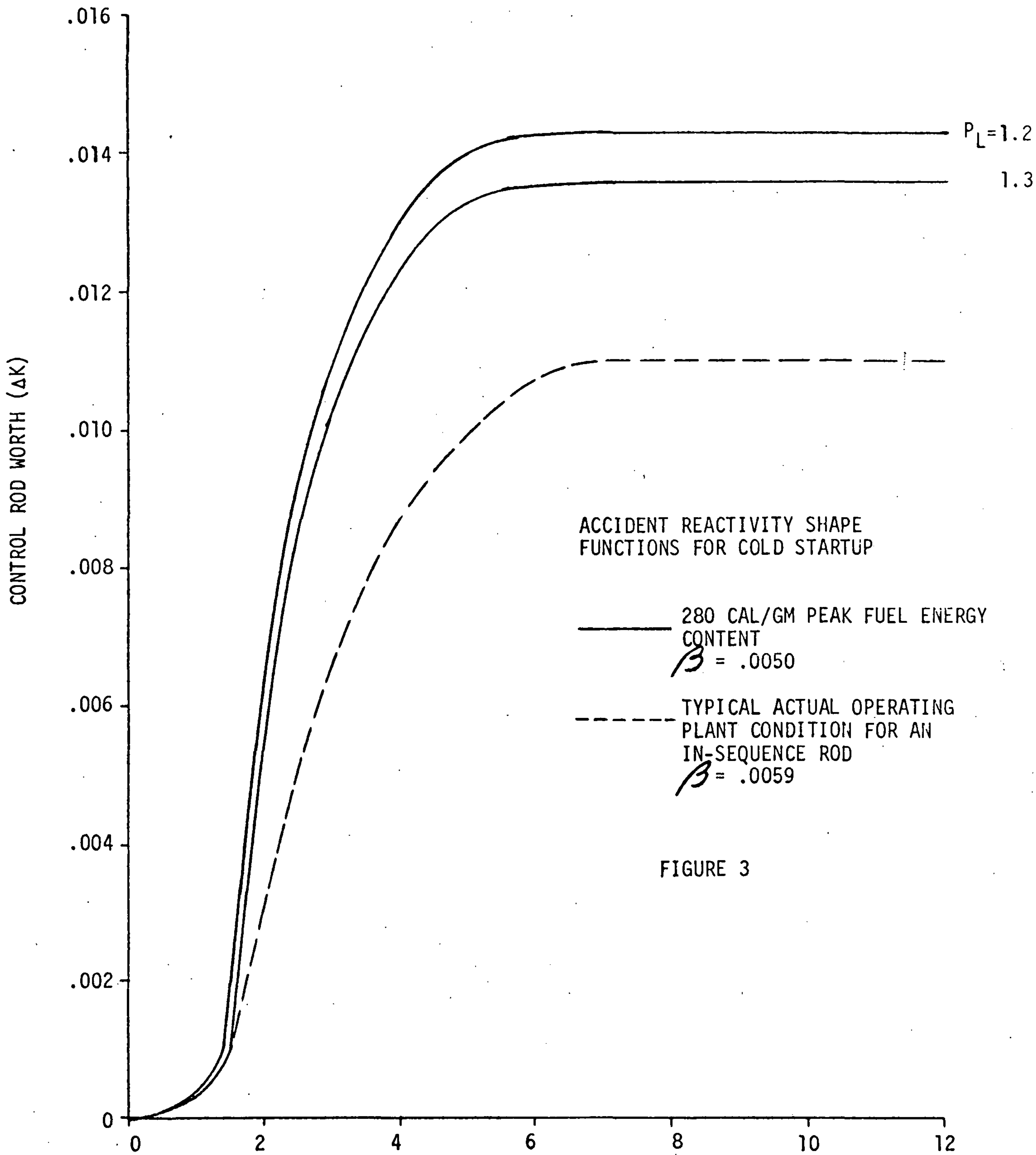
.016
.014
.012
.010
.008
.006
.004
.002
0
CONTROL ROD WORTH (ΔK)
0
2
4
6
8
10
12
CONTROL ROD POSITION (ft. withdrawn)
P_L=1.1
1.2
1.3
1.4
ACCIDENT REACTIVITY SHAPE
FUNCTIONS WHICH WOULD RESULT
IN A PEAK FUEL ENERGY CONTENT
OF 280 CAL/GM PARAMETERIZED
AS A FUNCTION OF LOCAL PEAKING
FACTORS FOR COLD STARTUP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

