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ANALYZING THE BWR ROD DROP ACCIDENT IN HIGH-BURNUP CORES[1]

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EXECUTIVE SUMMARY

This study was undertaken for the U. S. Nuclear Regulatory Commission to determine the fuel enthalpy during a rod drop accident (RDA) for cores with high burnup fuel. The calculations were done with the RAMONA-4B code which models the core with 3-dimensional neutron kinetics and multiple parallel coolant channels. The calculations were done with a model for a BWR/4 with fuel bundles having burnups up to 30 GWd/t and also with a model with bundle burnups to 60 GWd/t.

The calculations were done assuming initial conditions at zero power with the coolant 70°C subcooled. The control rod pattern was at 50 percent control density and the rod dropped had a static worth of 950 pcm. The RDA caused a power excursion that was initially terminated by fuel temperature (Doppler) feedback. The power remained at relatively high levels until void feedback and reactor trip reduced the reactivity sufficiently.

The maximum increase in fuel enthalpy in the core was less than 70 cal/g for the medium burnup core which is low relative to existing acceptance criteria for this event. The enthalpy rise was determined not only by the dropped rod worth and magnitude of the feedback but also by the timing of the feedback. With large subcooling, the generation of void feedback is delayed and the fuel enthalpy continues to rise after the initial increase in enthalpy due to the power pulse.

The maximum increase in fuel enthalpy was calculated for the bundles surrounding the dropped rod and then plotted versus the burnup of the node in which the maximum occurred. The results of the calculations were consistent with the expectation that the peak fuel enthalpy in any bundle would be a complicated function of the dropped control rod worth, the distance of the bundle from the dropped rod, and the burnup at that location. This result was found to be the case for both the medium and high burnup cores for which the RDA was calculated.

This paper also discusses potential sources of uncertainty in calculations with high burnup fuel. One source is the "rim" effect which is the extra large peaking of the power distribution at the surface of the pellet. This increases the uncertainty in reactor physics and heat conduction models that assume that the energy deposition has a less peaked spatial distribution. Two other sources of uncertainty are the result of the delayed neutron fraction decreasing with burnup and the positive moderator temperature feedback increasing with burnup. Since these effects tend to increase the severity of the event, an RDA calculation for high burnup fuel will underpredict the fuel enthalpy if the effects are not properly taken into account. Other sources of uncertainty that are important come from the initial conditions chosen for the RDA. This includes the initial control rod pattern as well as the initial thermal-hydraulic conditions.

INTRODUCTION

Reactivity-initiated accidents and certain design-basis transients lead to power excursions which are considered acceptable if they meet specified acceptance criteria. For rapid power excursions, these criteria are based on the energy deposition in the fuel pellet which is approximately equal to the fuel enthalpy. In recent years, experiments have been performed to examine the behavior of high burnup fuel subjected to power pulses. Some fuel has failed at energy depositions that are low relative to the acceptance criteria. Furthermore, other recent studies of high burnup fuel show that property changes, especially in the cladding and at the surface of the pellet, could make the fuel more vulnerable to power pulses. These activities have called into question the current acceptance criteria, and new studies to address this issue have been undertaken by the light water reactor community throughout the world.

The U. S. Nuclear Regulatory Commission (NRC) has expressed its concern regarding the above in two Information Notices that have been issued: (NRC, 1994) and (NRC, 1995). In addition, the NRC sponsored a research program to improve their understanding of the situation and to see if regulatory action is needed. This program has three different thrusts. One is to study the new experimental data being collected in France, Japan, and Russia as well as the data available from measurements made in the past, especially in the U. S. This is intended to enable the NRC to better understand how burnup affects fuel behavior during power excursions. Another is to improve analytical models of fuel behavior so that they are applicable at high burnup. This would enable analytical studies of fuel behavior to be completed. The third thrust is to review the transient/accident analysis that has been done in the past and to perform new calculations that will estimate the amount of energy that can be deposited in high burnup fuel. This paper reports on a part of this last thrust. Specifically, it presents results of RAMONA-4B calculations of the rod drop accident in a boiling water reactor (BWR). This event leads to the largest energy deposition for BWR design basis accidents.

ANALYSIS OF THE BWR ROD DROP ACCIDENT

Description of RAMONA-4B

RAMONA-4B (Wulff, 1984) is a systems transient code for boiling water reactors. The code uses a 3-dimensional neutron kinetics model coupled with a multichannel, 2-phase flow model of the thermal-hydraulics in the reactor vessel. The code is designed to analyze a wide spectrum of BWR core and system transients. The 3-dimensional neutron kinetics makes the code well-suited for predicting transients and accidents where the spatial core power variations are expected to be significant.

