XN-NF-78-51(NP)

EXXON NUCLEAR CONTROL ROD DROP ACCIDENT ANALYSIS FOR BIG ROCK POINT

JUNE 1981

RICHLAND, WA 99352

EXXON NUCLEAR COMPANY, Inc.

XN-NF-78-51(NP)
Issue Date: 7/06/81

EXXON NUCLEAR CONTROL ROD DROP
ACCIDENT ANALYSIS FOR BIG ROCK POINT

Prepared By

R. L. Feuerbacher

APPROVED: R B Stout
R. B. Stout, Manager
Neutronics and Fuel Management
30 JUN 81
Date

APPROVED: G F Owsley
G. F. Owsley, Manager
Reload Fuel Licensing
6/30/81
Date

APPROVED: George Sofer
G. A. Sofer, Manager
Nuclear Fuels Engineering
2 JUL 81
Date

gf

EXXON NUCLEAR COMPANY, Inc.

Table of Contents

List of Tables

List of Figures

1.0 INTRODUCTION AND SUMMARY

Exxon Nuclear Company (ENC) has performed a reference control rod drop accident analysis for Consumers Power Company's Big Rock Point Boiling Water Reactor. A control rod drop accident was simulated for an exposed core consisting entirely of ENC manufactured G-3 reload fuel assemblies. The second reload batch of the G-3 fuel type will be loaded for Big Rock Point's next operational period (Cycle 16). This reference control rod drop accident analysis applies directly to Cycle 16 operation and also generically to future operating cycles loaded with fuel similar to the G-3 design.

A transient, two dimensional (r-z cylindrical geometry) computer model with fuel temperature reactivity feedback is utilized for this analysis. The model simulates the reactivity insertion caused by a control rod being rapidly removed from the reactor core followed by the subsequent shutdown due to Doppler feedback and the scram rod bank entering the core. Prior to the start of the control rod drop accident, the initial core condition assumed is a hot standby, near critical state. For the analysis, a bounding minimum scram worth curve is employed to ensure that the reference rod drop accident results will apply for future cycles. Finally, the transient model computes the limiting consequences of the control rod drop accident in terms of the resultant peak energy deposition in the fuel.

In order to apply this reference rod drop analysis specifically to the Big Rock Point Cycle 16 operation as well as to future cycles, all important fuel assembly and core neutronic parameters are enveloped. The core variables that significantly affect the control rod drop accident consequences are determined to be:

- The subject control rod reactivity worth
- The Doppler coefficient
- The power peaking (after the control rod has been completely removed from the core)
- The delayed neutron fraction.

These parameters encompass the effects of potential variations in core loading patterns, fuel assembly enrichments, core exposures, and exposure distributions.

The results of this reference rod drop accident analysis are comprehensively parameterized with respect to maximum control rod worth, the Doppler reactivity coefficient, maximum power peaking, and the delayed neutron fraction. The parametric results of the Big Rock Point rod drop accident analysis are summarized in Figures 1.1 through 1.3 in terms of peak energy deposition in the fuel. These parametric results are presented in a format that facilitates application of this reference rod drop accident analysis to Big Rock Point operating cycles. Consequently, the results of these analyses may be applied to not only the core loading for Cycle 16 but to future Big Rock Point core conditions or configurations.

A sensitivity analysis was performed to determine the effects of the axial exposure distribution upon the outcome of the control rod drop accident analysis. This sensitivity analysis indicates that a typical axial exposure distribution with the higher exposed fuel peaked in the lower half of the core is conservative as compared to a uniform axial exposure distribution.

2.0 CONTROL ROD DROP ACCIDENT ANALYSIS

2.1 CONTROL ROD DROP ACCIDENT DESCRIPTION

The sequence of events postulated during a control rod drop accident are described as follows:

1. A fully inserted control rod becomes decoupled from its drive mechanism.
2. The drive mechanism is completely removed during the normal rod withdrawal sequence with the control rod remaining stuck in the reactor core.
3. Normal withdrawal sequence continues in the approach to criticality, despite the disconnected and stuck control rod.
4. At the worst time during the rod withdrawal sequence, the stuck rod suddenly becomes freed and falls out of the core.
5. The reactor subsequently becomes prompt critical resulting in a rapid power increase that eventually initiates a scram signal.
6. The core power reaches a maximum and then decreases rapidly due to Doppler reactivity feedback.
7. Subsequent power behavior depends upon the dropped rod velocity and worth, the scram delay time, and the scram bank velocity and reactivity worth. (The dropped rod is not included in the scram bank).
8. The reactor becomes and remains subcritical due to the combination of the scram bank insertion and Doppler feedback.

2.2 GENERAL DESCRIPTION OF ANALYSIS

The limiting consequence due to a control rod drop accident is calculated in terms of peak energy deposition in the fuel. Guideline values of stored energy content, corresponding to various degrees of fuel and/or cladding failure, have been established based on experimental studies (References 1 and 2). Thus, the objective of the control rod drop calculations is to determine if any fuel will exceed these guideline values during the unlikely occurrence of a control rod drop accident.