FIGURE 1



.016
.014
.012
.010
.008
.006
.004
.002
0
CONTROL ROD WORTH (ΔK)
P_L=1.1
1.2
1.3
1.4
0
2
4
6
8
10
12
CONTROL ROD POSITION (ft. withdrawn)
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



CONTROL ROD WORTH (ΔK)
.016
.014
.012
.010
.008
.006
.004
.002
0
0
2
4
6
8
10
12
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

FIGURE 3

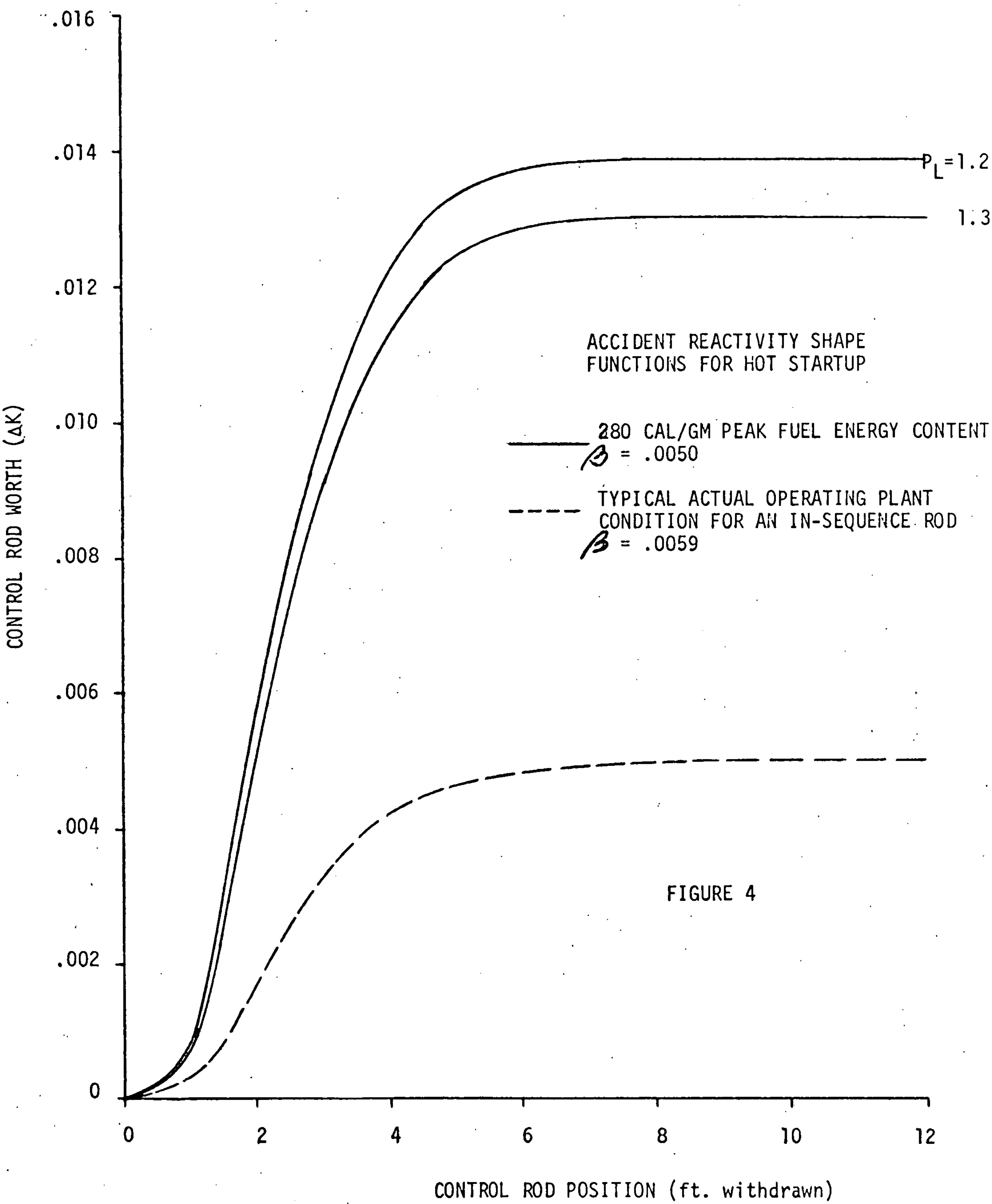

.016
.014
.012
.010
.008
.006
.004
.002
0
CONTROL ROD WORTH (ΔK)
P_L=1.2
1.3
ACCIDENT REACTIVITY SHAPE FUNCTIONS FOR HOT STARTUP
280 CAL/GM PEAK FUEL ENERGY CONTENT
β = .0050
TYPICAL ACTUAL OPERATING PLANT CONDITION FOR AN IN-SEQUENCE ROD
β = .0059
0
2
4
6
8
10
12
CONTROL ROD POSITION (ft. withdrawn)