The reactor core is modeled with multiple parallel coolant channels and a bypass channel. Each coolant (i.e., thermal-hydraulic) channel is interfaced with one or more fuel bundles. The reactor power, including decay heat, is calculated in 3-dimensional geometry. The fission power calculation takes into account control rod movement (including accidental rod drop and reactor trip) and the feedback throughout the core due to changes in the fuel and coolant temperatures and steam void fraction. Energy deposited directly into the coolant and bypass channels is taken into account. Thermal conduction through the fuel pellet, gas gap, and fuel cladding is modeled to obtain the heat transfer from the fuel to the coolant.

The neutron kinetics model of RAMONA-4B is based on 2-group diffusion theory with up to six delayed neutron precursor groups. Simplifications are made in treating the thermal neutron flux to reduce the formulation to a 1½ group, coarse mesh diffusion model in a 3-dimensional rectangular coordinate system. Neutronic boundary conditions are specified at the axial and radial core periphery.

The thermal-hydraulic calculation in RAMONA-4B is based on a 4-equation, nonequilibrium, drift-flux model. The four balance equations are conservation of: (1) vapor mass, (2) mixture mass, (3) mixture momentum, and (4) mixture energy. Although the thermal-hydraulics modeling extends outside of the core to the vessel, steamline, and recirculation loop, this is not important to the calculation of the rod drop accident.

BWR Reactor Model

A BWR/4 reactor core was modeled with RAMONA-4B using half-core mirror symmetry. The model included 382 neutronic channels where each channel represented a single fuel bundle. The core model included 100 control rods. The number of calculational nodes in the vertical direction was 24.

The thermal-hydraulics of the core region was modeled using 160 thermal-hydraulic channels associated with fuel bundles and one bypass channel representing the area between the bundles. The majority of the thermal-hydraulic channels were "shared" by several neutronic channels. The thermal energy released in those several neutronic channels was collectively deposited into the liquid flowing in that particular thermal-hydraulic channel. Each of the neutronic channels in three rows of bundles adjacent to the core's axis of symmetry had a dedicated thermal-hydraulic channel in order to most accurately represent the thermal-hydraulic reactivity feedback effects (void fraction and moderator and fuel temperature) following a control rod drop.

Two cores were modeled. One model was for a medium burnup core and represented fuel bundles with burnups up to a maximum of approximately 30 GWd/t. The cross sections for this core had been generated using the CASMO code (Ahlin, 1978) for a previous study.

The other model was meant to represent the situation with bundle average burnups up to 60 GWd/t (and, hence, fuel rod burnups of up to approximately 65 GWd/t). Since no data were available to the authors for actual or planned cores with this burnup, a method was used which allowed for the medium burnup data to be extrapolated to produce the high burnup core. New cross sections were generated using the CPM code (Ahlin, 1975). This core model is only an approximation to an actual core. However, it provides sufficient information to test certain hypotheses and add to our understanding of high burnup cores.

Initial Conditions for RDA Analysis

The calculation of rod drop accidents was done for both the medium and high burnup core models. Table 1 contains some of the neutronic and thermal-hydraulic parameters used to describe initial conditions, plant response, and modeling in RAMONA-4B for these calculations.

Table 1 Reactor Model Parameters for Medium and High Burnup RDA Cases

Parameter/Condition	Value/Description	Comment
Fuel bundle maximum burnup	30/60 GWd/t	For medium/high burnup calculations
Reactor power	3.29 kW ("zero" power)	10^{-6} of rated power
Control rod insertion pattern	Checkerboard; 50% control rod density	See Figure 1
Fraction of energy deposited directly into coolant	0.04	Total for the in-channel and bypass liquid
Delayed neutron fraction	0.006/0.005	For medium/high burnup calculations
Xenon inventory	Fully depleted	
Reactor trip setpoint	15% of rated power with 0.2 s delay	
Scram insertion speed	1.2 m/s (3.9 ft/s)	
Control rod drop speed	0.94 m/s (3.1 ft/s)	
System pressure	0.1 MPa	Non-condensible atmosphere
Liquid temperature	30°C	70°C subcooled
Core flow rate	3260 kg/s	25% of rated flow

The initial control rod pattern with 50 percent control density, shown in Figure 1, was chosen for several reasons. The most important was that in a study of limited scope, it would be too difficult to search through all of the possible patterns to obtain a pattern with the highest worth dropped control rod or the worst fuel enthalpy increase. BWR reactors use systems that lead to patterns such as those in the banked position withdrawal sequence (Paone, 1977). Not only would one have to go through all the patterns possible using this system but also patterns possible if a single failure criterion was applied. With the 50 percent control density, control rod worths of up to 950 pcm were calculated along the axis of symmetry. This highest worth corresponds to rod worths obtained by other analysts using the banked position withdrawal sequence; and, therefore, it was felt justified to use for the rod drop analysis.