The general core conditions (hot standby, near critical state) assumed in this rod drop analysis for Big Rock Point are outlined in Table 2.1. For this analysis, the reactor core is assumed to be at essentially equilibrium cycle conditions loaded entirely with G-3 reload fuel assemblies. The core axial exposure distribution (presented in Table 2.2) input into this transient analysis is approximately the same as that actually calculated for the start of Cycle 15 operation.

The scram bank insertion velocity and delay time are obtained from Big Rock Point Technical Specifications that state:

10% of stroke	0.6 seconds (maximum time after receipt of scram signal)
90% of stroke	2.5 seconds (maximum time after receipt of scram signal)

The reference scram bank insertion as a function of the time from the receipt of the scram signal is plotted in Figure 2.1. The statically computed scram bank worth is conservatively set at 89×10^{-3} $\Delta k/k$ in order to envelope future cycles.

Since there are no control rod velocity limiters in Big Rock Point, the control rod dropped is assumed to be a free falling object accelerated by gravity. In the analysis, the postulated stuck rod begins to fall at the start of the transient calculation after a static power distribution has been solved.

To enable the application of this reference control rod drop analysis to the Big Rock Point Cycle 16 operation as well as to future cycles, all important fuel assembly and core neutronic parameters are conservatively represented. The significant kinetics parameters used in the reference rod drop analysis are presented in Table 2.3.

The Big Rock Point Cycle 16 core will consist of a mixture of G fuel types except for four F type assemblies which will be placed in low power locations not adjacent to control rods. These F types will therefore have no significant effects upon the control rod drop accident consequences. As stated previously, the reference control rod drop analysis is performed for a full core of all reload fuel. The G-3 fuel type has neutronic characteristics similar to and compatible with the previous G type designs (Reference 4). Therefore, the results of the reference control rod drop analysis can be applied to Cycle 16 by conservatively calculating the appropriate kinetics parameters discussed in Section 3.0.

2.3 TRANSIENT ANALYSIS METHOD

2.3.1 Transient Computer Model

A version of the XTRAN computer code (Reference 5) applicable to BWR cores is utilized for the Big Rock Point control rod drop accident analysis. The XTRAN code has been specifically developed to analyze the control rod drop accident. XTRAN 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 dependent neutron diffusion equation. The one-group cross sections used in the iterative flux solution are determined from input two-group values and modified at each time step by thermal feedback.

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 leaves 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. 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 times of large changes in power level, and conversely, large time steps during periods of slow perturbations. Therefore, the code efficiently solves the transient problems without the user choosing time step sizes. For the Big Rock Point control rod drop analyses, calculations are performed for a total time interval of six seconds.

Six groups of delayed neutron precursors are employed in the transient analyses. The decay constants and delayed neutron fractions utilized in the Big Rock Point rod drop analysis are presented in Table 2.4. These decay constants are obtained from Reference 6.

2.3.2 Control Rod Drop Analysis Method

The following is a step-by-step description of the procedure employed to perform the reference control rod drop accident analysis for Big Rock Point.

1. All input two-group cross sections, both uncontrolled and controlled, are calculated using standard ENC design methods for the hot standby conditions outlined in Section 2.2.
2. The reactor core is subdivided into three major radial zones that represent the total volume of the core (84 fuel assemblies). The core is subdivided into nine nodes axially.
   - The center radial zone is the dropped rod zone. Its volume is equivalent to a module of four-fuel assemblies.
   - The next inner zone is a partially controlled zone with a variable control fraction referred to as α_1. This radial zone is further divided into two radial subzones of equal widths approximately equivalent to an assembly pitch. The scram rod bank enters this zone.
   - The outer radial zone is also a partially controlled zone with a variable control fraction, α_2. This radial zone is subdivided into two radial subzones of

equal widths approximately equivalent to an assembly pitch. The scram rod bank also enters this zone. The cylindrical core geometry modeled in XTRAN for the rod drop accident simulation is illustrated in Figure 2.2.

3. The control fractions of the outer two radial zones, α_1 and α_2, are varied to obtain the desired reference control rod worths of 10, 20, and 30 mk. Static XTRAN solutions are made to calculate the control fractions. With the central zone controlled, an iterative search is made for the amount of control necessary in the outer zones to achieve K_{eff} = 1.000. Figure 2.3 represents a plot of α_2 versus α_1 for K_{eff} = 1.000 with the center zone fully controlled. With the center zone uncontrolled, a series of calculations are performed to determine the amount of control necessary in the outer zones to yield $K^*_{eff} = 1.000 + \Delta K_{CR}$ where K^*_{eff} denotes the reactivity of the core with the center rod removed and ΔK_{CR} is the desired rod worth. The calculated control fractions, α_2 versus α_1, are graphically depicted in Figure 2.3 for control rod worths of 10, 20, and 30 mk. The intersection points on the generated curves are the required pairs of α_1 and α_2 that will result in a K_{eff} = 1.000 when the center control rod is inserted and a $K^*_{eff} = 1.000 + \Delta K_{CR}$

with the center control rod removed. These sets of control fractions are utilized to produce the desired control rod worths in the transient analyses.

4. From the static XTRAN calculations in Step 3, the relative axial x radial power peaking factor is determined with the center control rod fully out for the three characterized control rod worths. Doppler and moderator density feedbacks are conservatively not included. For the parametric rod drop studies, the relative power peaking factors are varied for each control rod worth by changing the released energy per fission constants in the center control rod zone.