FIGURE 4

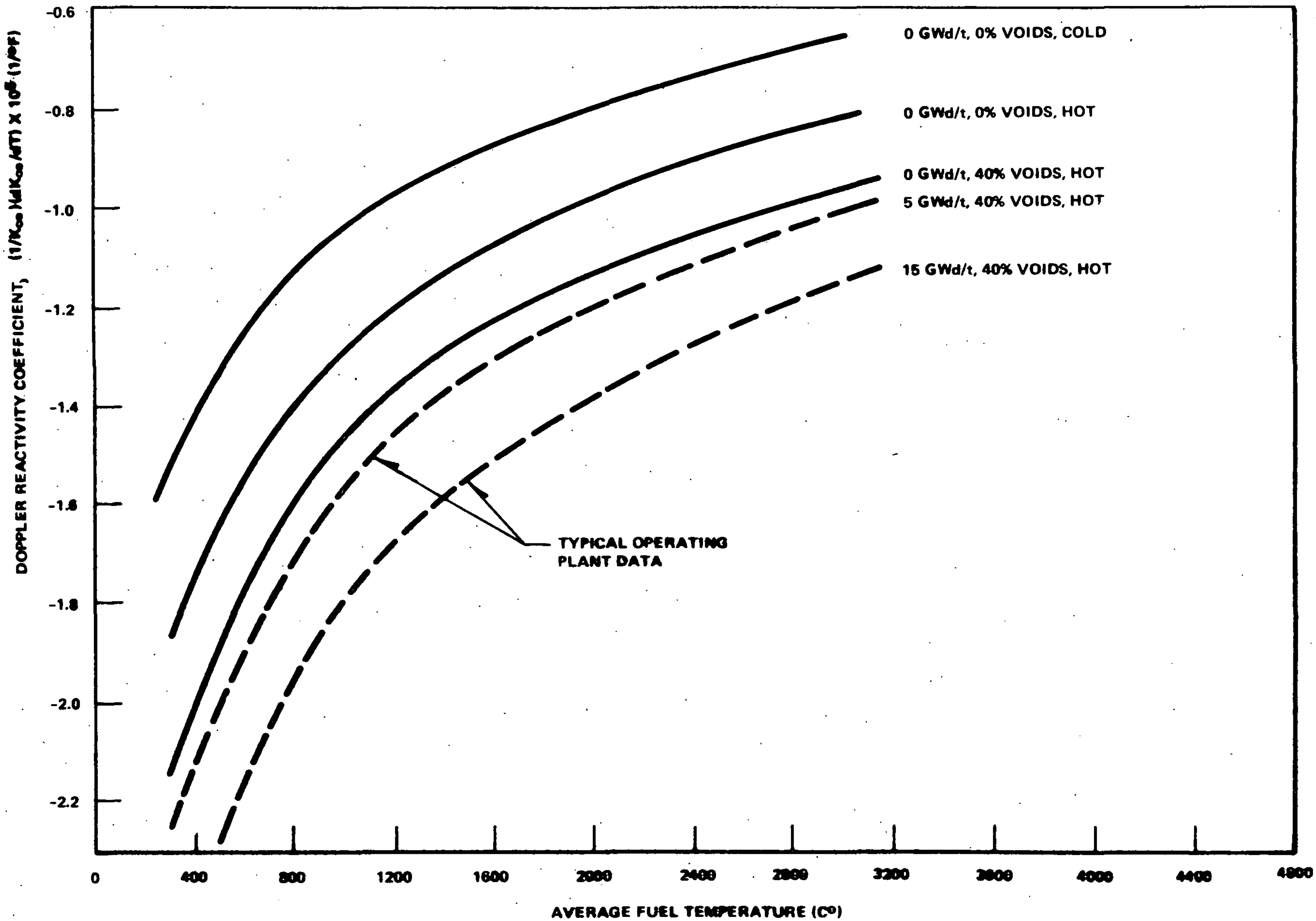