The initial thermal-hydraulic conditions in the reactor corresponded to cold startup. The power was 10^{-6} of full power, and the core coolant temperature was 30°C. This represented 70°C subcooling at atmospheric pressure which was assumed to be the system pressure. This delays the onset of steam generation caused by the RDA and, therefore, the addition of negative void reactivity feedback which tends to mitigate the accident. The single-phase coolant decreases the heat transfer to the coolant relative to the case with 2-phase flow. This has the effect of keeping the fuel temperature (and fuel enthalpy) higher; and, as with the void reactivity, this is in the direction so as to make the results more severe, i.e., more limiting. The higher fuel temperature also increases the fuel temperature reactivity feedback which limits the severity of the accident, but this is expected to be a smaller effect. The high subcooling at low initial temperature means that coolant/moderator temperature reactivity feedback can be important. For a BWR at cold conditions, the feedback is positive and, therefore, can exacerbate the power excursion.

In most BWRs, the reactor becomes critical when only approximately one fourth of the control rods are withdrawn. Hence, cold conditions would correspond to higher control rod densities than the 50 percent used in this study. At 50 percent control density, higher temperatures and pressures are expected as the power would have increased from its initial level at the cold condition. Best estimate calculations would have to take into account the change in thermal-hydraulic conditions with changing control rod patterns. The thermal-hydraulic conditions control the positive moderator feedback, the heat transfer to the coolant, and the onset of negative void feedback.

The initial conditions for the medium burnup core results in a (high) axial peaking factor of 3.5 at the top of the core--typical of shutdown conditions in a BWR. This axial peaking tends to increase the rate of reactivity insertion when the rod drops out of the core. This means that the power increases rapidly while the control rod is still in the top half of the core.

Results for a Medium Burnup Core

The accident was initiated at time zero with CR #14 (see Figure 1) dropping at a speed of 0.94 m/s (3.1 ft/s). The prompt power excursion started at about one second, as can be seen in Figure 2 which shows the power during the transient on a logarithmic scale relative to nominal, or rated, power. The power increases more than six decades which is typical for this type of RDA.

The figure also shows the position of the control rod which is initially completely inserted. As can be seen, when the tip of the control rod traveled only three to four feet through the core, sufficient reactivity had been inserted to cause the power excursion which, in turn, was terminated by fuel temperature feedback (primarily due to the Doppler effect). This means that when realistic control patterns are considered in setting up conditions for the RDA, it is only necessary that the control rod drive mechanism be withdrawn halfway out of the core in order to set the stage for the assumption that the corresponding blade has been decoupled and has stuck so that it can later drop to the position of the drive mechanism.

The reactor power reached a peak value of approximately 2.4 of nominal power at about 1.3 seconds. At that time, the negative Doppler reactivity feedback is large enough so that the power excursion is terminated. The history of the different reactivity feedback components is shown in Figure 3 which also shows the power excursion on a linear scale. This figure shows that the accident can be separated into four major phases. In the first phase, reactivity is being inserted due to withdrawal of the dropped control rod. The second phase starts when the power surge is reversed due to fuel temperature (Doppler) reactivity. The third phase covers the period from the initiation of boiling in the core and its associated negative reactivity. The fourth phase occurs when the void feedback and scram become effective enough to completely shut down the core.

The plot of reactivity effects shows that the control rod worth germane to this event is approximately 750 pcm. This is 80 percent of the total static worth of the rod and primarily is the result of the fact that the rod does not withdraw all the way before the event is terminated. The figure also shows the positive reactivity feedback due to moderator heatup.

The axial power distribution also changes during the transient, but because it is peaked at the top of the core initially and the rod is dropping from the top of the core, the axial node with the peak power remains at the top (node 21 where node 24 is at the top of the core).

Although core-average thermal-hydraulic parameters do not change significantly, the localized values change dramatically. The coolant temperature rises to saturation and then boiling begins in the bundles surrounding the dropped rod. This is primarily the result of direct energy deposition; although after approximately one second, heat transfer across the cladding also becomes important. At the incipience of boiling, RAMONA-4B predicted flow oscillations and reversal in the hot channels. This in turn led to critical heat flux in a number of channels.

Boiling introduces negative reactivity and, therefore, could be important in mitigating the total enthalpy increase. In other situations, with little or no subcooling, boiling could begin very soon into the transient and reduce the power excursion and the immediate enthalpy increase. In these situations, there is a burden on the accuracy of the thermal-hydraulics model being used. Although it is clear that a certain amount of energy deposition in the coolant leads to boiling, the timing could be important, and current void generation models are based on experiments that do not mimic the dynamic conditions found during a RDA.

