XN-NF-78-44

A GENERIC ANALYSIS OF THE CONTROL ROD EJECTION TRANSIENT FOR PRESSURIZED WATER REACTORS

JANUARY 1979

RICHLAND, WA 99352

EXXON NUCLEAR COMPANY, Inc.

7902220679

XN-NF-78-44
ISSUE DATE: 02/06/79

A GENERIC ANALYSIS OF THE CONTROL ROD EJECTION TRANSIENT FOR PRESSURIZED WATER REACTORS

PREPARED BY:

R. J. BURNSIDE
T. L. KRYSINSKI
D. W. PRUITT

XN-NF-78-44
ISSUE DATE: 02/06/79

A GENERIC ANALYSIS OF THE CONTROL ROD EJECTION TRANSIENT FOR PRESSURIZED WATER REACTORS

PREPARED BY: RJ BURNSIDE
TL KRYSINSKI
DW PRUITT

APPROVED BY: JN Morgan 1-23-79
J. N. MORGAN, MANAGER
NEUTRONICS & FUEL MANAGEMENT

CE Leach 1/23/79
C. E. LEACH, MANAGER
THERMAL HYDRAULIC ENGINEERING

George Sofer 1-25-79
G. A. SOFER, MANAGER
NUCLEAR FUELS ENGINEERING

ACCEPTED BY: for 1-25-79
W. S. NECHODOM, MANAGER
LICENSING & COMPLIANCE

for
G. J. BUSSELMAN, MANAGER
CONTRACT PERFORMANCE

U. S. CUSTOMER DISCLAIMER

IMPORTANT NOTICE REGARDING CONTENTS AND USE OF THIS DOCUMENT

PLEASE READ CAREFULLY

Exxon Nuclear Company's warranties and representations concerning the subject matter of this document are those set forth in the Agreement between Exxon Nuclear Company, Inc. and the Customer pursuant to which this document is issued. Accordingly, except as otherwise expressly provided in such Agreement, neither Exxon Nuclear Company, Inc. nor any person acting on its behalf makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this document, or that the use of any information, apparatus, method or process disclosed any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method or process disclosed in this document.

The information contained herein is for the sole use of Customer.

In order to avoid impairment of rights of Exxon Nuclear Company, Inc. in patents or inventions which may be included in the information contained in this document, the recipient, by its acceptance of this document agrees not to publish or make public use (in the patent sense of the term) of such information until so authorized in writing by Exxon Nuclear Company, Inc. or until after six (6) months following termination or expiration of the aforesaid Agreement and any extension thereof, unless otherwise expressly provided in the Agreement. No rights or licenses in or to any patents are implied by the furnishing of this document.

Table of Contents

Section		Page
1.0	INTRODUCTION	1
1.1	DESCRIPTION OF ACCIDENT	1
1.2	DESIGN AND LIMITING CRITERIA	1
1.3	OBJECTIVE	2
2.0	SUMMARY	3
2.1	SUMMARY AND CONCLUSIONS	3
3.0	METHODS OF ANALYSIS	5
3.1	EJECTED ROD TRANSIENT COMPUTER MODEL	6
3.2	REACTIVITY FEEDBACK TREATMENT	7
3.3	PEAKING FACTORS AND FUEL TEMPERATURE TREATMENT	8
4.0	SELECTION OF PARAMETERS AND SENSITIVITY ANALYSIS	10
4.1	SELECTION OF PARAMETERS	10
4.2	SENSITIVITY ANALYSIS	10
4.2.1	Doppler Reactivity Feedback	11
4.2.2	Moderator Temperature Feedback	11
4.2.3	Reactivity Worth of Ejected Control Rod	11
4.2.4	Power Peaking Factors	12
4.2.5	Delayed Neutron Fraction	12
4.2.6	Mean Prompt Neutron Lifetime, ℓ^*	13
4.2.7	Ejected Control Rod Velocity	13
4.2.8	Reactor Trip Delay Time	13

Table of Contents Continued

Section	Page
4.2.9 Heat Transfer Coefficients	13
4.2.10 Initial Fuel Enthalpy	13
5.0 RESULTS	20
5.1 POWER LEVEL AND FUEL TEMPERATURE TRANSIENT RESULTS	20
6.0 OVERPRESSURIZATION ASSOCIATED WITH ROD EJECTION ACCIDENT	33
6.1 INTRODUCTION	33
6.2 DESCRIPTION OF MODEL	33
6.3 EXAMPLE CALCULATION	36
7.0 APPLICATION OF GENERIC ANALYSIS	40
7.1 NEUTRONIC DESIGN PARAMETERS	40
7.2 APPLICATION OF PARAMETRIC RESULTS	41
8.0 REFERENCES	43

LIST OF TABLES

Table		Page
6.1	EXAMPLE OVERPRESSURIZATION CALCULATION	38

LIST OF FIGURES

Figure		Page
4.1	CHANGE IN DEPOSITED ENTHALPY VS DOPPLER COEFFICIENT, HOT FULL POWER	14
4.2	CHANGE IN DEPOSITED ENTHALPY VS DOPPLER COEFFICIENT, HOT ZERO POWER	15
4.3	DEPOSITED ENTHALPY VS CONTROL ROD WORTH, HOT FULL POWER	16
4.4	DEPOSITED ENTHALPY VS CONTROL ROD WORTH, HOT ZERO POWER	17
4.5	CHANGES IN DEPOSITED ENTHALPY VS β_{eff}, HOT FULL POWER	18
4.6	CHANGES IN DEPOSITED ENTHALPY VS β_{eff}, HOT ZERO POWER	19
5.1	RELATIAVE POWER VS TANSIENT TIME, ROD WORTH, HFP	22
5.2	RELATIVE POWER VS TRANSIENT TIME. β_{eff}, HFP	23
5.3	RELATIVE POWER VS TRANSIENT TIME, ROD WORTH, HZP	24
5.4	RELATIVE POWER VS TRANSIENT TIME, β_{eff}, HZP	25

List of Figures Continued

Figure Page

5.5 CENTERLINE AND AVERAGE FUEL TEMPERATURE VS TIME, $\Delta\rho$ = .31%, β_{eff} = .0061, HFP 26

5.6 CENTERLINE AND AVERAGE FUEL TEMPERATURE VS TIME, $\Delta\rho$ = .58%, β_{eff} = .0061, HFP 27

5.7 CENTERLINE AND AVERAGE FUEL TEMPERATURE VS TIME, $\Delta\rho$ = .58%, β_{eff} = .0050, HFP 28

5.8 CENTERLINE AND AVERAGE FUEL TEMPERATURE VS TIME, $\Delta\rho$ = .89%, β_{eff} = .0061, HZP 29

5.9 CENTERLINE AND AVERAGE FUEL TEMPERATURE VS TIME, $\Delta\rho$ = 1.19%, β_{eff} = .0061, HZP 30

5.10 CENTERLINE AND AVERAGE FUEL TEMPERATURE VS TIME, $\Delta\rho$ = .89%, β_{eff} = .0050, HZP, P.F. = 12.8 31

5.11 CENTERLINE AND AVERAGE FUEL TEMPERATURE VS TIME, $\Delta\rho$ = .89%, β_{eff} = .0061, HZP, P.F. = 12.8 32

6.1 ν_2^* AS A FUNCTION OF PRESSURE 39

1.0 INTRODUCTION

Exxon Nuclear Company (ENC) has performed a generic control rod ejection accident analysis for thermal pressurized water reactors (PWR's). The control rod ejection accident was simulated for a reload type reactor core. The ejected control rod accident can be parameterized by the following variables: 1) reactivity worth of ejected control rod, 2) power peaking factor, 3) reactivity coefficients and 4) delayed neutron fraction, β_{eff}. With these variables defined, the core size, bank worth, etc., are not significant. Therefore, the ejected rod analysis presented here will be applicable to all future ENC reloads for PWR type reactors.

1.1 DESCRIPTION OF ACCIDENT

A control rod ejection accident is defined as the mechanical failure of a control rod mechanical pressure housing such that the coolant system pressure would eject a rodded control assembly (RCA) and drive shaft to a fully withdrawn position. The consequences of this mechanical failure is a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

The rod ejection accident is the most rapid reactivity insertion that can be reasonably postulated. The resultant core thermal power excursion is limited primarily by the Doppler reactivity effect of the increased fuel temperatures and is terminated by reactor trip of all remaining control rods, activated by neutron flux signals.

1.2 DESIGN AND LIMITING CRITERIA

Although the rod ejection accident is not expected to occur, design and limiting criteria are applied to insure that the power reactor system is sufficiently protected against this accident. These design and limiting criteria are:

1. The average fuel pellet enthalpy at the hot spot will be equal to or less than 280 cal/gm.
2. The peak reactor pressure during any portion of the transient will be less than the value that will cause stresses to exceed the emergency condition stress limits as defined in Section III of the ASME boiler and pressure vessel code.
3. Fuel melting will be limited to keep the off-site dose consequences well within the guidelines of 10 CFR Part 100, "Reactor Site Criteria".

These limiting criteria are taken from the NRC Regulatory Guide 1.77 "Assumptions used for evaluating a control rod ejection accident for pressurized water reactors".

1.3 OBJECTIVE

The objective of this study is to develop a parameteric set of curves, based on the criteria of Section 1.2, which quantify the consequences of the control rod ejection accident for combinations of significant parameters.

As the detailed cycle designs are completed for reactors reloaded by Exxon Nuclear Company, an analysis will be performed to demonstrate that the reactor system control rod ejection parameters limit the accident within the specified safety criteria of this generic report.

2.0 SUMMARY

2.1 SUMMARY AND CONCLUSIONS

This control rod ejection accident is a result of the assumed failure of a control rod mechanism pressure housing which ejects the control rod from the core. It is considered that this accident will not occur due to the low probability of a control rod housing failure.

The limiting criteria, given in Section 1.2, ensure that no long term reactor core cooling problems exist or that the radioactivity release limits according to 10 CFR 100, in the event the accident does occur, are not exceeded. The objective of this work is to demonstrate how the limiting criteria relate to the important neutronic design parameters to ensure the safety of the neutronic design of the reactor core.

A transient, two dimensional (R - Z geometry) computer model with fuel temperature feedback is utilized in this analysis. The model simulates the reactivity insertion caused by a control rod being ejected from the reactor core followed by the subsequent shutdown due to Doppler feedback and the scram bank entering the core. Prior to the start of the accident, the core initial conditions are set at a near critical state. The transient model computes the consequences for the accident in terms of the resultant peak energy (and temperature) deposition in the fuel. More details on the method are given in Section 3.0.

The result of this generic rod ejection analysis is presented as a set of curves for both hot full power (HFP) and hot zero power (HZP) conditions which allow a determination of the peak deposited enthalpy for the specific reload design parameters. This calculation will determine if the plant will meet design and limiting criteria given in Section 1.2. No attempt was made to determine the limiting value of each parameter. Rather the analysis as performed here, was to bound the parameters which impact on the rod ejection accident. Essentially the important parameters are: 1) reactivity worth of the ejected rod, 2) power peaking, 3) reactivity coefficients, and 4) delayed neutron fraction β_{eff}. Some other parameters and their effects are discussed in Section 4.0. Based on the current analysis the ejected rod accident is seen never to exceed the criteria set forth in Section 1.2 for expected values of the parameters affecting the rod ejection accident.

3.0 METHOD OF ANALYSIS

The limiting consequence due to a control rod ejection accident is calculated in terms of peak energy deposition in the fuel. Guideline values of stored energy content are set out by the NRC in Reference 1. Thus the objective of the control rod ejection analysis is to determine if any fuel will exceed these guideline values during the unlikely occurrence that a control rod is ejected.

The analysis and its results are applicable to all ENC PWR reload reactor cores since all important fuel assembly and core neutronic parameters used as input to the calculations were selected to envelope all current reload designs for which ENC has reload contracts. The sensitivity analysis discussed in Section 4.0 is to ensure this objective is met.

The general reactor core conditions assumed for this analysis are:

A - Hot full power

B - Hot zero power

Only the two power levels were calculated in this analysis. By analyzing hot full power and hot zero power conditions, the core parameters affecting the control rod ejection accident are bounded. Hence, the operation at other power levels between HFP and HZP will meet the criteria since that power level lies between the values already analyzed.

Beginning of cycle and end of cycle conditions are accounted for by the range of delayed neutron fractions utilized in the study.

The accident transient was assumed to last for five seconds, whereas the ejected control rod is completely out of the core in $\sim$0.1 seconds. All of the calculations herein reported have used a transient time of five seconds. The scram bank worth used in the model for hot full power is 2.62% $\Delta\rho$ and 3.48% $\Delta\rho$ at hot zero power. Both of these values are conservative when compared to the nominal scram bank worth available in PWR reload cores.