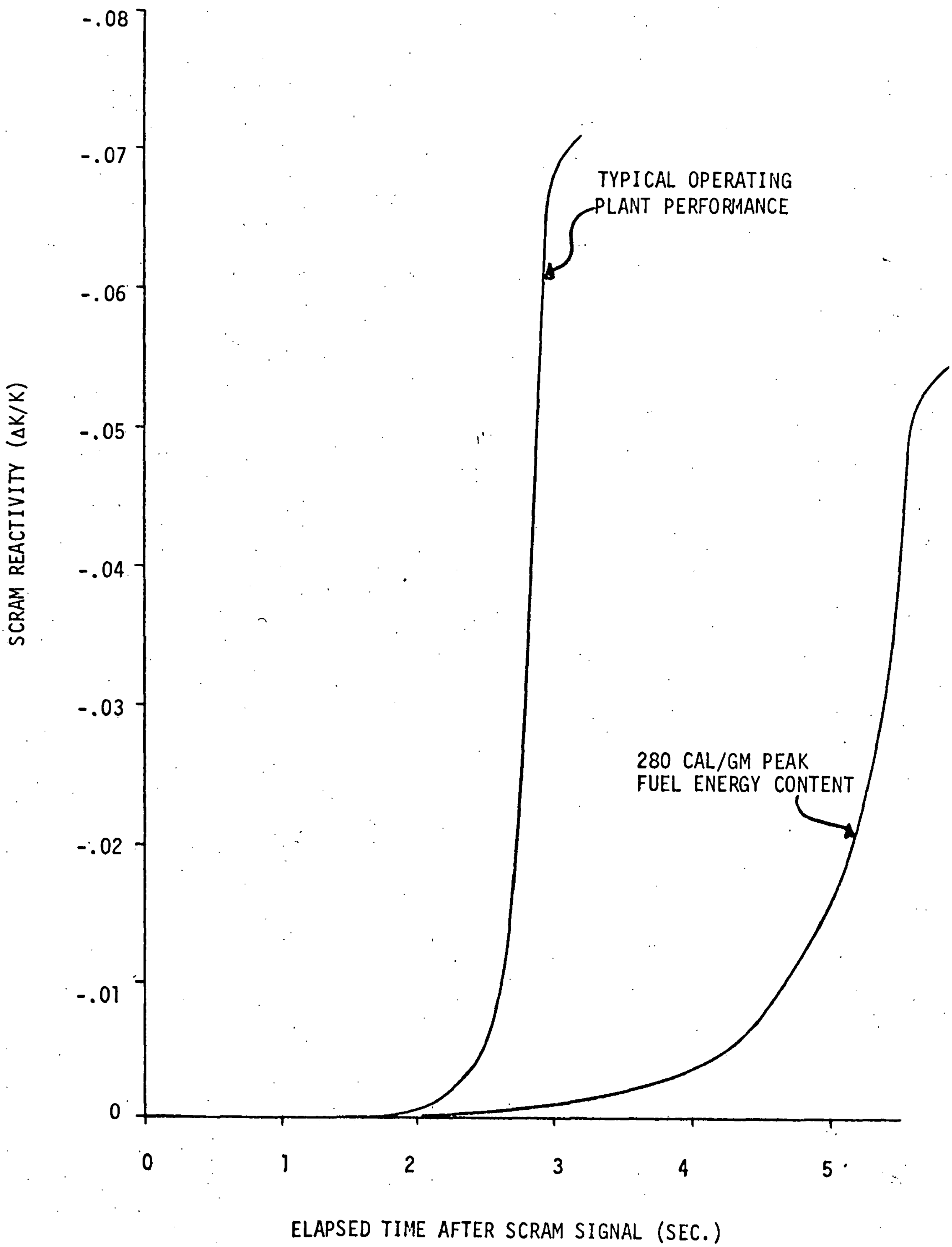