The results of most interest in this study are for fuel enthalpy (defined as the pellet radial average at any location in the core) as that is the parameter which is currently used to determine the acceptance limit for the RDA (280 cal/g in the U.S.) and the condition for fuel failure (170 cal/g in the U.S. for BWRs at low or zero power) for the purpose of calculating the radiological response. In the past, only the peak fuel enthalpy throughout the core has been of interest in licensing calculations, i.e., the maximum in both space and time. However, in the present study, it was of interest to understand the peak during the event as a function of the burnup of the fuel and that requires knowing the peak enthalpy in all the nodes in the region around the dropped rod. In the following, bundle enthalpy is considered recognizing that if the results could be translated to an individual rod within a bundle, the fuel enthalpy would be higher. In order to know how much higher, one would have to do detailed calculations for the region surrounding the dropped rod. The hottest rod in a steady state might have a power 10-15 percent above the bundle average, but in the transient situation, the peaking could be quite higher.

Figure 4 shows the maximum fuel bundle enthalpy in three neutronic channels (fuel bundles) as a function of time. The maximum in time occurs in Channel 27, which is one of the bundles directly adjacent to the dropped control rod (CR #14 in Figure 4.1). Channel 56 is diagonally adjacent to Channel 27, and Channel 89 is one pitch removed from Channel 27. The predicted enthalpies are for an interval of 15.9 cm (6.3 in) corresponding to axial node 21which is the node with maximum fission power. The legend shows the bundle burnup at the node in the bundle for which the enthalpy is a maximum. Reactor power history is also shown on the figure.

There are three distinct phases in the enthalpy plots: (1) prompt heatup due to the fission power excursion, (2) continuing fission power heatup, and (3) shutdown cool-off. Observation of the enthalpy curves indicates that in this particular calculation, the amount of prompt heatup is roughly equal to the fission power heatup. This results from the initial conditions, mainly from the high initial moderator subcooling which delays bulk boiling in the core--an important factor responsible for shutting down the fission reaction by introducing large negative void reactivity. A lower initial coolant subcooling would result in a lower maximum fuel enthalpy reached during the accident. Note that the separation of the fuel enthalpy increase into the first two phases may become particularly important if studies of fuel behavior in the future lead to acceptance criteria that are based on both the initial fuel enthalpy rise and the ultimate value.

The peak fuel enthalpy for this event (see Figure 4) is less than 70 cal/g which is considerably below the current values of interest from a licensing point of view. However, for this study, it was of interest to consider the fuel enthalpy as a function of burnup for a given RDA. Figure 5 shows enthalpy versus burnup not only for the three bundles used to generate Figure 4 but rather for all 16 bundles (identified by number on the graph) of most interest surrounding the position of the dropped rod. The figure shows the orientation of these bundles relative to the dropped rod position of CR #14 which is between bundles 27 and 28. The crosses indicate control rods initially inserted.

These results do not indicate a simple correlation between fuel enthalpy and burnup. Rather, they suggest that for the given rod worth, the peak fuel enthalpy in a bundle is a complex function of factors, such as the distance of the bundle from the dropped rod and the burnup of the fuel. In other cases for different control rod worths, the enthalpy in a given bundle could be higher or lower depending on the specific circumstances.

This conclusion is probably valid in spite of the fact that there are several other factors influencing Figure 5--namely, that (1) bundles 30 and 60 are on the core periphery and, therefore, the power surge is mitigated by the neutron leakage into the reflector and (2) the bundles with burnups of about 5 GWd/t have reactivities impacted by the burnout of gadolinium and, therefore, cannot be expected to have the same burnup dependence as bundles with higher burnups where gadolinium is no longer an important factor.

Results for a Pseudo High Burnup Core

The psuedo high burnup core model was used to calculate the effect of dropping CR #14 from a control rod pattern corresponding to 50 percent control rod density. The power versus time is shown in Figure 6 on a logarithmic scale. The behavior is similar to that for the medium burnup case except that the peak power is higher. Although the fuel has a higher burnup in this case, the reactivity is not necessarily lower. More reactivity is designed into the fuel so that the reactor can continue to produce power at the higher burnup. Therefore, it is not surprising that results for the two burnup cases are similar.

The results for maximum fuel enthalpy versus burnup are shown in Figure 7 for the bundles surrounding the position of the dropped rod. Again, there is no clear correlation between burnup and enthalpy, and the conclusions discussed above seem to apply here as well, i.e., that the enthalpy in any node depends on control rod worth, distance from the rod and also on burnup. In this figure and in Figure 5 for the medium burnup core, only the axial node with the peak enthalpy has been considered for a given bundle. Since the bundle burnup will be higher at nodes that are closer to the center of the core, if these additional nodes were added to the plot, they would show points at higher burnup and lower fuel enthalpy relative to each of the points on the present plot. This would tend to create more points on the graph to the right and down from existing points. However, the nodes further away from the center (e.g., nodes 23 and 24) would have lower enthalpy and lower burnup adding points to the left and down from the existing points. These additional axial points would, therefore, not be expected to reveal any trends and would not negate the possibility of relatively high enthalpy in a high burnup node if it were close to a high worth dropped control rod.