3.1 EJECTED ROD TRANSIENT COMPUTER MODEL

The XTRAN computer code (Reference 2) is utilized for the ejected rod accident analysis. The XTRAN code, specifically developed to analyze the ejected rod accident, is a two-dimensional (r - z cylindrical geometry) computer program which solves the space and time dependent neutron diffusion equation with fuel temperature and moderator density reactivity feedbacks. XTRAN employs a nodal method based directly on a one energy group finite difference technique for the solution of the time dependant neutron diffusion equation. The one-group macroscopic cross sections used in the iterative flux solution are collapsed from macroscopic two-group values modified at each time step by reactivity feedbacks.

The space and time dependent neutronic model incorporated in XTRAN is capable of computing a rapid reactor transient initiated by the reactivity insertion due to a control rod being removed from the core. Since the model utilizes two-dimensional (r-z) geometry, the code can calculate the rapidly changing flux distribution as a control rod travels out of the reactor core and the scram rod bank subsequently enters the reactor core.

XTRAN initially determines the static flux and power distribution corresponding to the problem input. This steady-state calculation includes heat transfer and determines the temperature distribution in the fuel rod and the peak center line temperature. The heat transfer coefficients are then set to zero for an adiabatic transient calculation and the initial time step for the transient analysis is 0.0001 seconds. The code then automatically determines the time step interval based on the number of iterations necessary to achieve convergence. This method permits small time steps during periods of slow change. Therefore, the code efficiently solves the transient problems without the user choosing time step sizes.

Six groups of delayed neutron precursors are employed in this transient analysis. The decay constants and delayed neutron fractions utilized in the generic rod ejection analysis are typical of those calculated during normal PWR reload design efforts.

XTRAN has been evaluated and the results compared to other transient models, and has shown good agreement. Details of these comparisons are given in Refrrence 2.

3.2 REACTIVITY FEEDBACK TREATMENT

The XTRAN code has the ability to model both moderator and Doppler feedback effects. In this study, the moderator feedbacks are conservatively set equal to zero and the transient performed adiabatically. Due to the rapid power excerions, typical of the control rod ejection transient, tne scarm banks are tripped and enter the core before signficant perturbations occur in the moderator temperature. Therefore all the analysis completed in this report have no moderator feedbacks included.

Although the XTRAN model is two-dimensional, (r-z geometry) there is a radial and axial component to its calculation. Due to this type of calculation no special weighting is performed for the Doppler feedbacks. The Doppler feedback is modelled by inputting the change in the macroscopic cross sections due to the change in fuel temperature. The effect of a change in fuel temperature upon the cross sections is modelled in terms of the square root of the two temperatures, where one is the reference fuel temperature of the cross sections. The modelling of the Doppler effect in this manner shows that for a change in temperature. the change in the cross section is constant.

3.3 PEAKING FACTORS AND FUEL TEMPERATURE TREATMENT

The XTRAN model calculates the peaking factors at each node where a node is defined by the radial and axial mesh spacing of the geometry. This allows the simulation of the power peaking in the reactor core.

The power peaking factors parameterized in this analysis are calculated by XTRAN at the time when the ejected control rod has just moved out of the core. This transient peaking factor reflects some amount of Doppler feedback in its calculation. The neutronic calculation of the ejected rod parameters for a reload core is performed statically, that is, with no pointwise feedbacks. Hence, the neutronic design calculation will yield a conserative evaluation of the power peaking.

The fuel temperature is calculated for each radial mesh interval as if there were a single fuel rod in that radial location. The fuel rod is divided into eight equal volume nodes plus one cladding node. The axial direction is explicitly defined. Temperatures are calculated for each of the nine fuel rod nodes at each time interval based on the specific heat data for UO_2 of R. A. Hein and P. N. Flogell (Reference 3). The modelling details of this procedure are also described in Reference 2.

4.0 SELECTION OF PARAMETERS AND SENSITIVITY ANALYSIS

4.1 SELECTION OF PARAMETERS

Exxon Nuclear Company (ENC) has been contracted to reload a variety of pressurized water reactors. This generic rod ejection analysis report should cover all these reactor types. After reviewing the reactor types for which ENC has responsibility, D. C. Cook Unit 1 was chosen as the representative plant. This plant has a high power density (about 100 kw/ft). All parameters input in the analysis are typical D. C. Cook values. The parametric analysis then extends these D. C. Cook typical values to cover the range of values the parameters may have for other specific plants.

Uncertainties were not explicity applied to any of the values in the ejected control rod analysis. The neutronic parameter conservatism is accounted for in uncertainties applied to the peaking factors and ejected rod worths as the design calculations are completed for each specific plant. Also for this ejected rod analysis, the thermal heat transfer parameter uncertainties are not vital since the calculations were completed with no heat transfer from the fuel. This procedure is conservative with respect to the calculation of the deposited enthalpy in the core.

4.2 SENSITIVITY ANALYSIS

This section describes the results of varying the important parameters to show their sensitivity as well as enable the future fuel management schemes for ENC plants to be covered by this analysis. This sensitivity study comprehensively parameterizes all the important parameters to the ejected rod analysis.

4.2.1 Doppler Reactivity Feedback

The Doppler feedback effect on the control rod ejection accident is shown in Figures 4.1 and 4.2 for hot full power and hot zero power, respectively. The Doppler feedback has a larger effect at HZP than HFP. The Doppler reactivity coefficients used in the calculations are covered by the range of .8 to 1.35 pcm/^{o}F which is conservative with respect to nominal design values of about 1.7 pcm/^{o}F.

The Doppler feedback is more effective at hot zero power temperatures since the fuel will rise in temperature more for a given enthalpy increase than at hot full power temperatures. This can be observed from the heat capacity curve for UO_2 which in the hot zero power range of temperatures is relatively flat. For the full power temperature range (hot spot is >2500^{o}F) the heat capacity is initially larger than at HZP and rises rapidly with increasing temperature so the temperature change is smaller for a given enthalpy increase.

4.2.2 Moderator Temperature Feedback

No parameteric analysis was performed, since all of the calculations excluded moderator feedback.

4.2.3 Reactivity Worth of Ejected Control Rod

Figures 4.3 and 4.4 show the variation of the deposited enthalpy with the ejected rod worth for HFP and HZP conditions, respectively. As expected the magnitude of the accident increases with increasing rod worth. The reactivity worth of the ejected rod at HFP is smaller than at HZP due to the constraints imposed on the plant by the control rod insertion limits.

4.2.4 Power Peaking Factors

Also presented in Figures 4.3 and 4.4 are the effects of power peaking factors on deposited enthalpy for HFP and HZP, respectively. The magnitude of the accident increases with increasing peaking factors. As seen in the HZP case, there is an ejected rod worth below which the peaking factor has no effect on the accident. This is due to the fact that the lower rod worth reactivity insertion does not initiate a high power transient.

4.2.5 Delayed Neutron Fraction

The effective delayed neutron fraction, β_{eff}, effect on deposited enthalpy (and hence fuel center line temperature) is shown in Figures 4.5 and 4.6 for HFP and HZP, respectively. The HFP transients are less sensitive to β_{eff} since a smaller delayed neutron fraction results in a faster power reduction after the trip. Since a larger percentage of the energy deposition occurs after the trip for transients at HFP than at HZP a larger benefit is realized for the faster power reduction at HFP. This benefit partially compensates for the larger reactivity insertion, expressed in dollars, and results in a reduced sensitivity to β_{eff} at HFP.

Of all the parameters effecting the rod ejection accident only the moderator temperature coefficient (MTC), exposure distribution and the delayed neutron fraction significantly change from beginning of cycle to end of cycle. Since the analysis described herein has set the MTC equal to zero, and no credit has been taken for the flattening effect of the EOC exposure distribution on the core power distribution, only the β_{eff} changes from BOC and EOC. Therefore, this subsection describing the effect of changing β_{eff} on the rod ejection accident, also accounts for changes due to cycle burnup based upon the above assumption.

4.2.6 Mean Promp Neutron Lifetime ℓ^*

The deposited enthalpy was found to be independent of ℓ^* in the 10 to 15 μ/sec range.

4.2.7 Ejected Control Rod Velocity

The deposited enthalpy was found to be independent of the time for the ejected rod to leave the core. For a 20% increase to the ejected rod velocity (increase the reactivity insertion rate) there was no change in the deposited enthalpy.

4.2.8 Reactor Trip Delay Time

The deposited enthalpy was found to be independent of the reactor trip delay time in the .5 to .6 second range.

4.2.9 Heat Transfer Coefficients

No sensitivity studies were completed here since the transient analysis was performed with no heat transfer from the fuel.

4.2.10 Initial Fuel Enthalpy

The transient incremental deposited enthalpy was found to be insensitive to the initial fuel enthalpy at HFP and HZP. Therefore, any changes in the initial fuel enthalpy due to power redistribution or heat conduction parameters during the steady state can be applied as a bias to the total deposited enthalpy. The initial peak fuel enthalpies in the rod ejection region for this study at HFP and HZP were 40.8 and 16.7 cal/gm, respectively.