Figure 5. Doppler Reactivity Coefficient vs Average Fuel Temperature as a Function of Exposure and Moderator Condition

SCRAM REACTIVITY FUNCTION FOR COLD STARTUP

FIGURE 6



SCRAM REACTIVITY FUNCTION FOR HOT STARTUP

FIGURE 7

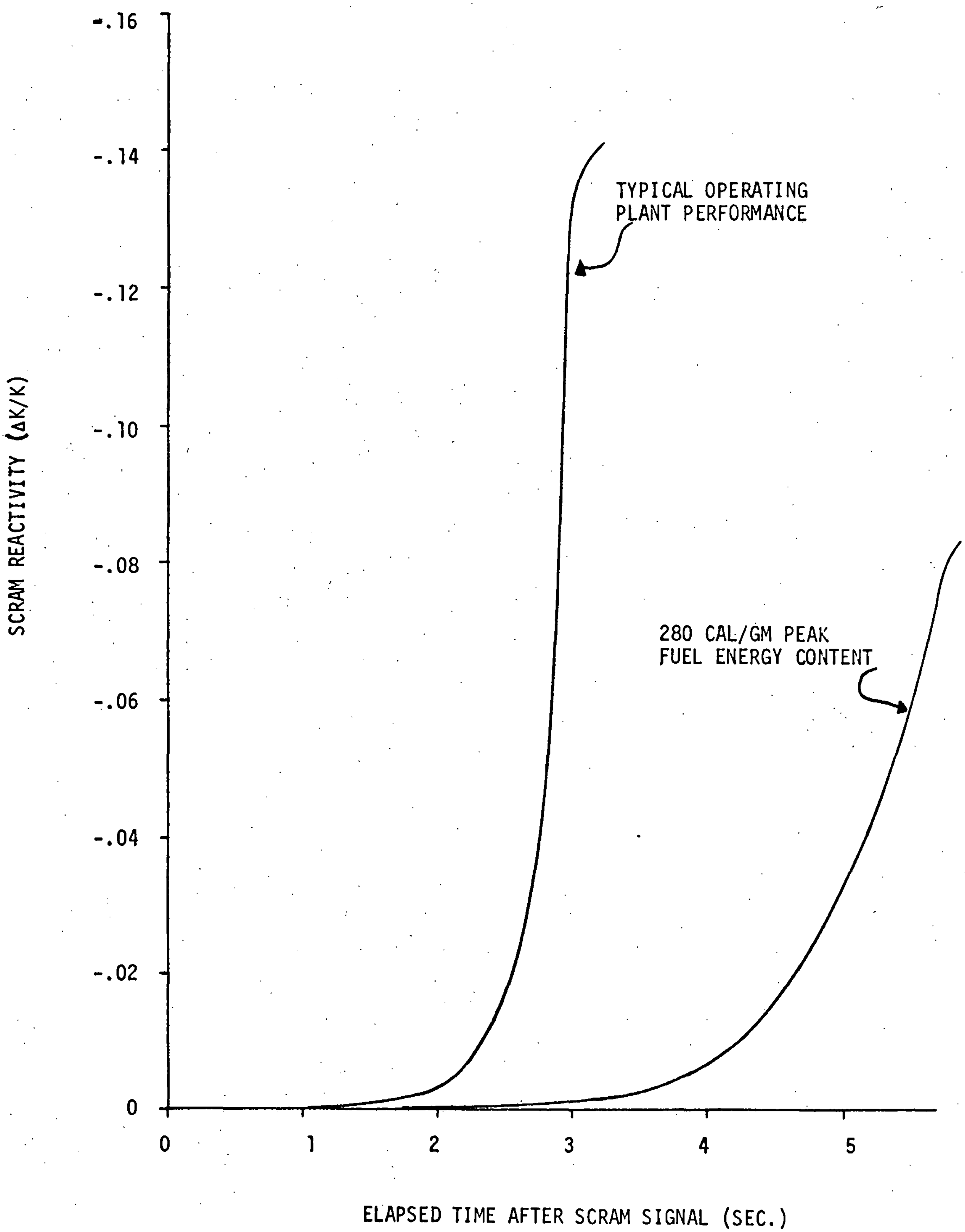


FIGURE 8

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
1.063	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