Sources of Uncertainty in RDA Analysis

There are two general sources of uncertainty: (1) the methodology and (2) the assumptions used to define the reactor state. The methodology consists of the computer models and the values of the neutronic and thermal-hydraulic parameters that are used in those models. The validation of computer codes for application to the rod drop accident has always been a difficult matter. Since there have never been any rod drop accidents in a BWR, no data exists to directly assess the uncertainty in the calculated fuel enthalpy during a rod drop accident. Instead, the approach in the past has been to generally validate the computer codes and then to use a conservative approach to determine the margin to the acceptance limits for the rod drop accident. The conservative approach biases the assumptions used to define the reactor state so that the calculated peak fuel enthalpy is maximized.

Although this has been an adequate practice in the past, it will be important in the future to provide an estimate of uncertainty if either the margin between expected fuel failure and calculated fuel enthalpy becomes much smaller than is currently the case or if calculations are done using a best-estimate approach rather than a conservative approach. It will then be important to know the sources of important uncertainties within the models and what impact these have on the uncertainty in fuel enthalpy in a given bundle.

One source of uncertainty is due to the "rim" effect in high burnup fuel. In general, the power distribution through a pellet is peaked at the surface due to self-shielding. This causes the plutonium concentration to grow at the surface. This effect accelerates with time so that the power and the plutonium distributions become highly peaked in a small region at the rim [see, for example, (Lassmann, 1994)]. Reactor physics models that generate cross sections make assumptions about the temperature and power distributions across the pellet. With the rim effect, these assumptions may not be as valid and the uncertainty in results may increase. In addition, the uncertainty may increase due to the need to include more actinides in the models.

The change in composition with burnup influences the thermal properties of the pellet. The rim effect introduces a spatial distribution of properties that may become important. Furthermore, the uncertainty in calculations may increase with heat conduction models that do not account for the peaked spatial distribution of energy deposition in the pellet.

Two physics properties that may become more important with high burnup are the effect of the delayed neutron fraction (β) and moderator temperature feedback. The power excursion during an RDA is made worse when the delayed neutron fraction becomes smaller. The delayed neutron fraction decreases with burnup, and the ideal model would allow for the spatial distribution of β to account for the burnup throughout the core.

The moderator temperature feedback is positive when the moderator is relatively cold. The effect is made worse if there is significant subcooling. Again, the effect becomes stronger with burnup, i.e., it is more important to model the effect for high burnup cores. This effect was somewhat quantified by redoing the calculation of the RDA for the medium burnup core with no moderator temperature feedback. The elimination of moderator temperature feedback had no significant effect on the initial power pulse and fuel enthalpy increase, but it did decrease the maximum enthalpy by approximately 5 cal/g. Since the moderator temperature reactivity coefficient is linear in the burnup range from 22 GWd/t (the burnup of the node with maximum enthalpy) to a high burnup value of 66 GWd/t, it is reasonable to expect that the effect may be on the order of three times as large or 15 cal/g for high burnup fuel.

Several other sources of uncertainty have been discussed above in the context of the assumptions used to model the initial conditions. The assumed initial control rod pattern (and the core design) determines the rod worth. The assumed initial thermal-hydraulic conditions determine the moderator feedback and the timing of negative void feedback. In a best-estimate calculation, it is necessary to take into account the withdrawal sequence being used at a particular plant, the possibility of an error in that withdrawal, and the corresponding thermal-hydraulic conditions. It is difficult to find a single worst initial condition because the highest worth rod may not lead to the most limiting fuel enthalpy if the acceptance criterion is based on burnup. Nevertheless, it should be possible to identify the leading contenders for worst initial condition so that only a few of the hundreds of theoretically possible accident situations would have to be calculated.