5. The input transient scram bank reactivity worth function employed in the rod drop accident is presented in Figure 2.4. Since for actual cycle design analysis the scram bank worth curve is calculated statically, a series of XTRAN static eigenvalue calculations are also performed to generate the input static scram worth curve presented in Figure 2.4. Doppler and moderator density feedbacks are conservatively not included. The input static scram bank reactivity worth is calculated to be 89×10^{-3} $\Delta k/k$ which should be definitely bounding for future cycles. For the actual cycle design calculations, the static scram bank should be demonstrated to be worth at least 89×10^{-3} $\Delta k/k$ in order to apply this reference rod drop analysis.

6. For the Big Rock Point reference control rod drop accident analysis, an adiabatic boundary condition is assumed at the fuel pellet-gap interface. In other words, no heat transfer to the coolant is allowed. This is a conservative assumption since in this way the peak deposited enthalpy in the fuel is determined from the sum of all the energy absorbed by the fuel during the transient period with no credit for heat escape from the fuel.

The calculational method is very automated since XTRAN was developed specifically for the control rod drop accident analysis. At the start of the transient solution (time = 0), the center control rod immediately begins to fall from the core. The rod is removed at the rate of input acceleration. When the core peak reaches scram level magnitude, the power trip is signaled. The scram rod bank then begins entering the outer two zones (see Figure 2.2) after an input delay time. The total time analyzed for this transient in the rod drop studies is six seconds.

3.0 RESULTS

3.1 GENERAL DISCUSSION

The Big Rock Point reference control rod drop accident analysis has been completed for the core conditions outlined in Tables 2.1 and 2.2 and the key neutronic parameters presented in Table 2.3. The nominally calculated results, in terms of peak deposited enthalpy in the fuel, are summarized in Table 3.1 and plotted in Figure 3.1.

The peak enthalpy results in Table 3.1 are presented as axial x radial assembly depositions. Thus, in computing the peak deposited enthalpy in the assembly's "hottest" rod, the axial x radial peaks are to be multiplied by the assembly local pin power factor. The local pin power is the peak to average power in the assembly as determined by auxiliary assembly diffusion theory calculations. For example, the total peak deposited enthalpy is calculated using Table 3.1 for the following parameters:

Control rod worth (mk)	=	10
Doppler coefficient ($\Delta k/k/^{o}F$)	=	-9.52×10^{-6}
Axial x radial assembly peaking factor	=	2.31
Delayed neutron fraction, β	=	.00525
Assembly local peaking factor	=	1.20

In order to illustrate the mechanics of the control drop accident, the relative core power experienced during the simulated transient is presented as a function of time in Figure 3.2. The three characterized rod worths (10, 20, and 30 mk) for a -9.52×10^{-6} $\Delta k/k/^{o}F$ Doppler coefficient and a .00525 β are depicted. As the postulated stuck control rod is initially removed from the core, Figure 3.2 shows that the power begins to increase rapidly after approximately 0.4 seconds. The scram power trip set at 122% of rated reactor power occurs after 0.37 seconds for the 30 mk rod, 0.42 seconds for the 20 mk rod, and 0.56 seconds for the 10 mk rod.

Due to the rapidly increasing reactor power, the fuel temperature also rises quickly causing the Doppler feedback to compensate the reactivity insertion produced by the falling rod. The primary power peak, shown in Figure 3.2 for the three rod worths, occurs when the Doppler feedback exactly balances the dropped rod reactivity insertion. After the primary peak, the Doppler feedback becomes the dominating factor, and the core power is reduced.

For the higher control rod worths and power peaking factors, the Doppler feedback quickly arrests the reactivity insertion before the control rod is completely removed. Consequently, there is a smaller, secondary power increase due to the additional reactivity added by the remainder of the control rod. In other words, the reactivity insertion by the falling control rod once again becomes the dominating factor. No secondary power increase occurs for the 10 mk rod case since the control rod is completely out after 0.60 seconds. Hence, there is no additional reactivity to be inserted after the Dopper feedback begins to reduce the reactivity.

The scram rod bank begins to enter the core 0.375 seconds after receipt of the scram signal. Therefore, the scram bank begins to enter after 0.74 seconds of elapsed time for the 30 mk rod, 0.79 seconds for the 20 mk rod, and 0.93 seconds for the 10 mk rod. Figure 3.2 shows when the power is reduced by the scram rod bank.

Based on these transient results, it is evident that the Doppler feedback is the primary mechanism for shutting the reactor down during a control rod drop accident. The scram reactivity worth function is only of secondary importance.

3.2 PARAMETRIC RESULTS

In order to apply the reference Big Rock Point control rod drop accident analysis to Cycle 16 operation and future cycles, the calculated results have been comprehensively parameterized. The important parameters that significantly affect the results of the control rod drop analysis are control rod reactivity worths, Doppler coefficients, power peaking factors, and delayed neutron fractions. These variables encompass the effects of core loading patterns, fuel assembly enrichments, core exposures and exposure distributions. The results of these parametric studies may therefore be applied to the specific core loading for Cycle 16 and also to future Big Rock Point core conditions or configurations.