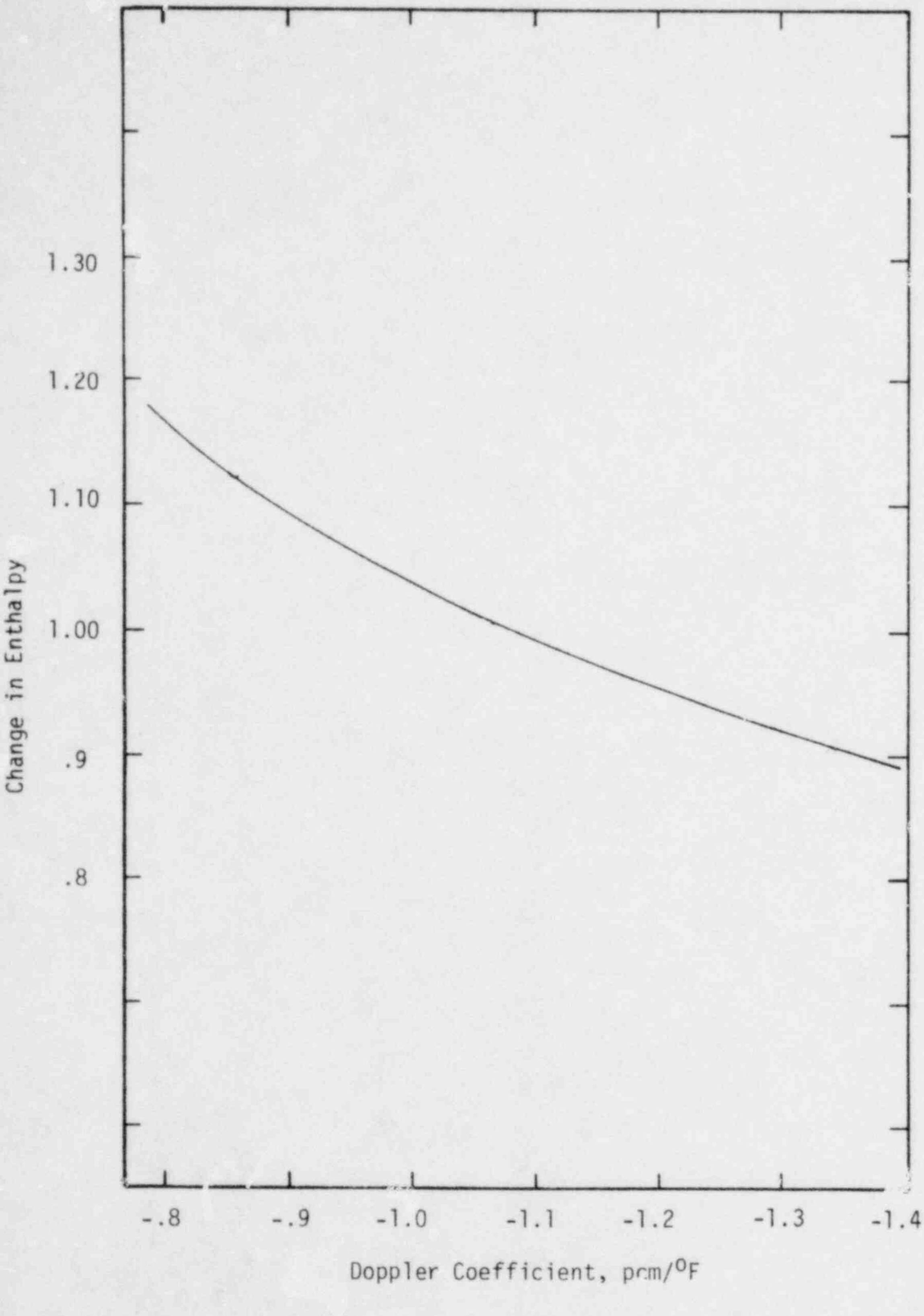

Figure 4.1 Change in Deposited Enthalpy vs Doppler Coefficent
Hot Full Power

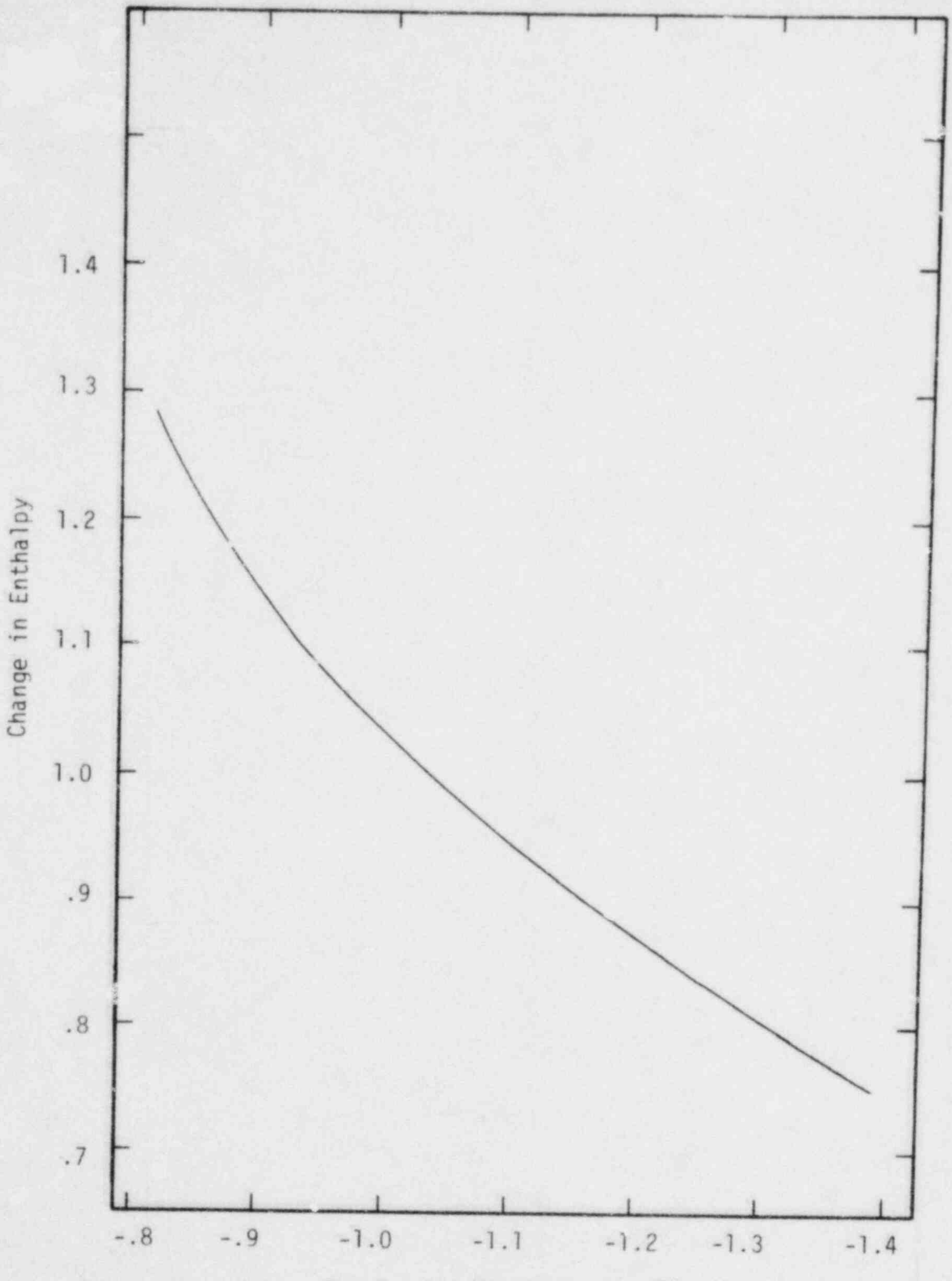

Figure 4.2 Change in Deposited Enthalpy vs Doppler Coefficient Hot Zero Power

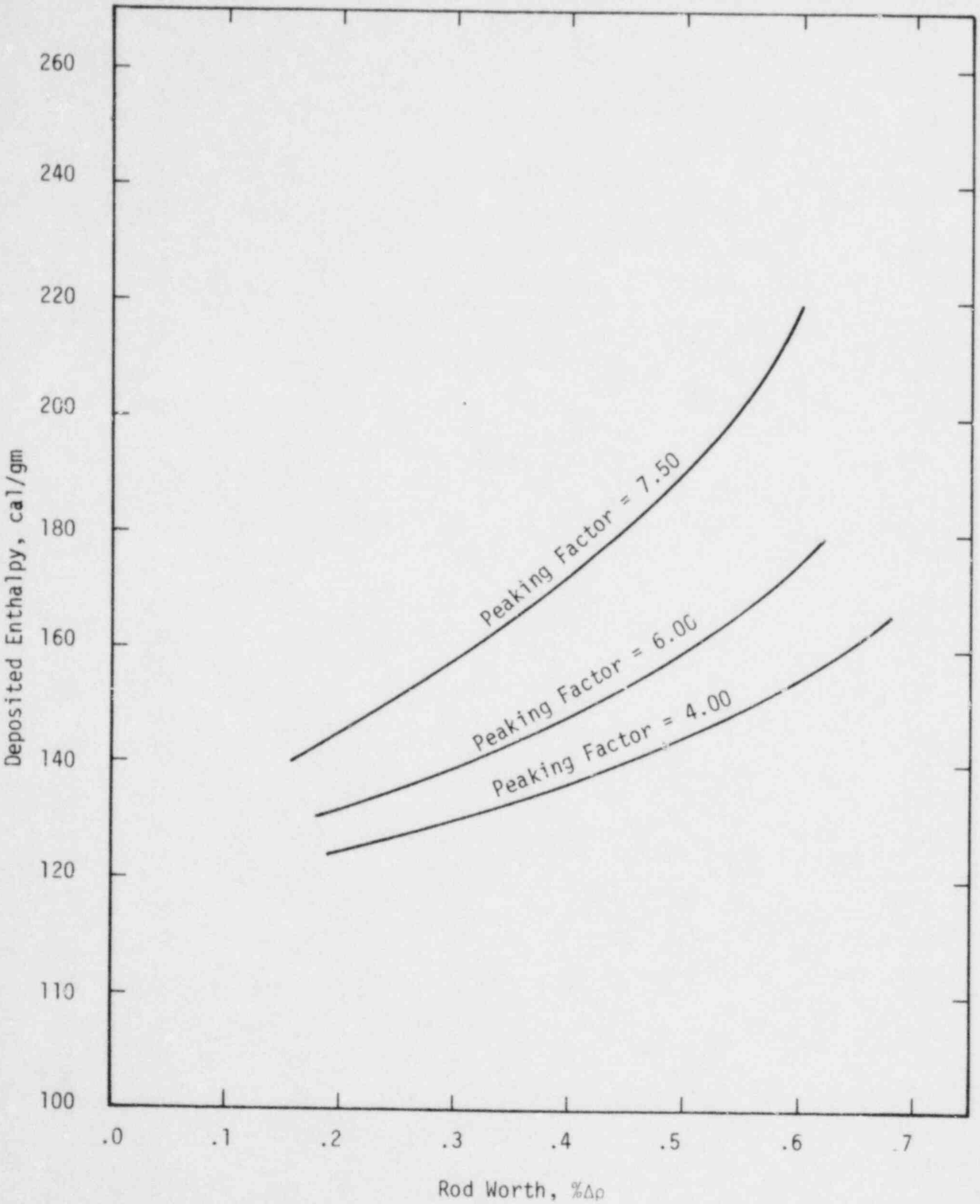

Figure 4.3 Deposited Enthalpy vs Control Rod Worth
Hot Full Power

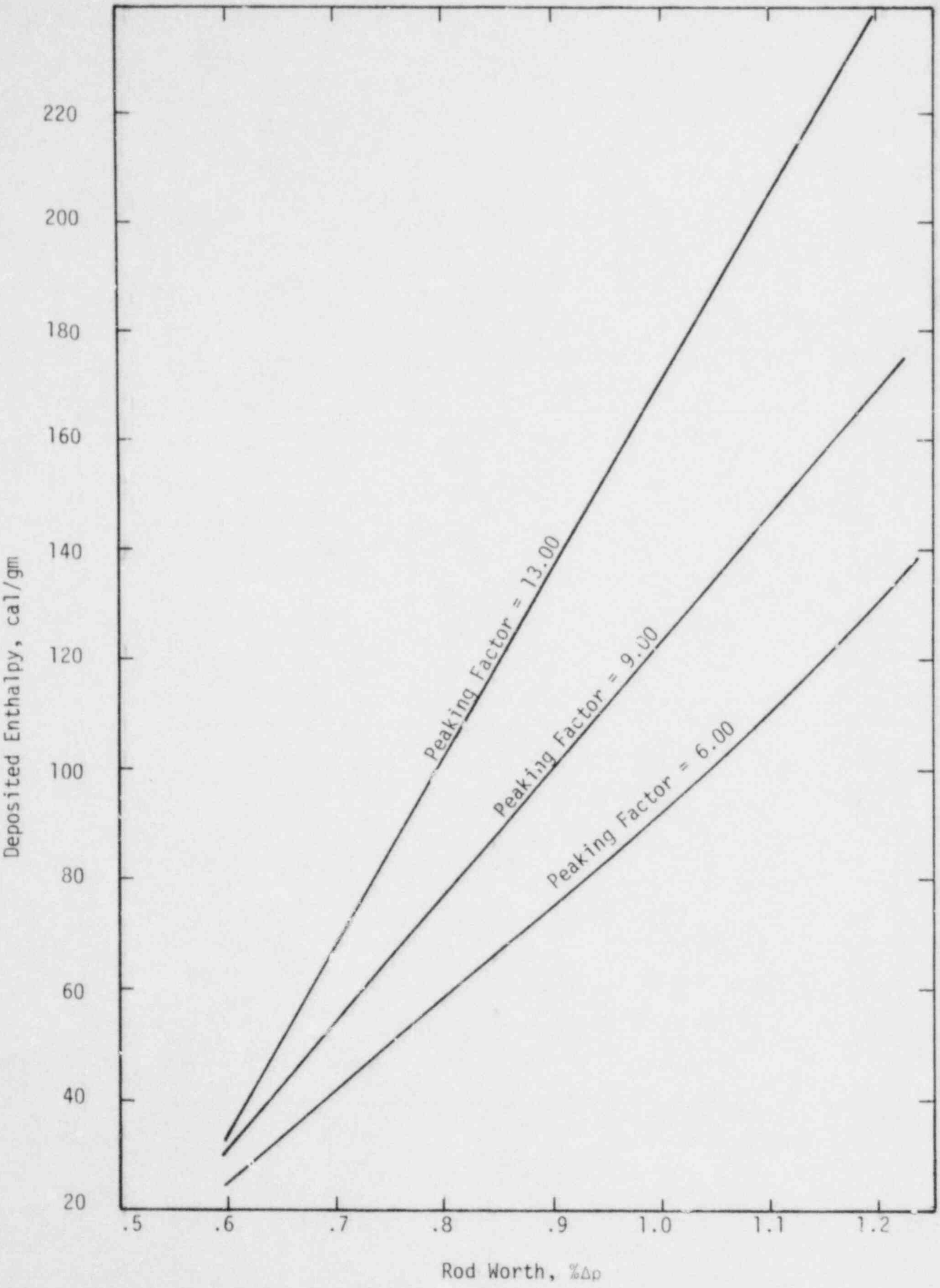

Figure 4.4 Deposited Enthalpy vs Control Rod Worth
Hot Zero Power

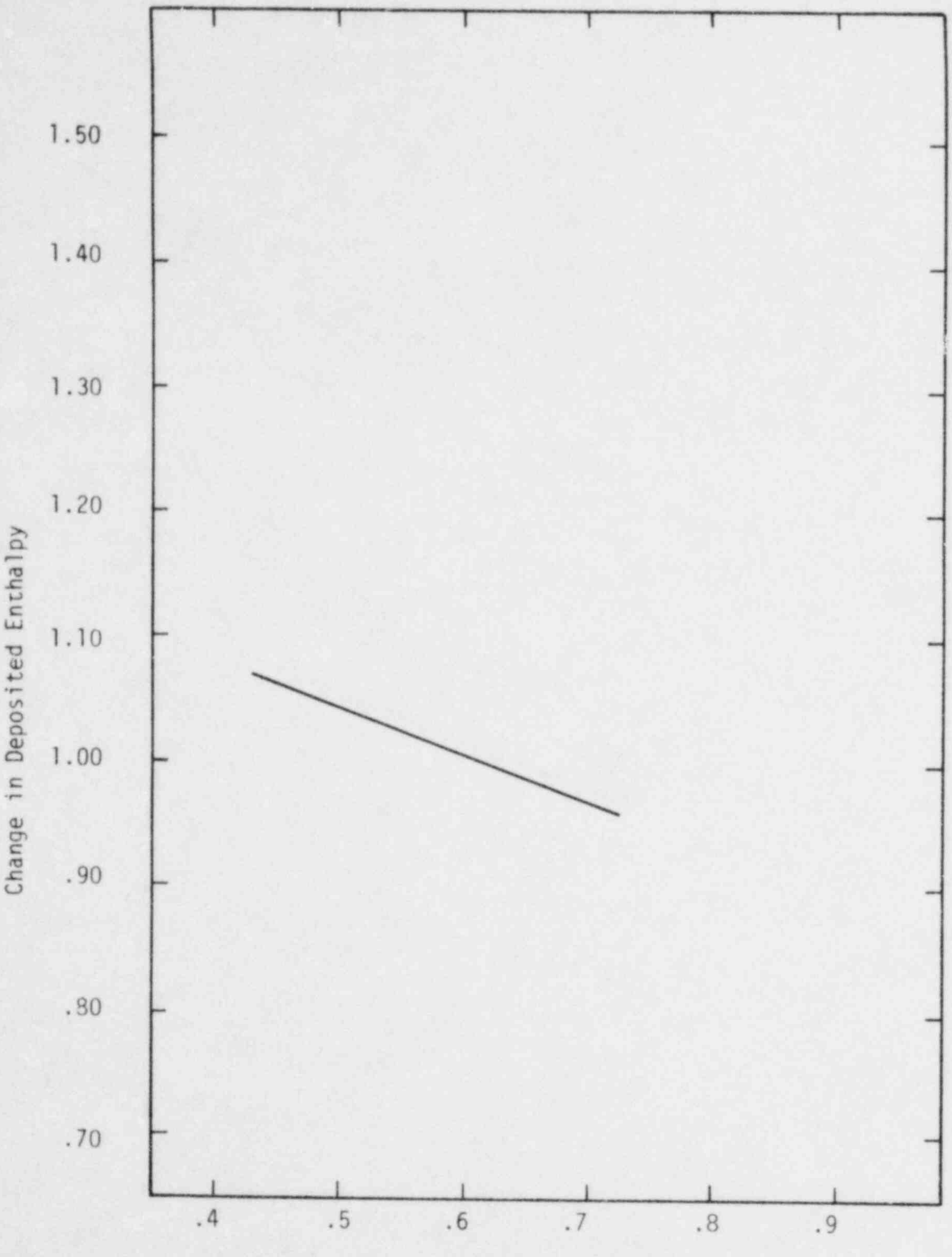

Figure 4.5 Changes in Deposited Enthalpy vs β_{eff}
Hot Full Power

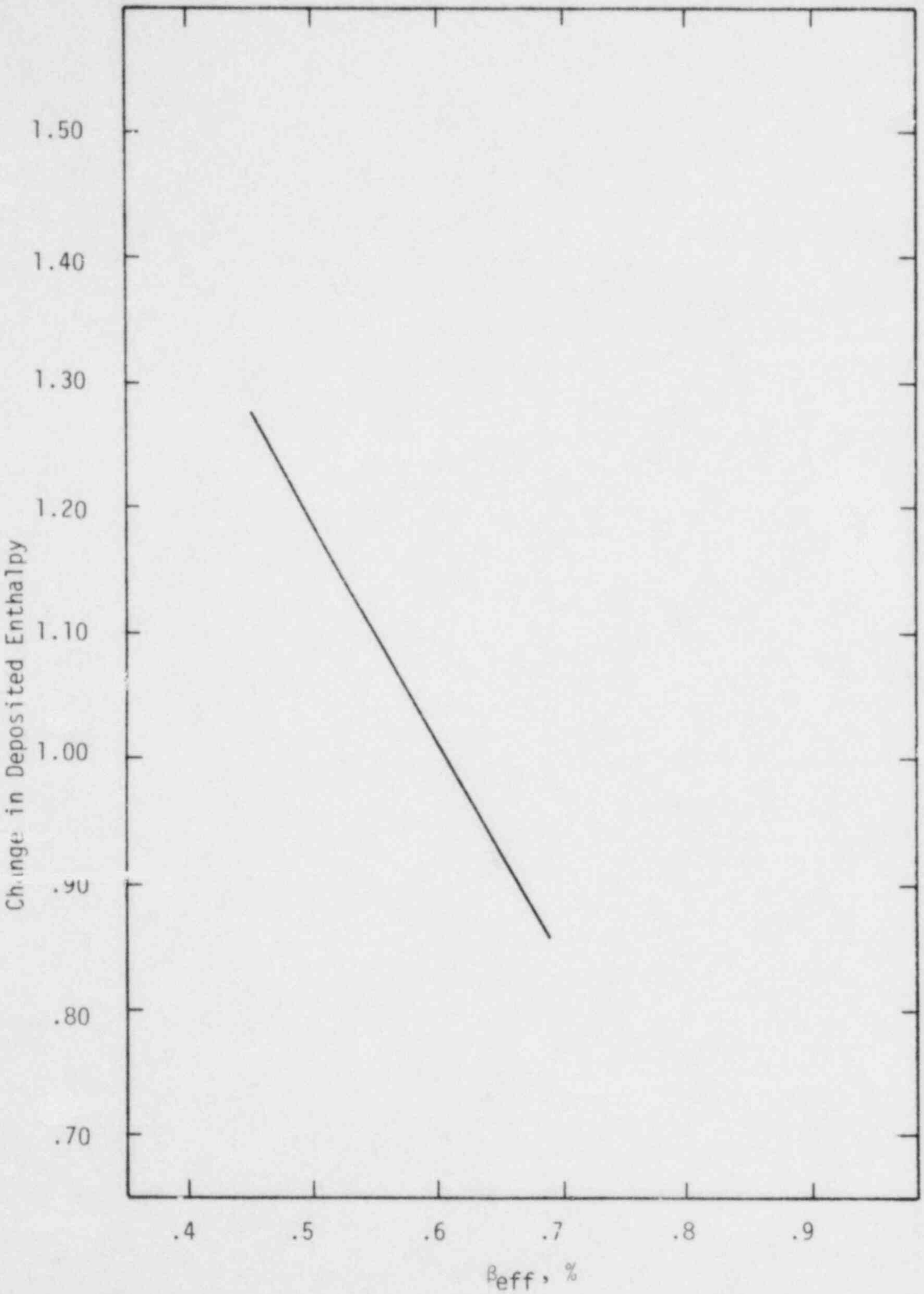

Figure 4.6 Changes in Deposited Enthalpy vs β_{eff}
Hot Zero Power

5.0 RESULTS

5.1 POWER LEVEL AND FUEL TEMPERATURE TRANSIENT RESULTS

Although the results of the calculations to determine the deposited enthalpy for the rod ejection accident are described in Section 4.0, no discussion has been made of the power level and fuel temperature transients.

The nuclear power transient calculation with no thermal heat transfer from the fuel was calculated with XTRAN. Figure 5.1 shows the nuclear power transient for rod worths of .58% $\Delta\rho$ and .31% $\Delta\rho$ at hot full power for the first 4.0 seconds of the ejected rod accident. The Doppler coefficient is -1.065 pcm/^{o}F and β_{eff} is .0061. Figure 5.2 shows the nuclear power transient for the BOC case (β_{eff} = .0061) and the EOC case (β_{eff} = .0050) for the .58% $\Delta\rho$ transient.

Figure 5.3 shows the nuclear power transient for the hot zero power case at rod worths of 1.191% $\Delta\rho$ and .890% $\Delta\rho$. The peaking factor here is 5.65, the Doppler coefficient is -1.027 pcm/^{o}F, and β_{eff} = .0061. Figure 5.4 shows the BOC (β_{eff} = .0061) and EOC (β_{eff} = .0050) nuclear transient for the $\Delta\rho$ insertion of .89%. For these cases the peaking factor is 12.80.

From the same calculations as discussed above, Figures 5.5., 5.6, and 5.7 show the peak fuel temperatures and the average fuel temperatures for the hot full power cases. The hot zero power fuel temperature transients corresponding to Figures 5.3 and 5.4 are shown in Figures 5.8 through 5.11. These XTRAN results are from the same calculations which generated the nuclear power transient data. Notice that for the cases shown the fuel temperature is always below 4400^{o}F. The peak fuel temperature in the cases shown, was 4333^{o}F corresponding to the case with $\Delta\rho$ = .58%, β_{eff} .0050 and a peaking factor of 5.65. This is only 69^{o}F higher than the identical case with β_{eff} equal to .0061.