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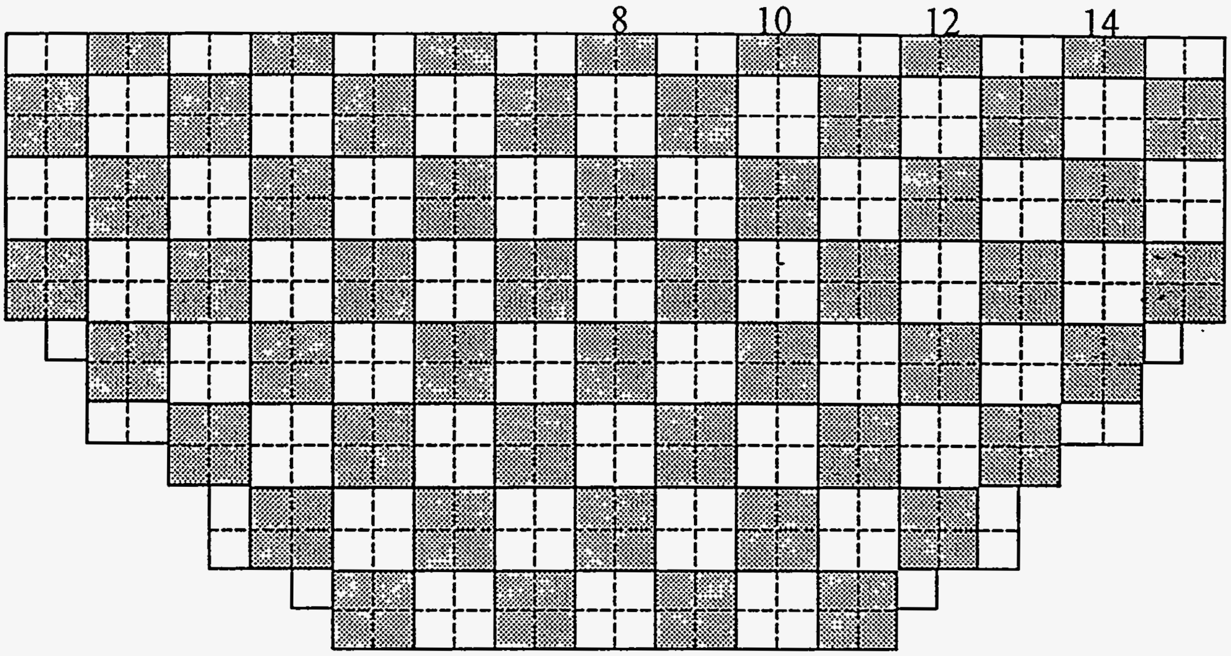

Unshaded core cells: control blade withdrawn
Shaded core cells: control blade inserted