The axial x radial peak deposited enthalpy in the fuel is presented as a function of control rod reactivity worth and axial x radial power peaking factors (statically calculated with the subject control rod entirely withdrawn and no Doppler feedback) in Figure 1.1. These peak results are given for a -9.52×10^{-6} $\Delta k/k/^{o}F$ Doppler reactivity coefficient and a .00525 β.

Figure 1.2 is to be employed to convert the peak deposited enthalpy obtained from Figure 1.1 for a -9.52×10^{-6} $\Delta k/k/^{o}F$ Doppler coefficient to that for the desired Doppler coefficient. In Figure 1.2, the relative deposited enthalpy factor is presented as a function of Doppler coefficient.

The relative peak deposited enthalpy is presented in Figure 1.3 as a function of the delayed neutron fraction, β. This figure is to be employed to include the effect of β on the peak enthalpy obtained from Figures 1.1 and 1.2. As demonstrated by Figure 1.3, the delayed neutron fraction is of secondary importance in the control rod drop analysis.

As stated in Section 3.1, the assembly local pin power factor must be applied to these axial x radial peak enthalpy results to determine the peak deposited enthalpy. The initial fuel enthalpy also must be added to compute the total peak enthalpy accrued during the rod drop accident. Section 3.3 outlines a sample case for applying these parametric rod drop accident results to determine the peak deposited enthalpy.

3.3 APPLICATION OF PARAMETRIC RESULTS

For a sample calculation, the peak deposited enthalpy resulting from hypothetical conditions is evaluated using the parametric results presented in Section 3.2. The conditions prescribed for this sample case are as follows:

Maximum insequence control rod worth (mk)	=	12
Doppler coefficient ($\Delta k/k/^{o}F$)	=	-10.1×10^{-6}
Axial x radial assembly peaking factor	=	3.90
Delayed neutron fraction, β	=	.0058
Assembly local peaking factor	=	1.20

The key neutronics parameters used for the actual control rod drop accident evaluation are to be calculated for each cycle with a core simulator model and other peripheral assembly design calculations. The most severe control rod to be dropped is normally the maximum insequence rod worth at hot standby conditions occuring at the peak core reactivity point in the cycle. The peak axial x radial power peaking factor is usually located in the same module as the highest worth rod. This axial x radial power factor (also referred to as the "nodal" power peaking) is calculated using a core simulator model. The calculation is to be performed at a hot standby conditions with the dropped rod completely removed and the scram bank not inserted. No Doppler feedback is included.

The Doppler reactivity coefficient as presented in Figure 1.2 is the differential coefficient evaluated at hot standby conditions (583^{o}F) for uncontrolled assemblies. In the reference transient analysis, the XTRAN model spatially treats the controlled and uncontrolled nodes with appropriate Doppler coefficients. However, to facilitate application of the parametric results for similar fuel types, only the uncontrolled Doppler coefficient needs to be calculated for each cycle in order to be consistent with the reference control rod drop accident.

The assembly local power peaking factor is to be calculated with peripheral fine mesh diffusion theory calculations. If justified, additional engineering factors can be applied to the control rod drop analysis by including the conservative factors with the local peaking factor.

The delayed neutron fraction, β, is to be evaluated at the appropriate core exposure for each cycle. For fuel designs with similar enrichments, β is primarily exposure dependent.

The same procedure, as applied here for a sample case, can be employed to compute the peak anthalpy resulting from a rod drop accident for Big Rock Point Cycle 16 operation as well as future cycles loaded with fuel similar to the G-3 design.

Table 2.1 Big Rock Point Core Condition for Control Rod Drop Analysis

Core loading: G-3 type reload fuel.

Beginning of cycle (BOC) core average exposure: 9.9 GWD/MTM.

Fuel exposure distributions: distribution axially; uniform radially.

Fuel temperature at beginning of transient: $583^{o}F$.

Moderator conditions (saturated): $583^{o}F$; 0 void fraction.

Reactor power level at beginning of transient: 240×10^{-6} MWt.

Scram bank insertion velocity: 2.46 ft/sec.

Scram delay time: 0.375 sec.

Dropped rod acceleration: -32.2 ft/sec.^{2}

Scram power (122% rated): 292.8 MWt.

Static scram bank worth: $89. \times 10^{-3}$ $\Delta k/k$.

Initial fuel stored enthalpy: 18. cal/gm.

Table 2.2 Big Rock Point Axial Exposure Distribution For Control Rod Drop Accident Analysis

Axial Node	Fuel Exposure (GWD/MTM)
9 (top)	
8	
7	
6	
5	
4	
3	
2	
1 (bottom)	
Core Average Exposure	

Table 2.3 Big Rock Point Key Kinetics Parameters For Control Rod Drop Analysis

Control Rod Worth (mk):	10.0
	20.0
	30.0
Power Peaking* (Relative axial x radial factor):	2.31
	3.12
	3.70
	5.00
Doppler Coefficient** ($\Delta k/k/^{o}F$):	-10.71×10^{-6}
	-9.52×10^{-6}
	-8.33×10^{-6}
Delayed Neutron Fraction ($\bar{\beta}$):	.00400
	.00525
	.00650

* Static, hot standby core conditions with the control rod to be dropped fully removed.