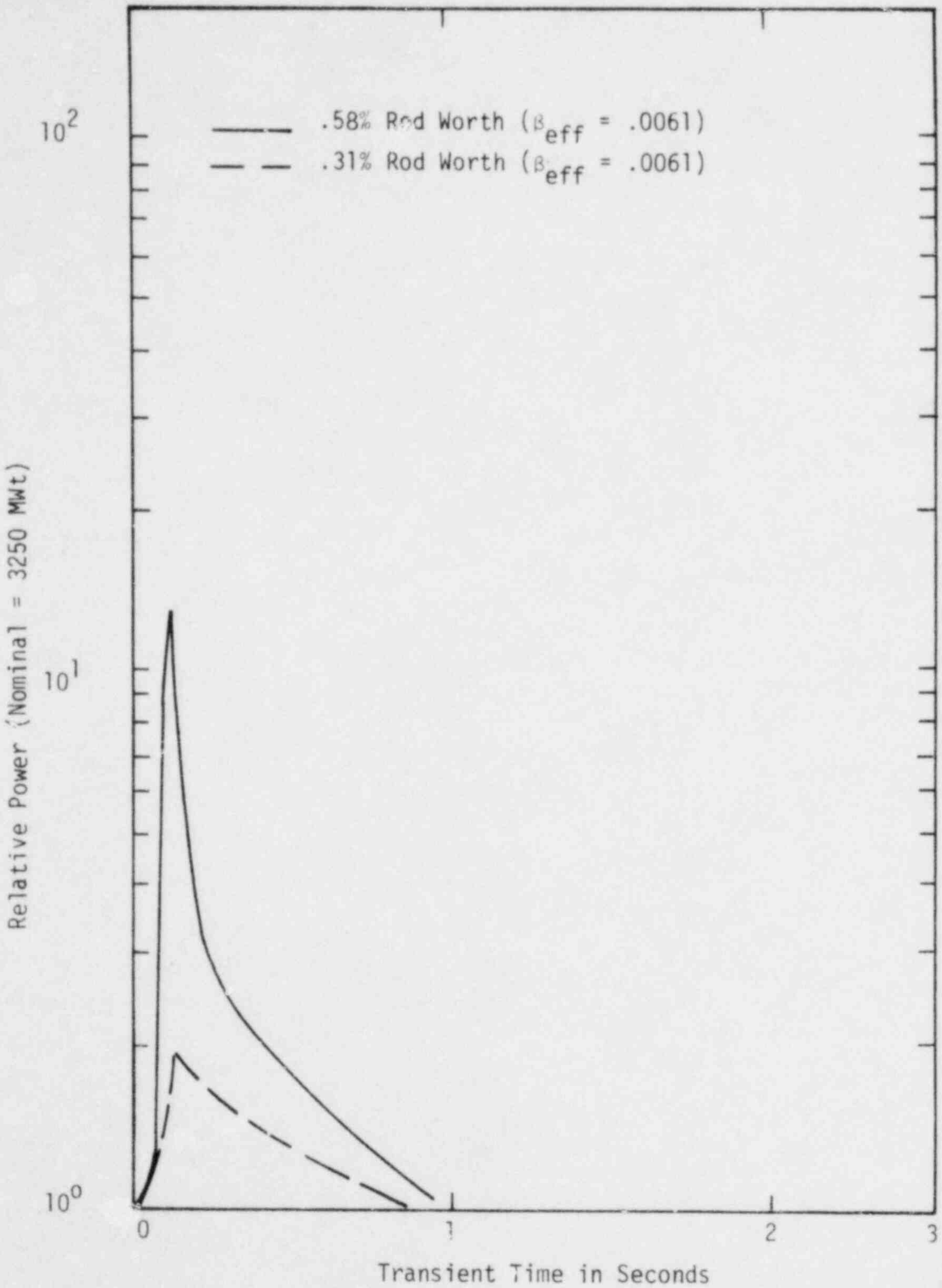

Figure 5.1 Relative Power vs Transient Time
Hot Full Power

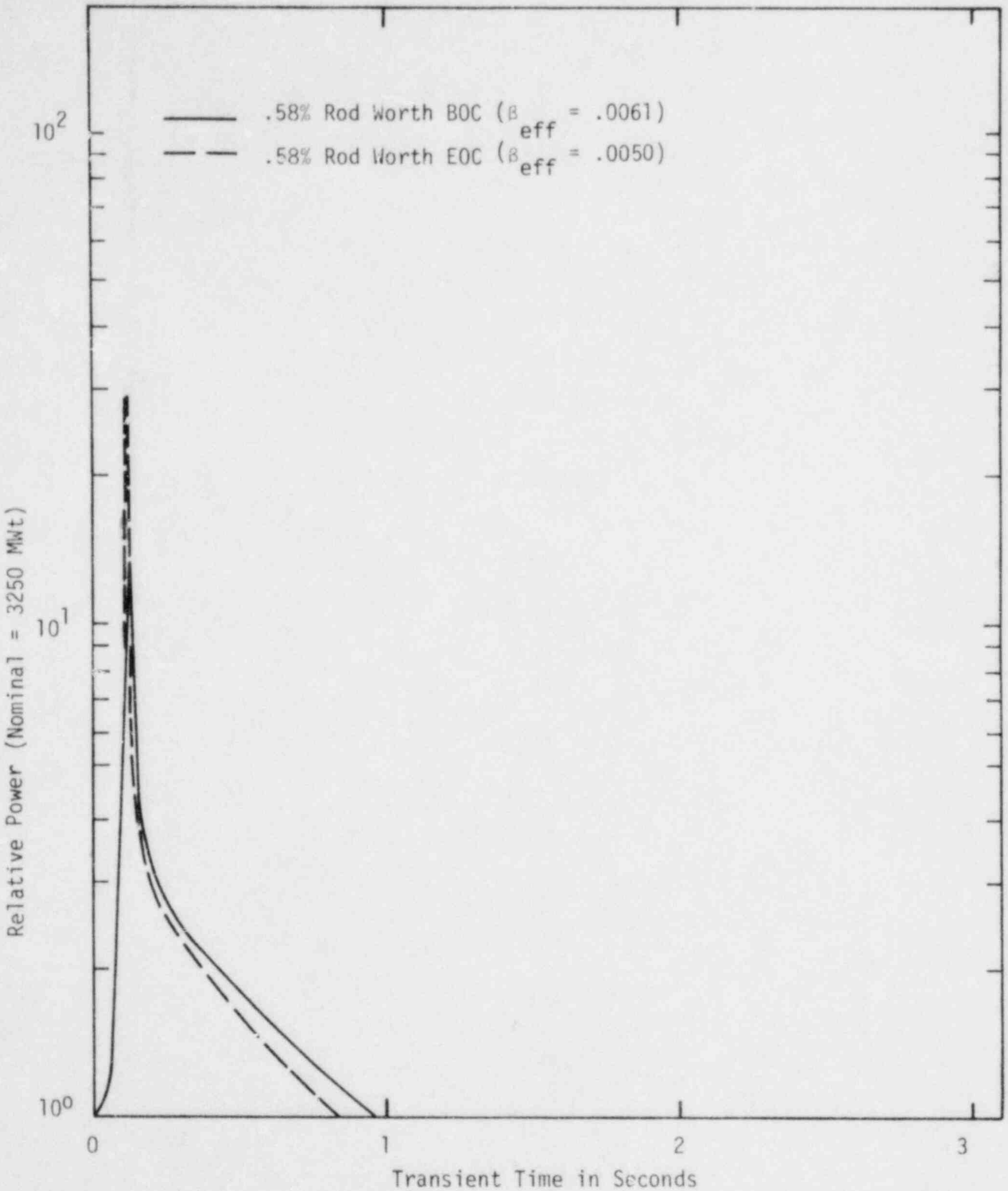

Figure 5.2 Relative Power vs Transient Time
Hot Full Power

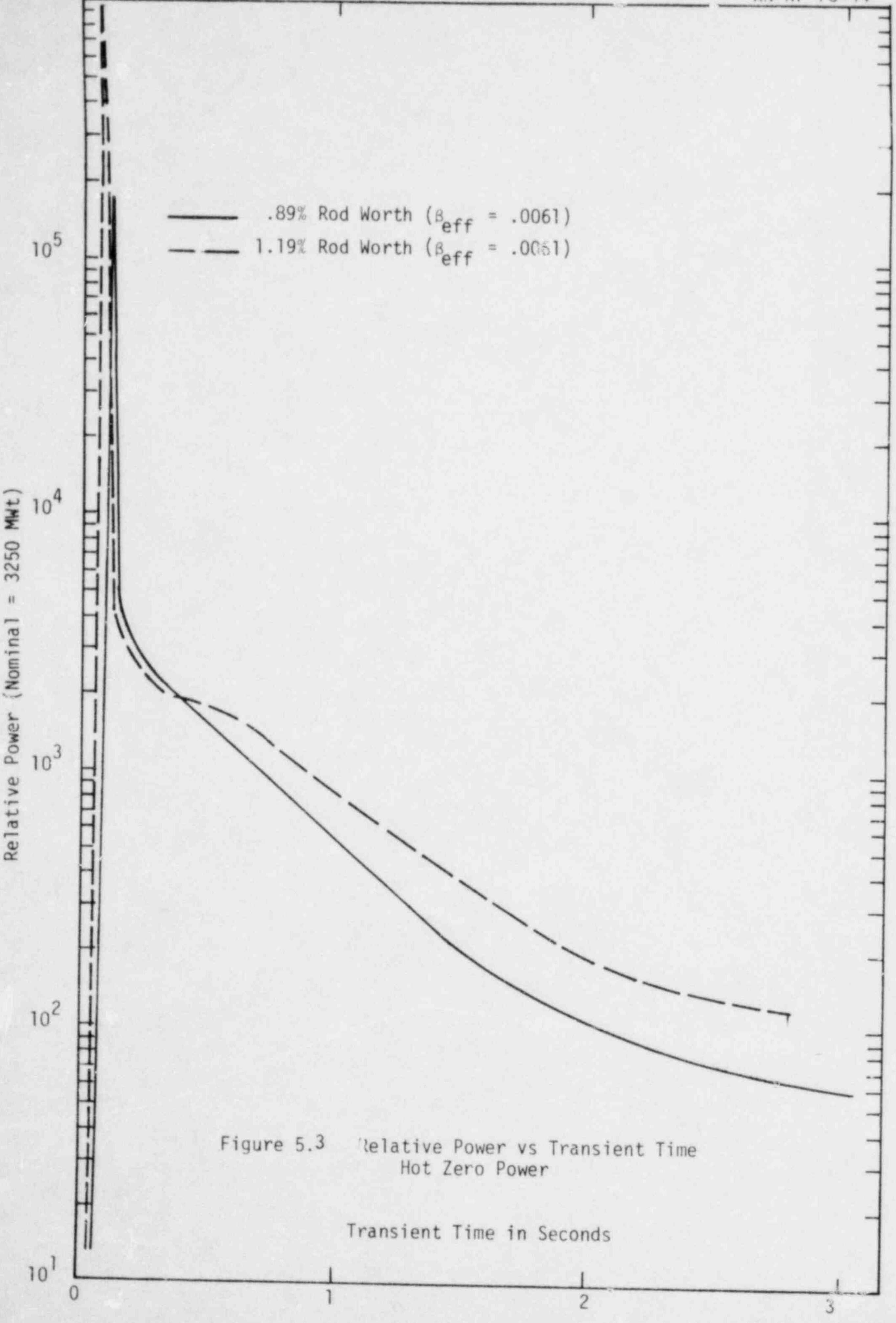

Figure 5.3 Relative Power vs Transient Time
Hot Zero Power

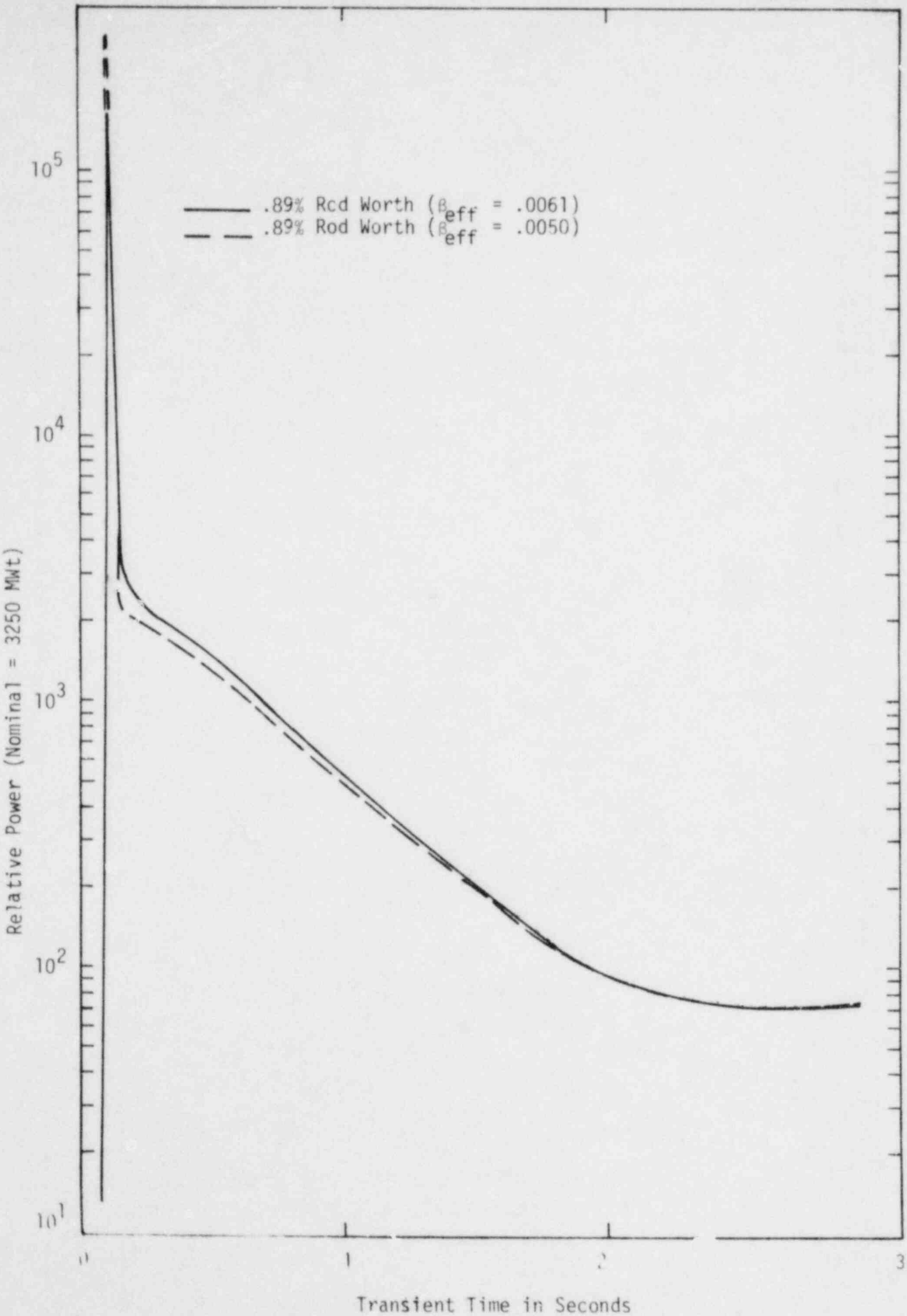

Figure 5.4 Relative Power vs Transient Time
Hot Zero Power

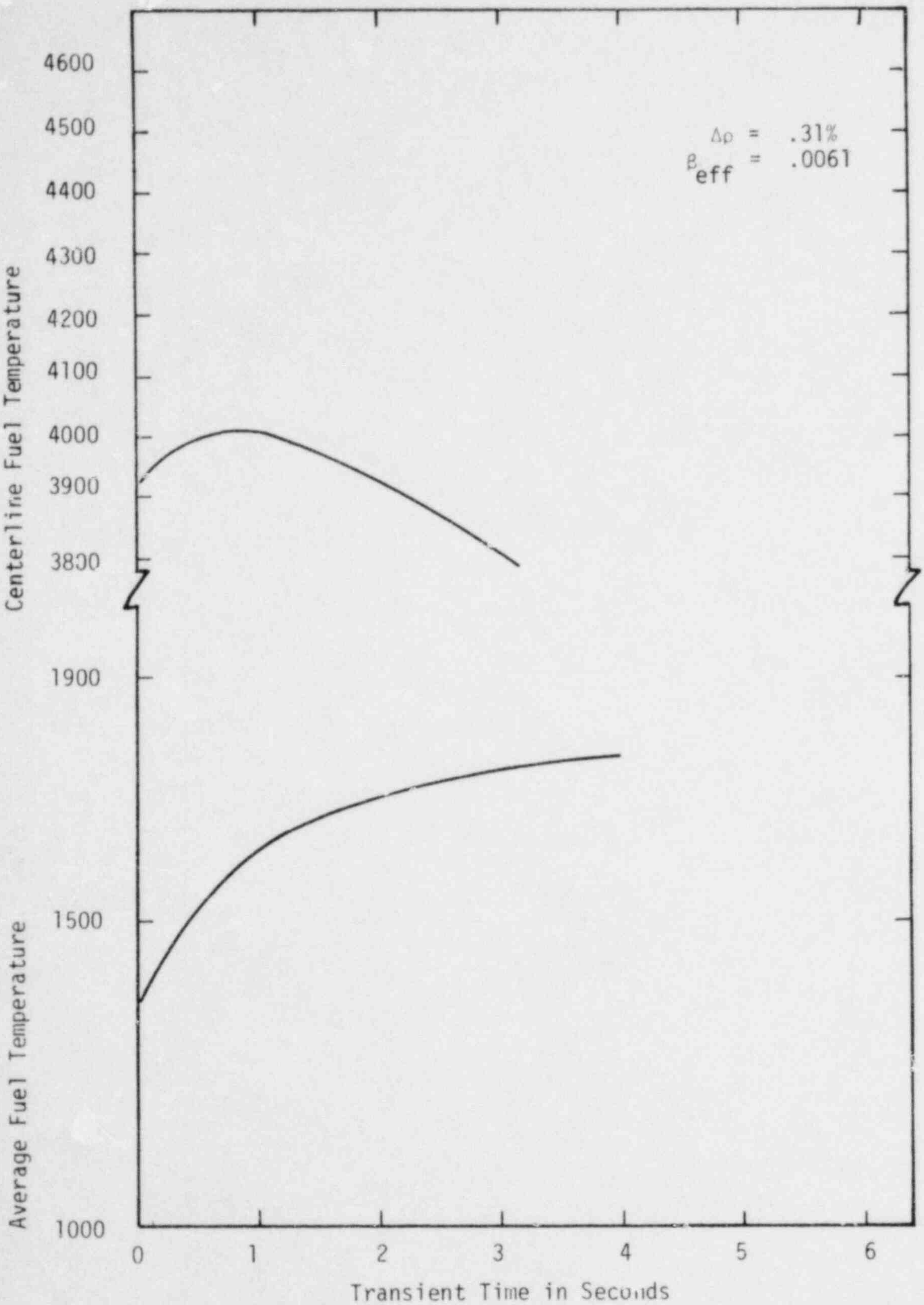

Figure 5.5 Centerline and Average Fuel Temperature vs Transient Time, Hot Full Power