Figure 1 Initial Control Rod Pattern for RDA Analysis

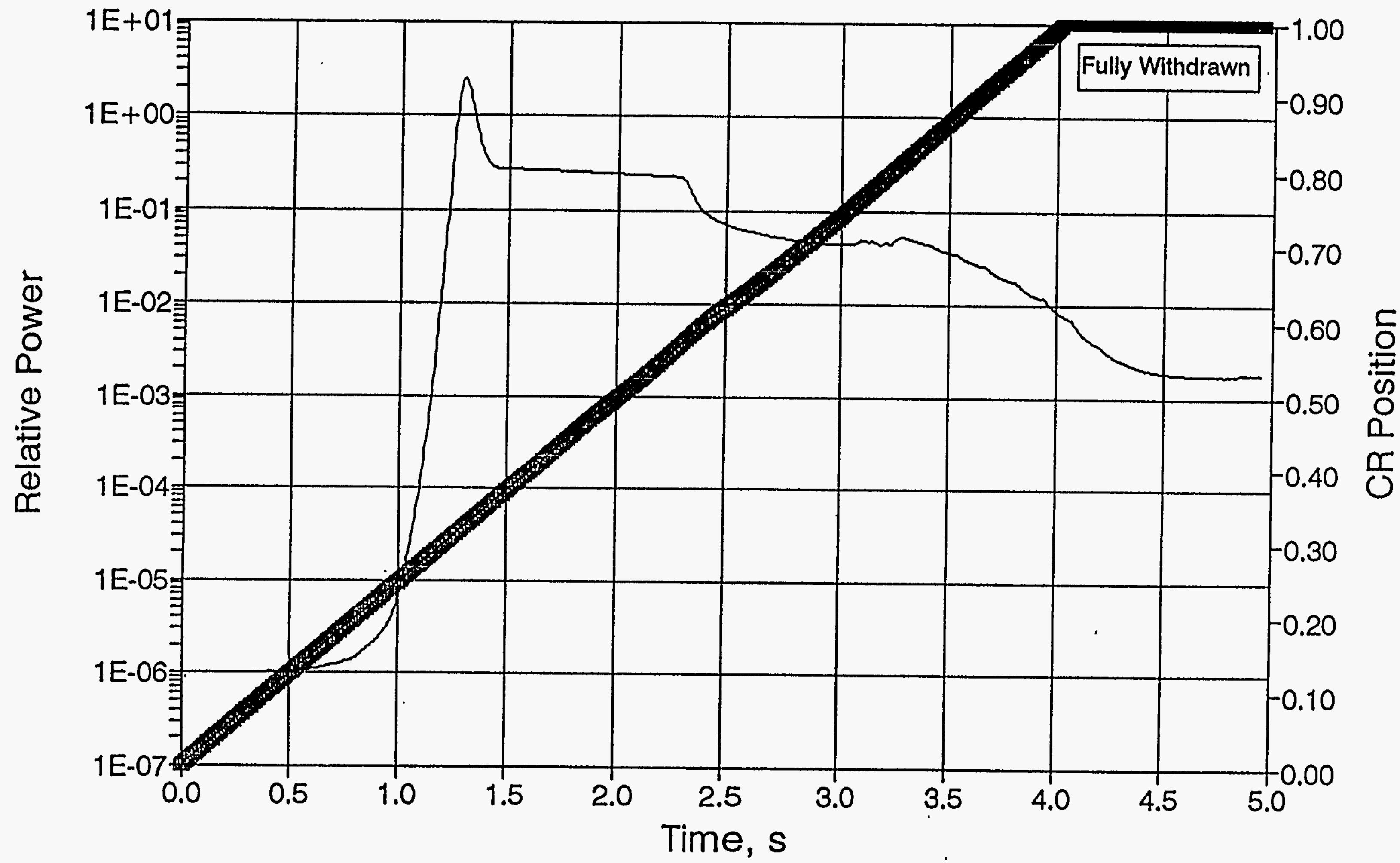

Figure 2 Power and Rod Position During RDA (Medium Burnup Case)

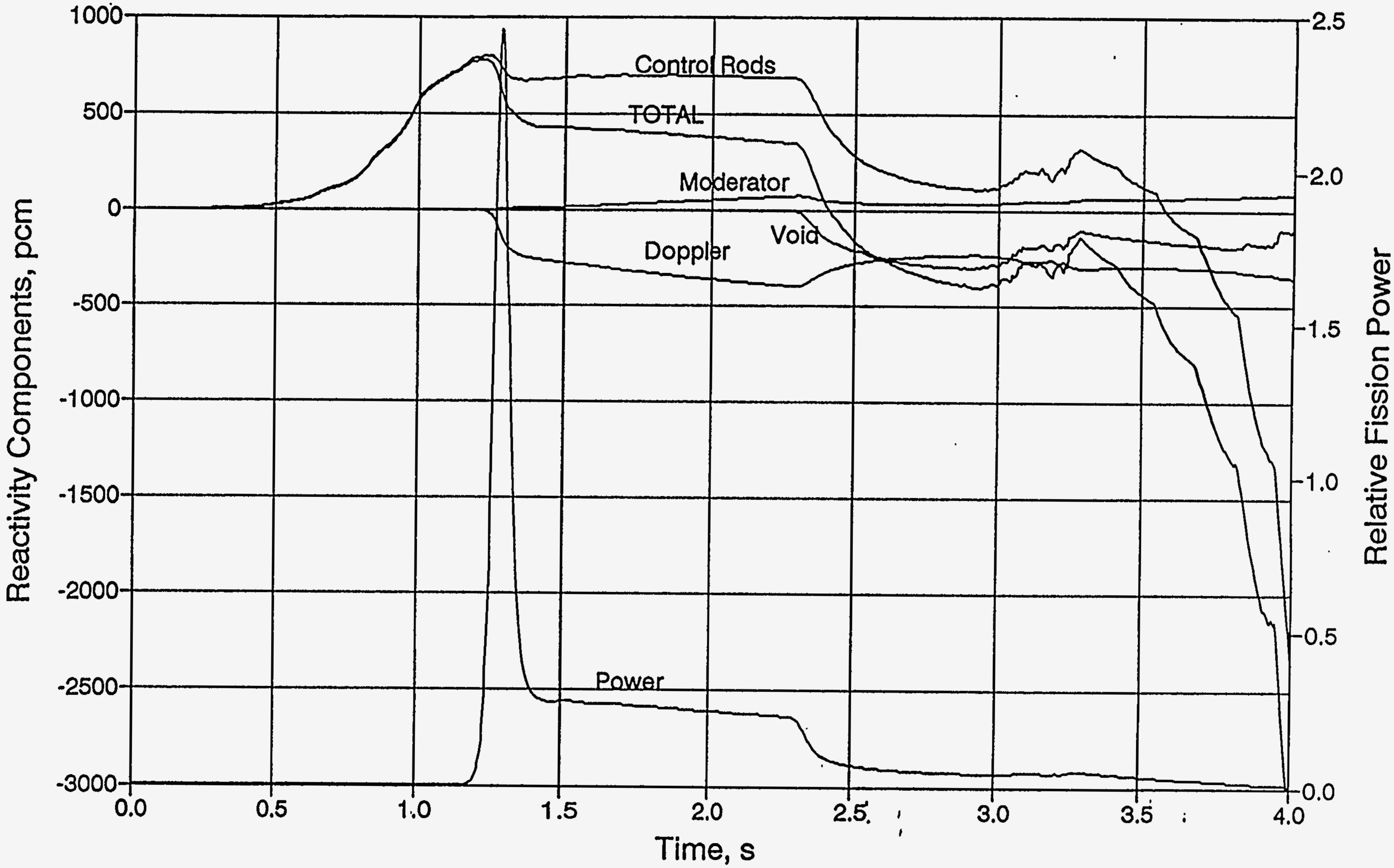

Figure 3 Reactivity Components During RDA (Medium Burnup Case)