** Differential Doppler coefficient evaluated at hot standby conditions (583^{o}F) for uncontrolled assembly configuration.

Table 2.4 Big Rock Point Control Rod Drop Analysis Input Delayed Neutron Constants

Delayed Neutron Group j	Fractional Group Yield, $\beta_j/\bar{\beta}$	Decay Constant λ_j (sec^{-1})
1	0.038	0.0127
2	0.213	0.0317
3	0.188	0.115
4	0.407	0.311
5	0.128	1.40
6	0.026	3.87
	1.000	

Table 3.1 Big Rock Point Control Rod Drop Accident Analysis Nominal Results

Control * Rod Worth Δk (mk)	Doppler Coefficient (10^6 $\Delta k/k/^oF$)	Power Peaking* (Axial x Radial Relative Factor)	Delayed Neutron Fraction, $\overline{\beta}$	Peak Deposited** Enthalpy (cal/gm)
10	-10.71	2.31	.00525	
20	-10.71	3.12	.00525	
30	-10.71	3.70	.00525	
10	- 9.52	2.31	.00525	
20	- 9.52	3.12	.00525	
30	- 9.52	3.70	.00525	
10	- 8.33	2.31	.00525	
20	- 8.33	3.12	.00525	
30	- 8.33	3.70	.00525	

* Control rod reactivity worth and relative power peaking factor determined by static calculations.

** The peak deposited enthalpy results are axial x radial assembly values. To obtain the total peak deposited enthalpy, apply the local peaking factor for the assembly. Finally, add the initial fuel enthalpy to compute the total peak enthalpy accumulated in the fuel.

Figure 1.1

Figure 1.2

Figure 1.3

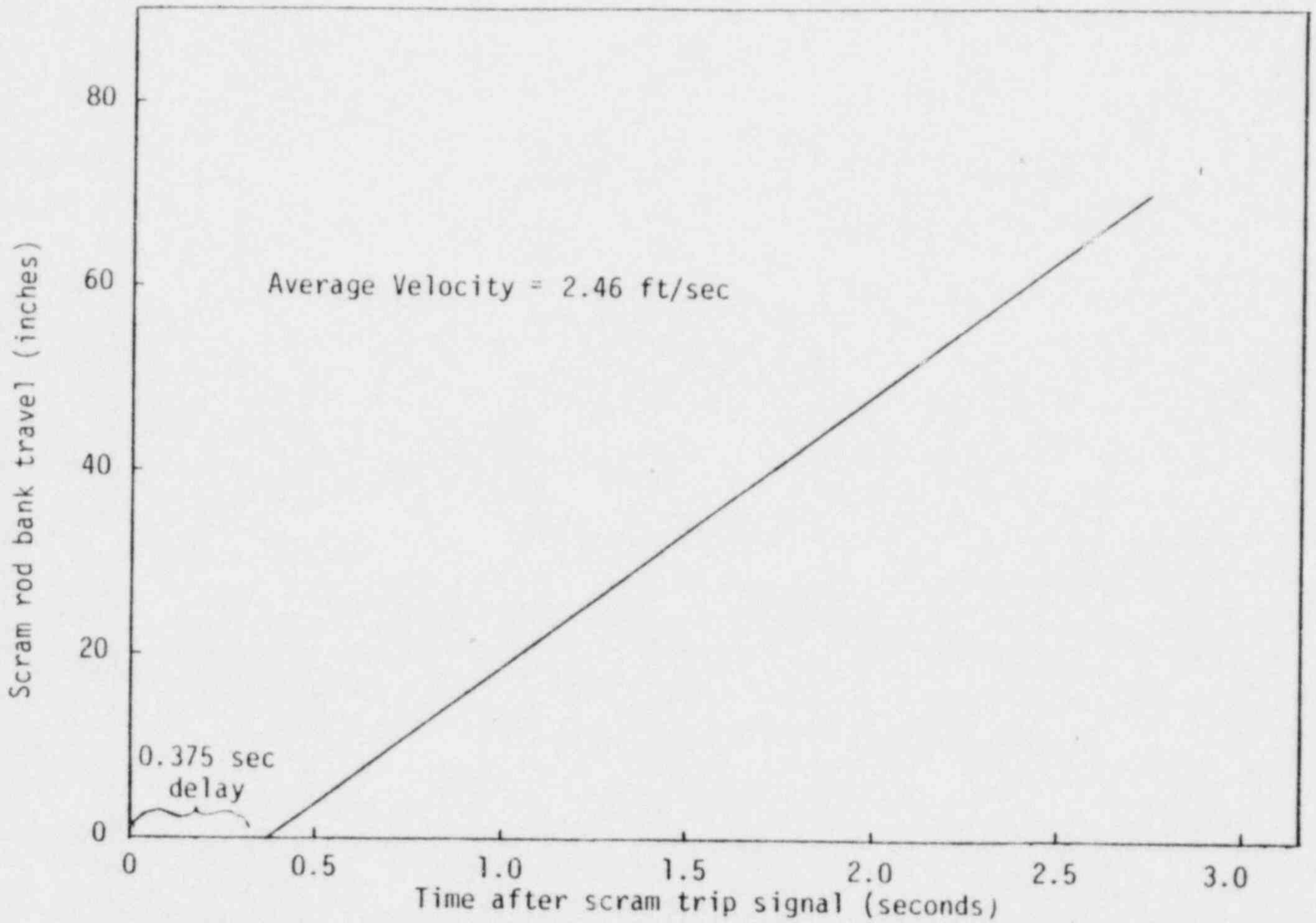

Figure 2.1 Big Rock Point Scram Bank Insertion vs. Time From Receipt of Scram Signal