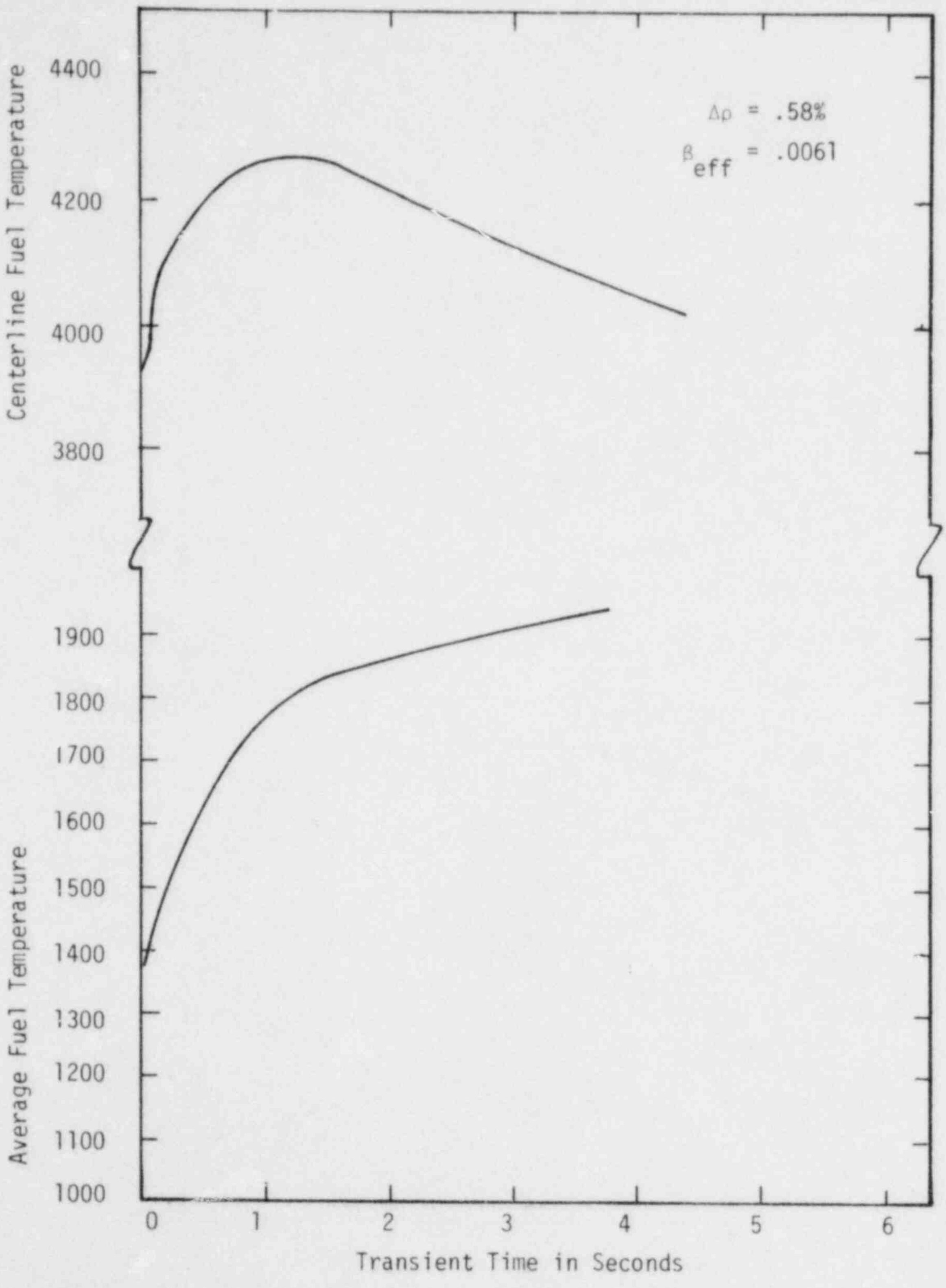

Figure 5.6 Centerline and Average Fuel Temperature vs Transient Time, Hot Full Power

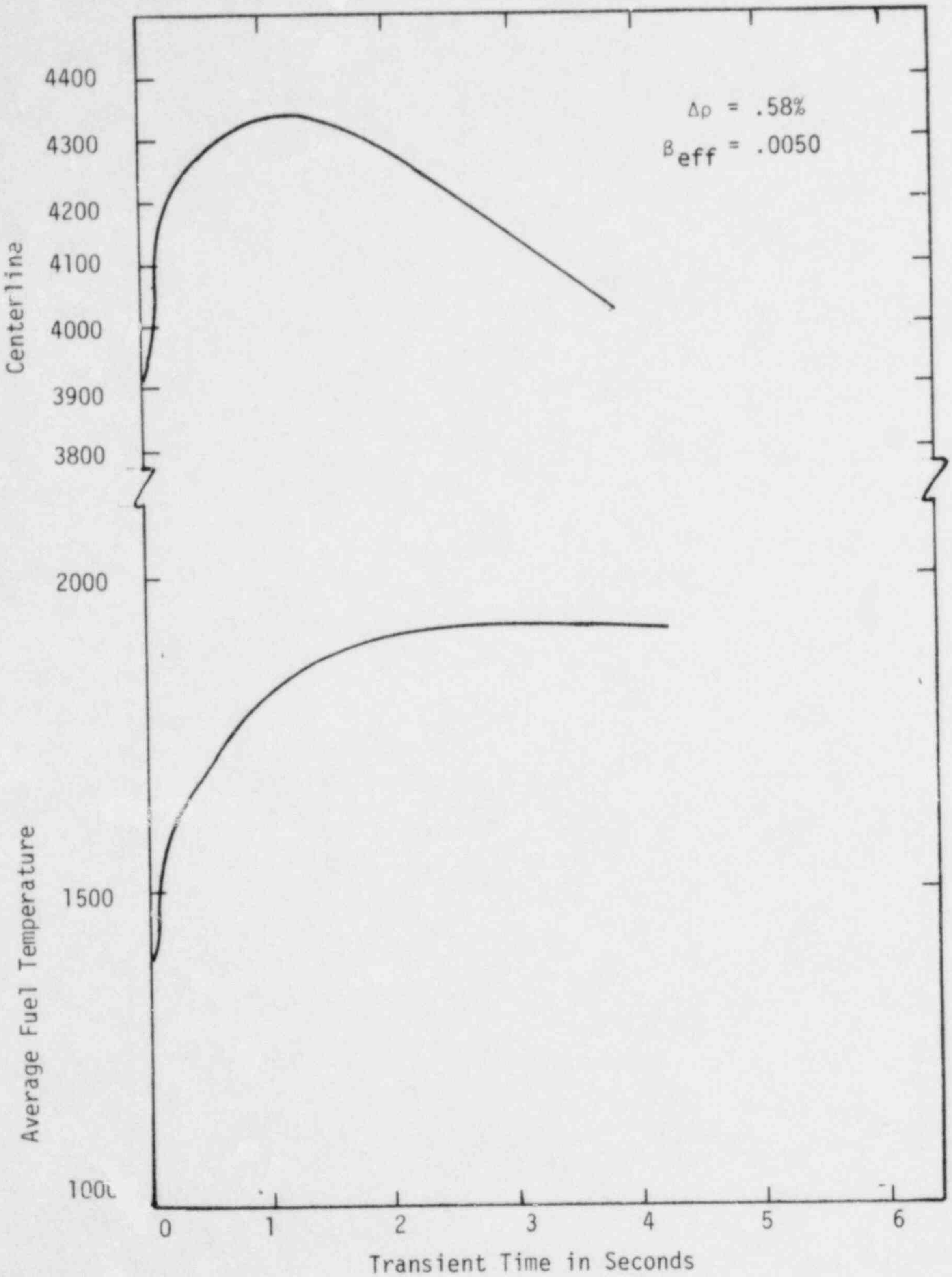

Figure 5.7 Centerline and Average Fuel Temperature vs Transient Time, Hot Full Power

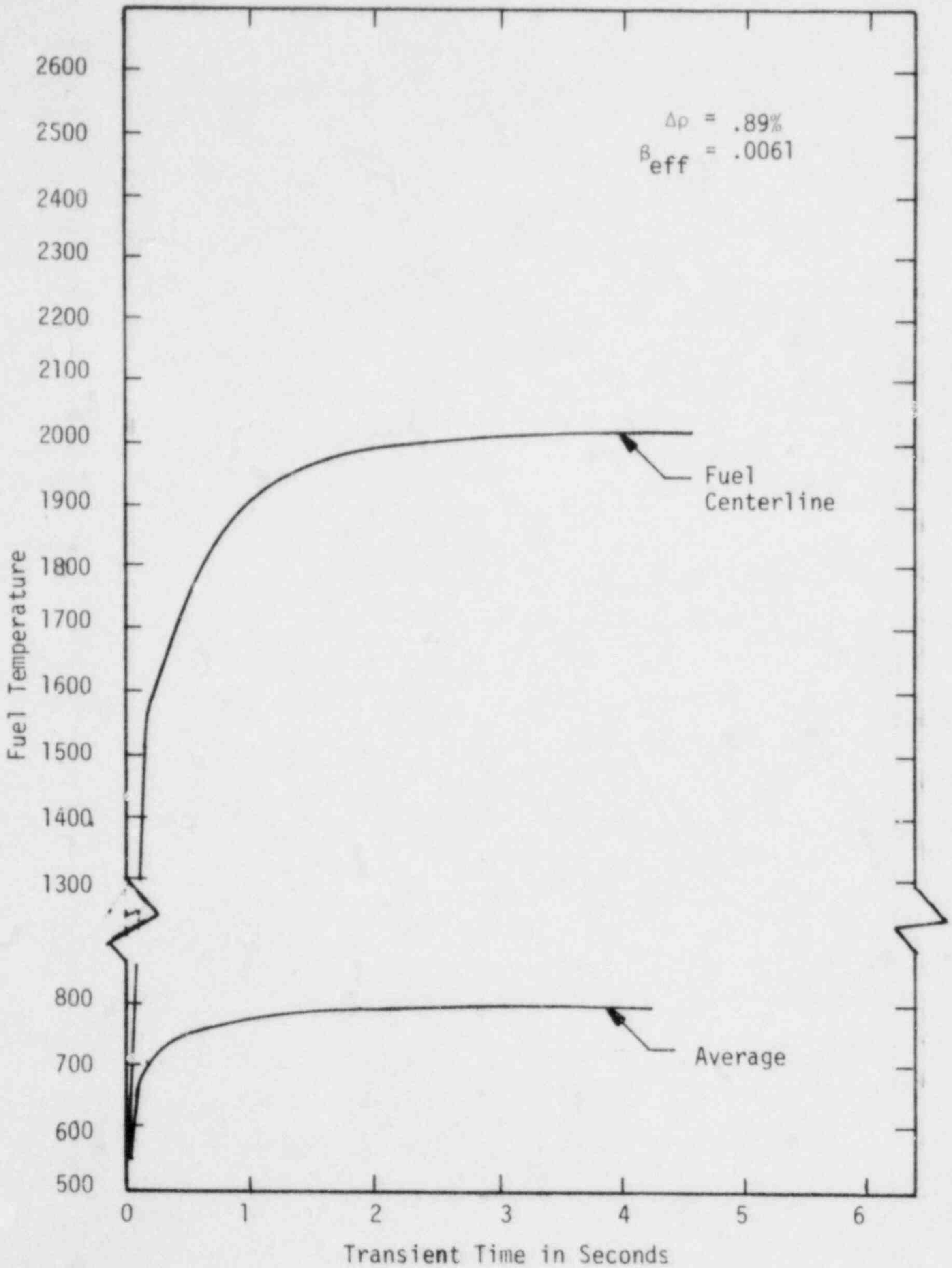

Figure 5.8 Centerline and Average Fuel Temperature vs Transient Time, Hot Zero Power