1.161	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

ATTACHMENT 1

Revised Pages for Technical Specifications

Appended to Facility Operating Licenses

DPR-19 and DPR-25

for Dresden Units 2 and 3

3.3 LIMITING CONDITION FOR OPERATION	4.3 SURVEILLANCE REQUIREMENT

3\. a. Control rod withdrawal sequences shall be established so that maximum reactivity that could be added by dropout of any increment of any one control blade would not make the core more than 0.013 ΔK supercritical.

b. Whenever the reactor is in the startup/hot standby or run mode below 10% rated thermal power, the rod worth minimizer shall be operable or a second licensed operator or other qualified technical station employee shall verify that the operator at the reactor console is following the control rod program.

4\. Control rods shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate equal to or greater than three counts per second.

5\. During operation with limiting control rod patterns, as determined by the nuclear engineer, either:

a. Both RBM channels shall be operable; or

b. Control rod withdrawal shall be blocked; or

c. The operating power level shall be limited so that the MCHFR will remain above 1.0 assuming a single error that results in complete withdrawal of any single operable control rod.

3\. The correctness of the control rod withdrawal sequence input to the RWM computer shall be verified after loading the sequence.

Prior to the start of control rod withdrawal towards criticality, the capability of the Rod Worth Minimizer to properly fulfill its function shall be verified by the following checks:

a. The RWM computer on line diagnostic test shall be successfully performed.

b. Proper annunciation of the selection error of one out-of-sequence control rod shall be verified.

c. The rod block function of the RWM shall be verified by withdrawing the first rod as an out-of-sequence control rod no more than to the block point.

4\. Prior to control rod withdrawal for startup or during refueling verify that at least two source range channels have an observed count rate of at least three counts per second.

5\. When a limiting control rod pattern exists, an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod (s) and daily thereafter.

indicative of a generic control rod drive problem and the reactor will be shutdown.

B. Control Rod Withdrawal

1. Control rod dropout accidents as discussed in the SAR can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod dropout accident is eliminated. The overtravel position feature provides a positive check as only uncoupled drives may reach this position. Neutron instrumentation response to rod movement provides a verification that the rod is following its drive. Absence of such response to drive movement would indicate an uncoupled condition.

2. The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage to the primary coolant system. The design basis is given in Section 6.6.1 of the SAR, and the design evaluation is given in Section 6.6.3. This support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing. Additionally, the support is not required if all control rods are fully inserted and if an adequate shutdown margin with one control rod withdrawn has been demonstrated since the reactor would remain subcritical even in the event of complete ejection of the strongest control rod.

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 0.013ΔK supercritical if they were to drop out of the core in the manner defined for the Rod Drop Accident. 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 supervised by the RWM or a second qualified station employe. This 0.013ΔK 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 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.

The analysis of the control rod drop accident was originally presented in Sections 7.9.3, 14.2.1.2 and 14.2.1.4 of the Safety Analysis Report. Improvements in analytical capability have allowed a more refined analysis of the control rod drop accident.

Bases (cont'd)

These techniques are described in a topical report[(1)] and two supplements.[(2)(3)]

By using the analytical models described in those 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 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 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 BWR's 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 K limit on in-sequence control rod or control rod segment worths. Details of this analysis are contained in Reference 4.

a. A maximum inter-assembly local power peaking factor not expected to be reached during future reloads.

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 minimum moderator temperature to reach criticality.

(4) Exhibit A attached to September 14, 1973 letter from Byron Lee, Commonwealth Edison Company, To J. F. O'Leary, U.S. Atomic Energy Commission.

Bases (cont'd)

In most cases the worth in in-sequence rods or rod segments will be substantially less than 0.013ΔK. Further, the addition of 0.013ΔK 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 0.013ΔK limit is applied in order to allow room for future reload changes and ease of verification without repetitive Technical Specification changes.

Should a control 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 an offsite dose well below the guideline value of 10CFR100. For 8 x 8 fuel, less than 850 rods are conservatively estimated to perforate with nearly the same consequences as for the 7 x 7 fuel case because of the rod power differences.

The Rod Worth Minimizer 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. Ref. Section 7.9 SAR. It serves as a backup to procedural control of control rod worth. In the event that the Rod Worth