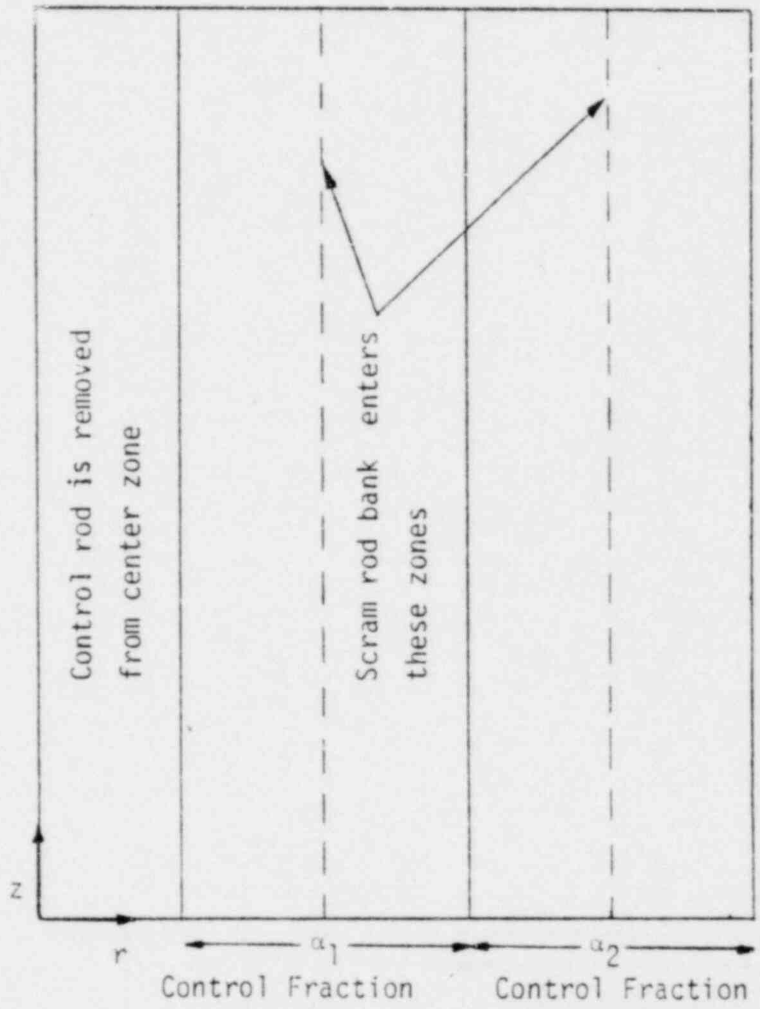

Figure 2.2 Big Rock Point Cylindrical Rod Drop Accident Model

Figure 3.1

Figure 2.4

Figure 3.1 Big Rock Point Peak Deposited Enthalpy
vs Control Rod Worth

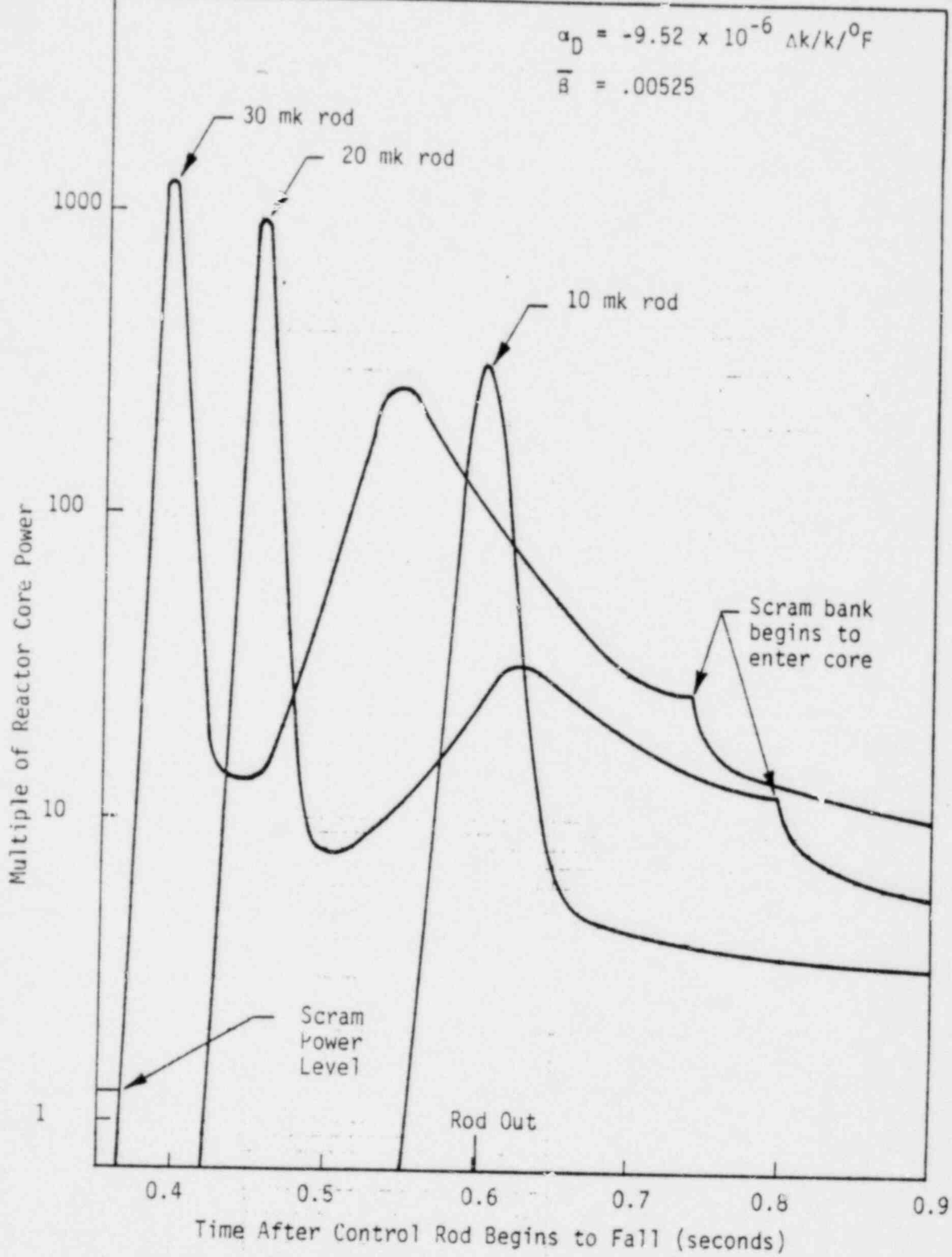

Figure 3.2 Big Rock Point Core Average Power vs Time During Control Rod Drop Accident

4.0 REFERENCES

1. C. J. Paone, et. al., Rod Drop Accident Analysis for Large Boiling Water Reactors, NEDO-10527, 72 NED 18, Class 1, March, 1972.

2. J. E. Grund, et. al., Subassembly Test Program Outline for FY 1969 and FY 1970, IN-1313, August, 1969 (IDO-17277).

3. Technical Specifications for Big Rock Point Plant, Consumers Power Company, Docket No. 50-155, As amended through October 17, 1977.

4. Design Report Big Rock Point Reactor Reload G Fuel, JN-72-17, July 31, 1972. (Addenda 1 through 4)

5. J. N. Morgan, XTRAN-PWR: A Computer Code for the Calculation of Rapid Transients in Pressurized Water Reactors with Moderator and Fuel Temperature Feedback, XN-CC-32, September, 1975.

6. R. J. Tuttle, Delayed-Neutron Data for Reactor-Physics Analysis, Nuclear Science and Engineering: 56, 1975.

APPENDIX A

ROD DROP ACCIDENT ANALYSIS

ADDITIONAL STUDIES

A.1 AXIAL EXPOSURE DISTRIBUTION SENSITIVITY

A sensitivity analysis was performed to determine the effects of the axial exposure distribution upon the outcome of the control rod drop accident analysis. The reference rod drop results presented in Section 3.0 are based on a typical axial exposure distribution (obtained at BOC 15). For this sensitivity analysis, a rod drop accident case was simulated assuming a uniform axial exposure distribution with the same average exposure as the reference analysis