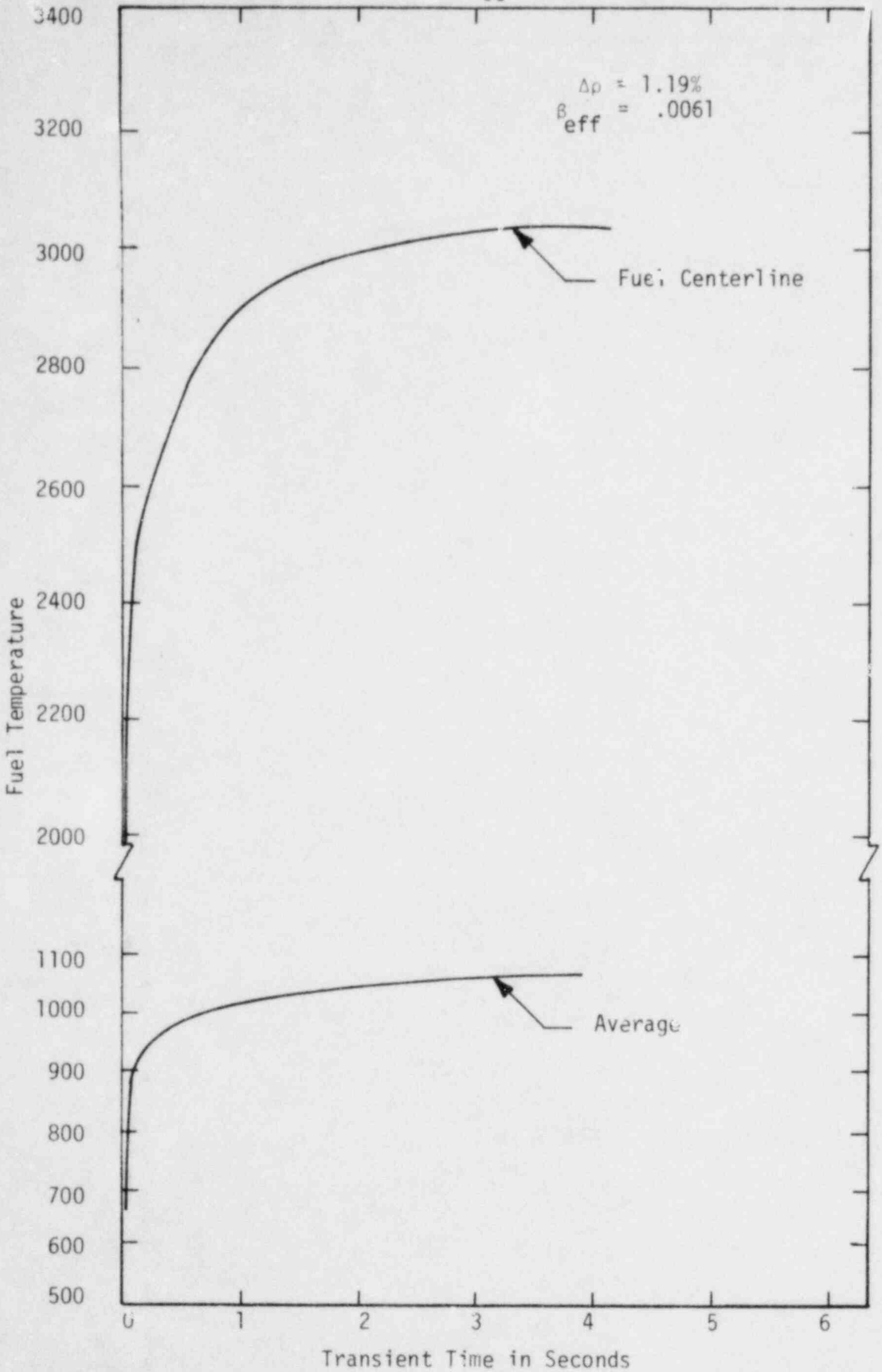

Figure 5.9 Centerline and Average Fuel Temperature vs Transient Time, Hot Zero Power

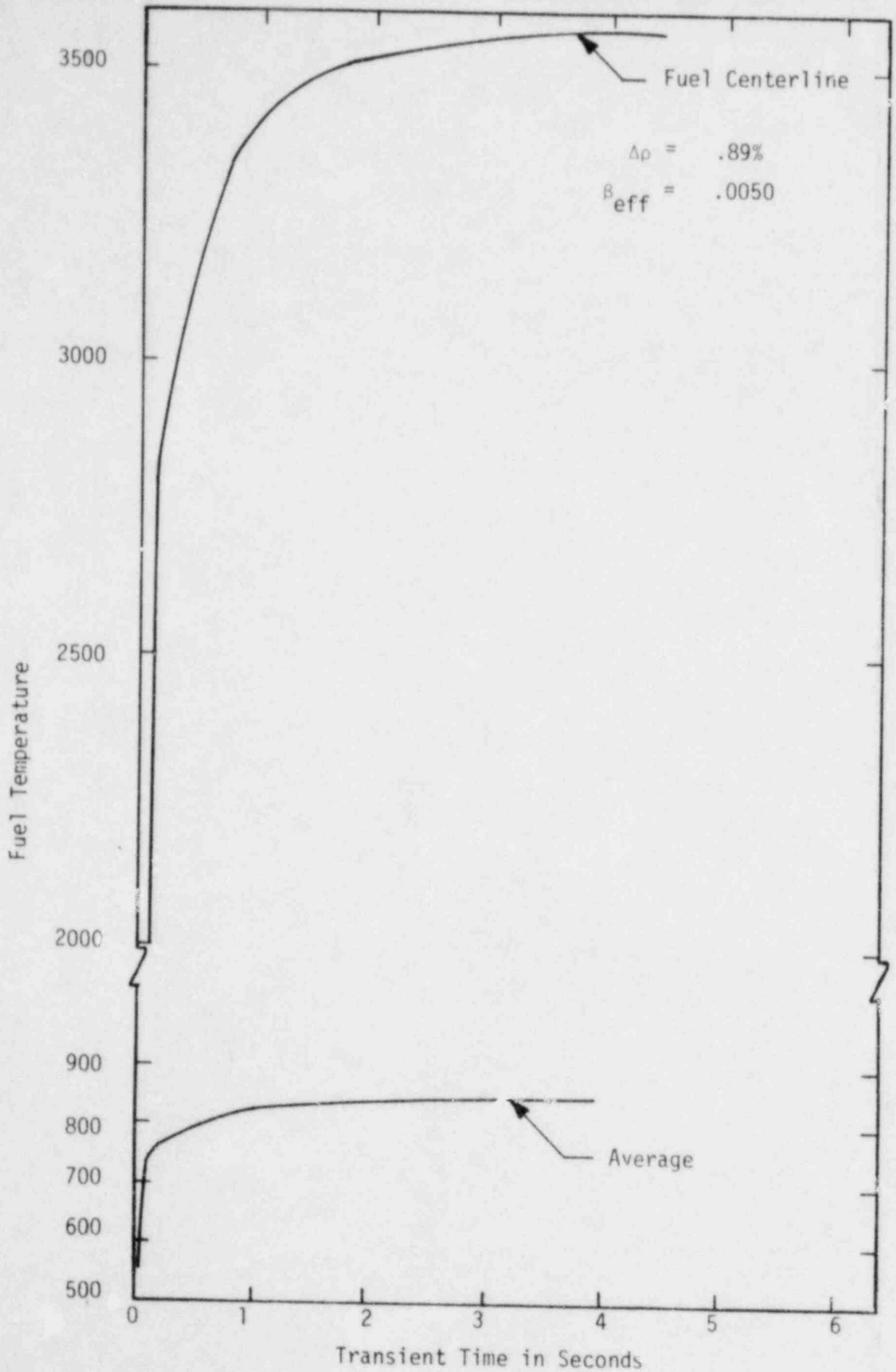

Figure 5.10 Centerline and Average Fuel Temperature vs Transient Time, Hot Zero Power

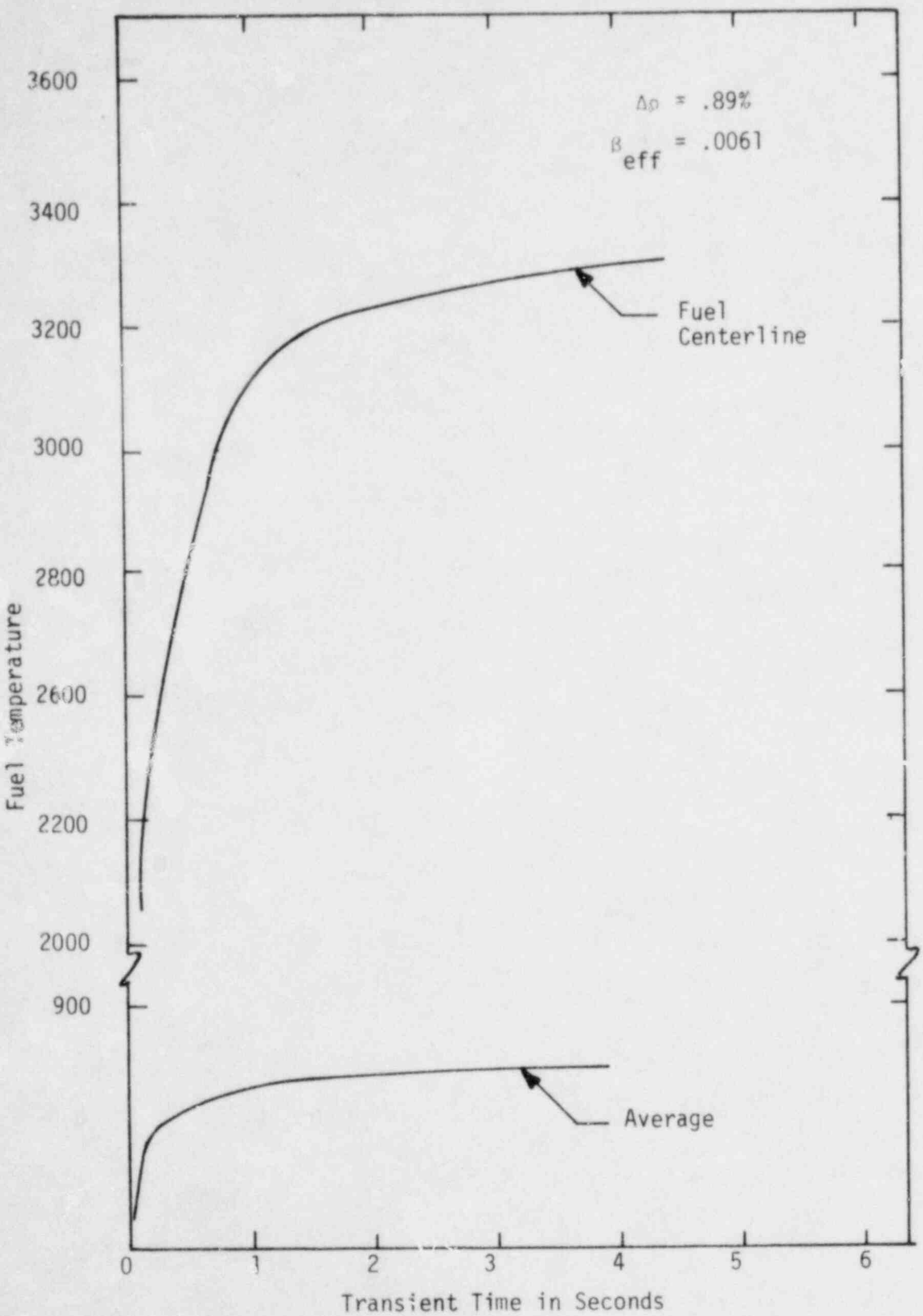

Figure 5.11 Centerline and Average Fuel Temperature vs Transient Time, Hot Zero Power

6.0 OVERPRESSURIZATION ASSOCIATED WITH ROD EJECTION ACCIDENT

6.1 INTRODUCTION

Following the unlikely occurrence of the ejection of a control rod, an increase in reactor system pressure results due to the deposition of the energy produced during the transient into the coolant. In order to insure that the reactor system is sufficiently protected against excessive overpressurization, a limiting criterion has been established and defined in Section 1.

A model to compute this pressure rise has been developed and is explained in detail in the following sections. The model was applied to the example rod ejection transient which reasonably envelopes the anticipated overpressurization. For this example, the model indicates that the maximum pressure anticipated is well below the allowable maximum transient pressure limit. Thus, adherence to the overpressurization criterion is met.