The particular case reanalyzed was the 20 mk control rod with an axial x radial peaking of 3.70, a -9.52×10^{-6} $\Delta k/k/^{o}F$ Doppler coefficient, and a .00525 $\bar{\beta}$. The transient results indicate that the uniform axial exposure distribution is not as conservative as the nonuniform distribution. The peak axial x radial deposited enthalpy calculated for the uniform case is

The primary reason for the calculated difference in deposited enthalpies is the effect of the axial exposure distribution upon the axial power shape. The reference case with the higher exposure peaked towards the core bottom produces an axial power shape peaked in the upper half of the core. On the other hand, the uniform axial exposure distribution produces an axial power shape that is symmetric about the core centerline. As compared to the uniform case, the power rise for the reference case begins sooner because the falling rod leaves the nodes with higher power earlier. The reactivity insertion rate for the reference case is thus higher during the initial part of the transient, and consequently, the reference case reaches its primary power burst before the uniform exposure case. The peak core power is approximately equivalent for both cases, but the primary power peak occurs at 0.45 seconds for the reference case and 0.50 seconds for the uniform case. Since the dropped control rod requires 0.6 seconds to fall completely out of the reactor core, there is a longer period of time between the primary peak and the rod out point for the reference case. Therefore, the reference case produces a higher secondary power increase than the uniform case. This increased secondary peak is the principal cause of the peak enthalpy difference between the reference and uniform axial exposure case.

A secondary factor contributing to the difference between the uniform and nonuniform axial exposure transient results is the scram. Since the scram rods enter from the bottom of the core, the scram rods reach the nodes with higher powers (and thus higher deposited enthalpies) earlier for the uniform case than the reference nonuniform case. Therefore, the scram bank terminates power generation in the peak enthalpy nodes earlier in the uniform exposure case than in the reference case. Since the impact of the rod drop transient is mainly constrained by Doppler feedback (as shown in Section 3.1), this effect due to the scram bank upon the power distribution is minimal. A comparison of the core power histories for the nonuniform and uniform axial exposure transients is presented in Figure A.1.

This sensitivity study demonstrates that a typical axial exposure distribution with the higher exposed fuel in the lower half of the core is conservative. The least conservative case would be a hot standby power profile with the peak near the bottom of the core.

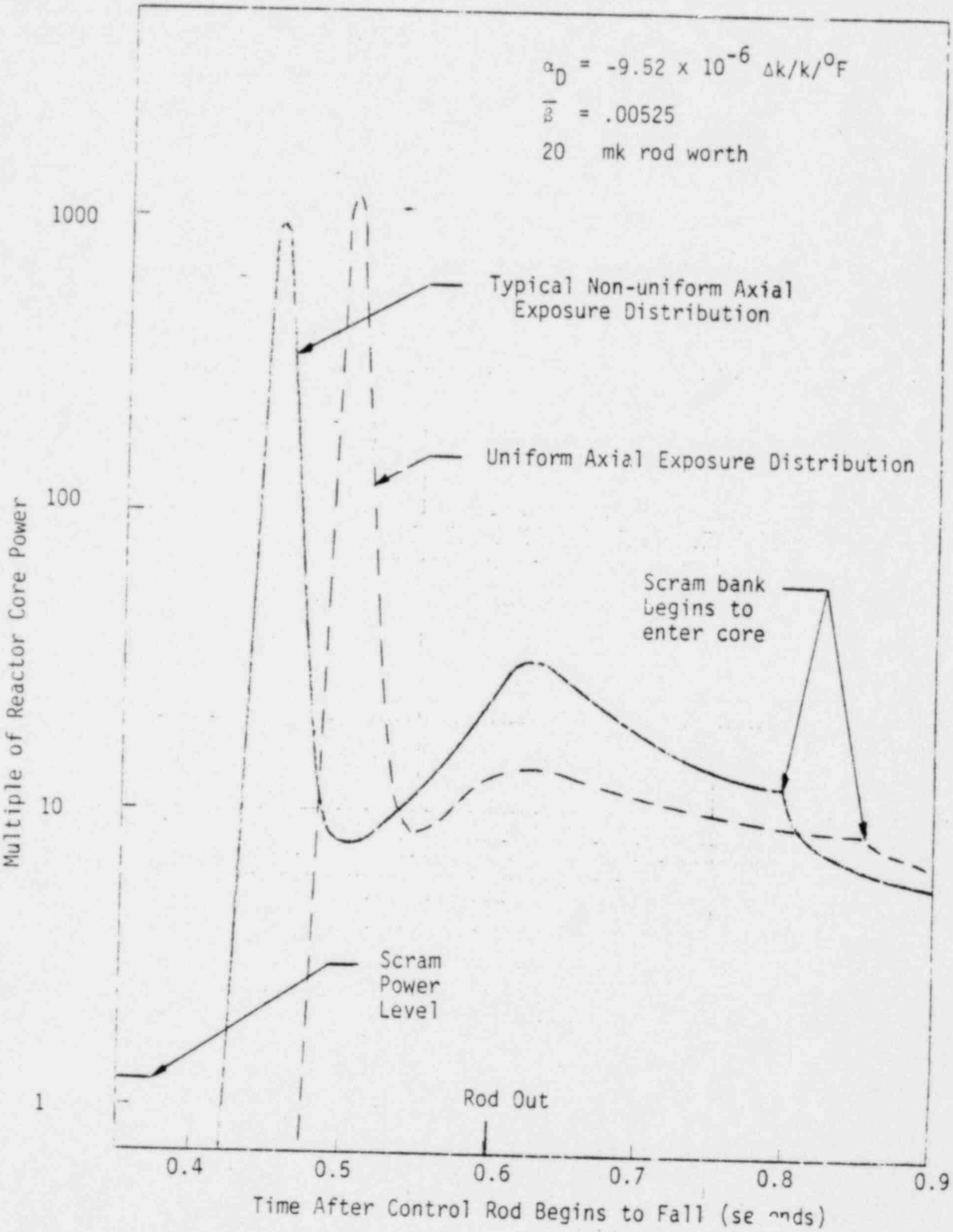

Figure A.1 Comparison of Core Average Power during Control Rod Drop Accident for Typical and Uniform Axial Exposure Distribution

XN-NF-78-51(NP)
Issue Date: 7/06/81

EXXON NUCLEAR CONTROL ROD DROP
ACCIDENT ANALYSIS FOR BIG ROCK POINT

Distribution

J. L. Maryott

G. F. Owsley

H. G. Shaw (5)

G. A. Sofer

R. B. Stout

Document Control (5)