6.2 DESCRIPTION OF MODEL

The objective of the model is to calculate the pressure increase due to the abrupt increase in reactor power following a rod ejection. The model is based on six major assumptions as discussed below:

(1) Energy is immediately transferred to the coolant. The energy produced by the rod ejection is produced in the fuel rods. Thus, there is a delay time due to the thermal resistance of the fuel and gap and the heat capacity of the fuel before the heat is released to the coolant. However, this delay time is conservatively ignored, producing a larger energy release to the coolant than exists during the accident.

(2) Single phase water is incompressible. The water in the loops and reactor vessel is treated as incompressible. This assumption is conservative since accounting for the compressibility of the water the pressure surge would be reduced by about five percent.

(3) No mixing between water in the pressurizer at the start of the transient and water entering the pressurizer from the loops. Since the water in the loop is cooler than the pressurizer water, if mixing were allowed, additional steam condensation would occur thus reducing and/or eliminating the pressure surge altogether.

(4) No heat removal from the steam generators. Since heat removal from the steam generators would lower the average temperature during the transient, the thermal expansion of the coolant would be lower, reducing the pressure surge. This assumption thus maximizes the magnitude of the calculated pressure surge.

(5) Thermodynamic equilibrium in the pressurizer. This assumption allows for immediate condensation of the steam in the pressurizer.

(6) Complete mixing in the reactor primary system. This assumes an equal temperature rise in all parts of the reactor primary system.

With these assumptions the calculation proceeds as follows for the pressure increase associated with the rod ejection. The total energy increase is computed as:

$$\Delta E = \int_{t_o}^{T} p(\tau)d\tau \qquad (1)$$

where, $p(\tau)$ is the calculated time dependent reactor power level over the calculated time transient time $T - t_o$.

$p(\tau)$ is calculated using the neutronics model described in this document.

The reactor primary system internal energy associated with Eq. (1) is then defined as:

$$U_1^* = U_1 + \frac{E}{M} \tag{2}$$

where, U_1 is the initial primary system internal energy

M is the primary coolant total mass.

From the steam tables the specific volume of the primary coolant at the end of the transient is determined and the increase in primary coolant volume is determined as:

$$\Delta V = (\nu_1^* - \nu_1)\ M \tag{3}$$

where, ν_1 and ν_1^* are the primary coolant specific volume before and after the transient.

Knowing the change in volume (ΔV) an isentropic compression of the fluid in the pressurizer is calculated, that is

$$V_2^* = V_2 - \Delta V\ . \tag{4}$$

and

$$S_2^* = S_2\ . \tag{5}$$

$$M_2^* = M_2 \tag{6}$$

Combining Equations 3 and 5 results in

$$\nu_2^* = \nu_2 - {}^{\Delta V}/M_2 \tag{7}$$

Equations 5 and 7 uniquely determine the thermodynamic state of the water in the pressurizer and, thus, the pressurizer pressure at the end of the transient.

The above method results in a conservative estimate of the pressure surge associated with the rod ejection transient. An alternative approach, which would result in a more reasonable pressure surge estimate, is to model the entire reactor system using the calculated $P(\tau)$ as a driving function. The alternate approach would use a plant transient simulation model consistent with that used in determining the effects of anticipated reactor transients on thermal margins.

It is recommended that the model described herein be used to conservatively estimate the pressure surge associated with the rod ejection transient. However, the use of the alternate approach is allowed if the estimate, as defined above, is overly conservative. The alternate approach will be used only on a case-by-case basis.

6.3 EXAMPLE CALCULATION

An example calculation of the pressure surge associated with the rod ejection transient was selected from the results used in determining the parametric curves as shown in Section 4. The neutronics parameters for the example calculations are shown in Table 6.1, along with the total energy released during the transient. This transient was selected as being representative of the rod ejection transients for the ENC reload fuel. The values of $P(\tau)$ throughout the transient were determined from the appropriate XTRAN computer output and numerically integrated to obtain ΔE.

Using the total energy deposition as appears in Table 6.1, one obtains the new pressurizer specific volume (ν_2^*) as equal to 0.04626 ft^3/lb. On the basis of an isentropic process, one obtains the new pressurizer pressure (P_2^*) in the example case as a function of ν_2^*. This appears in Figure 6.1. Using Figure 6.1, one can estimate the peak pressurizer pressure as no greater than 2400 psia. This value is well below the allowable peak transient pressure of 2720 psia for the reactor vessel and pressurizer, presenting no impact upon existing plant technical specifications.

Table 6.1 Example Overpressurization Calculation

Parameter	Value
Delayed neutron fraction, β_{eff}	0.00606
Control rod worth, %	1.191
Peaking factor	5.79
Doppler coefficient, pcm/^{o}F	-1.027
Initial power level, MW	1.0
Total energy released, MW-sec	8.342×10^{3} MW-sec
Peak reactor pressure, psia	2400
Maximum allowable pressure, psia	2720

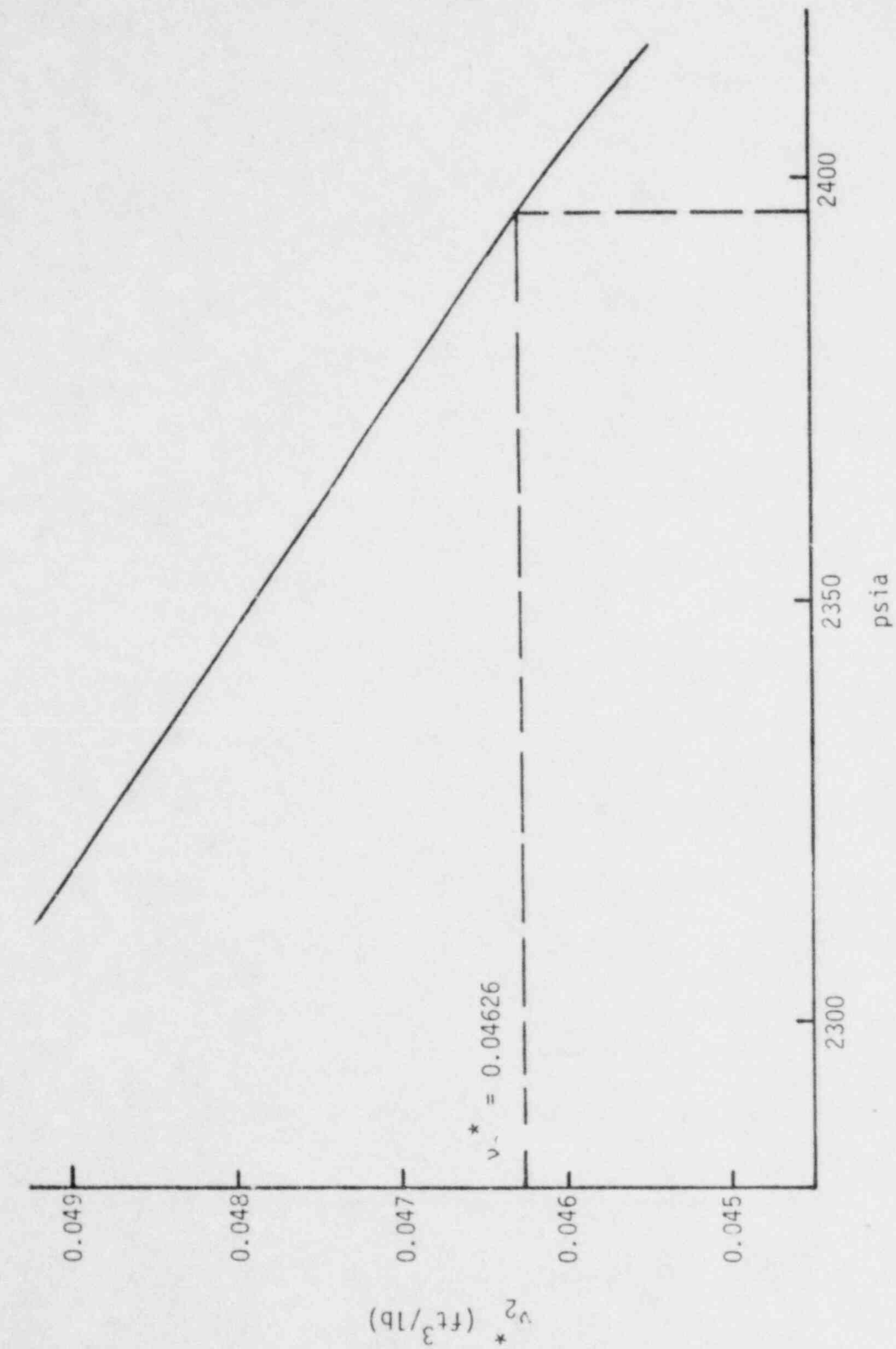

Figure 6.1 v_2^* as a function of pressure

7.0 APPLICATION OF GENERIC ANALYSIS

7.1 NEUTRONIC DESIGN PARAMETERS

The key neutronics parameters used for the actual control rod ejection accident evaluation are to be calculated for each cycle using PWR Neutronics methods consistent with ENC's methodology, which has been reviewed and accepted by the NRC.

The most severe control rod to be ejected is normally the maximum worth rod at hot full power and hot zero power conditions at any point in the cycle. The ejected rod worths and hot pellet peaking factors, before and after the ejection of the rod, are calculated with no pointwise feedbacks. Thus, no credit is taken for the power flattening effects of Doppler or moderator feedback in the calculation. The maximum rod worth and peaking factor, after ejection, are then applied to the parametric curves presented in Section 4.0 to determine the base deposited enthalpy for the accident. This base energy deposition is then corrected to account for differences in the Doppler coefficient, delayed neutron fraction, β, and initial conditions between the plant specific values and the generic rod ejection accident.

The Doppler reactivity coefficients as presented in Figures 4.1 and 4.2 are the differenctial coefficients evaluated for uncontrolled assemblies. In the reference transient analysis, the XTRAN model spatially treats the controlled and uncontrolled nodes with appropriate Doppler coefficient. However, to facilitate application of the parametric results for plants, only the uncontrolled Doppler coefficient needs to be calculated in order to be consistent with the reference control rod ejection analysis.

The delayed neutron fraction, β, is to be evaluated at the appropriate core exposure for each plant and cycle. As defined here, is determined by an importance weighted homogeneous core calculation of the effective delayed neutron fraction. For fuel designs with similar enrichments, β, is primarily exposure dependent.

If the rod ejection accident is to be evaluated at a different set of initial conditions than the generic report, a steady-state XTRAN calculation must be made. This calculation will provide a bias in the initial fuel enthalpies between the specific operation conditions and the generic report. Since the transient is performed adiabatically this bias can be applied directly to the parametric calculation as illustrated in the next subsection.

7.2 APPLICATION OF THE PARAMETRIC RESULTS

As a sample illustration, the peak deposited enthalpy resulting from a set of hypothetical conditions is determined using the parametric results presented in Section 4.0. The HZP conditions prescribed for this sample case are as follows:

Initial fuel enthalpy (cal/gm)	21.7
Maximum control rod worth (%$\Delta\rho$)	1.00
Doppler coefficient (pcm/^{O}F)	- 1.00
Power peaking factor	6.00
Delayed neutron fraction, β	.0058

Using Figure 4.3, the peak deposited enthalpy is determined to be 92.0 cal/gm for the 1.00% $\Delta\rho$ rod worth with a 6.00 power peaking factor. The difference in initial fuel enthalpy is determined as 5 cal/gm from Sections 4.2.10. This bias is summed to the 92.0 cal/gm to yield 97.0 cal/gm.

For a -1.00 pcm/^{O}F Doppler coefficient, the relative peak deposited enthalpy is to be increased by 1.03 as obtained from Figure 4.2. The deposited enthalpy is thus 1.03 * 97.0 cal/gm or 99.9 cal/gm. The multiplicative adjustment due to a β_{eff} of .0058 is 1.04 as determined from Figure 4.6 and the peak deposited enthalpy is 1.04 * 99.9 cal/gm or 103.9 cal/gm. Thus, the total enthalpy for this hypothetical case is 103.9 cal/gm. This resultant enthalpy is then compared to the 280 cal/gm limit to determine if the cycle design is acceptable with respect to a postulated control rod ejection accident.

The same procedure, as applied here for a sample case, can be employed to compute the peak deposited enthalpy resulting from a control rod ejection accident for any PWR plant.

8.0 REFERENCES

1. "Assumptions Used for Evaluating a Controlled Ejection Accident for Pressurized Water Reactors", NRC Regulatory Guide 1.77.
2. J. N. Morgan, XTRAN-PWR: A computer code for the calculation of rapid transients in pressurizea water reactors with moderator and fuel temperature feedback", XN-CC-32, September, 1975.
3. R. A. Hein, P. N. Flagett, "Enthalpy Measurements of UO_2 and Tungsten to 3260^{o}K, February, 1968 (GEMP-578).