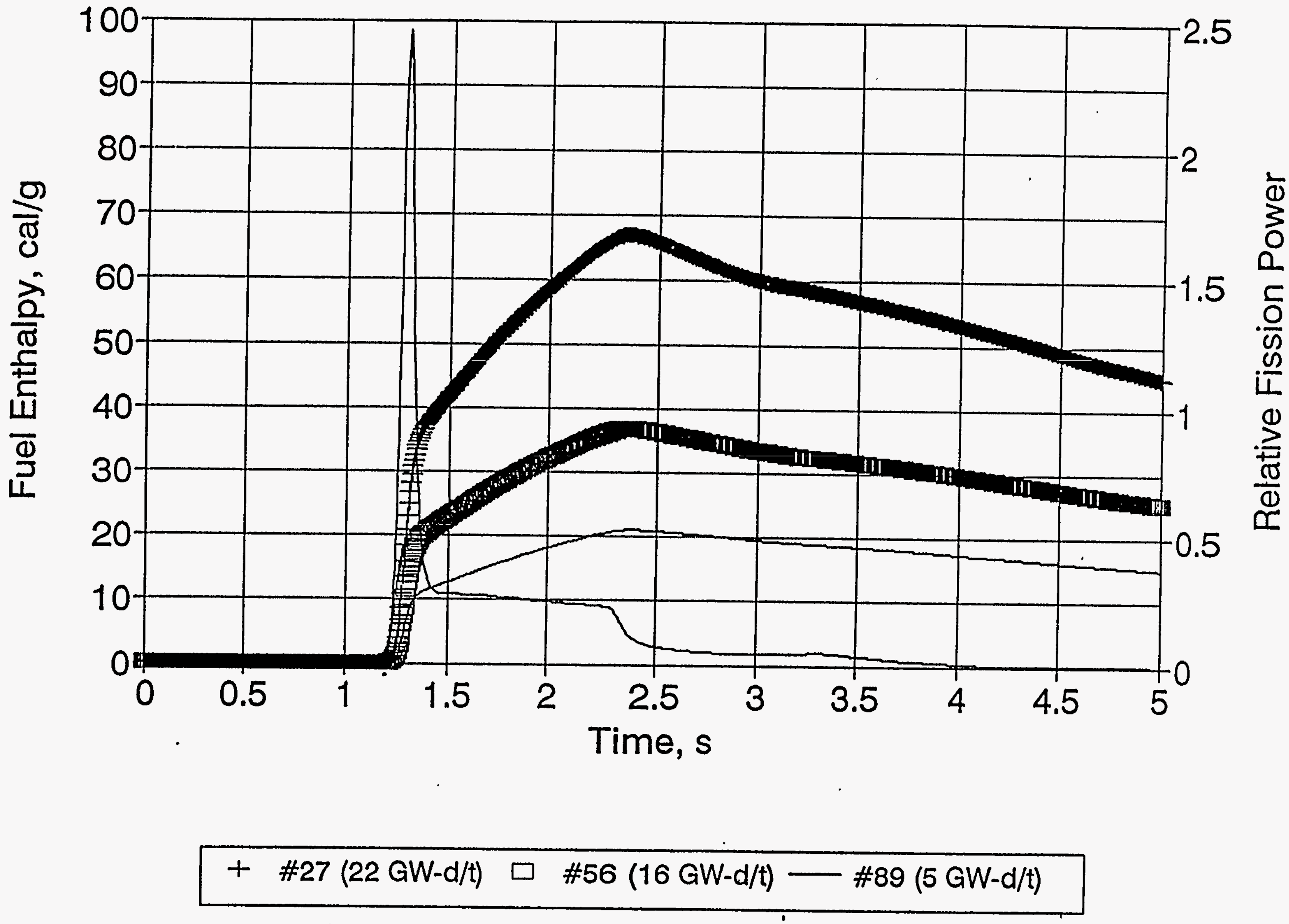

Figure 4 Maximum Fuel Bundle Enthalpy and Total Power During RDA
(Medium Burnup Case)

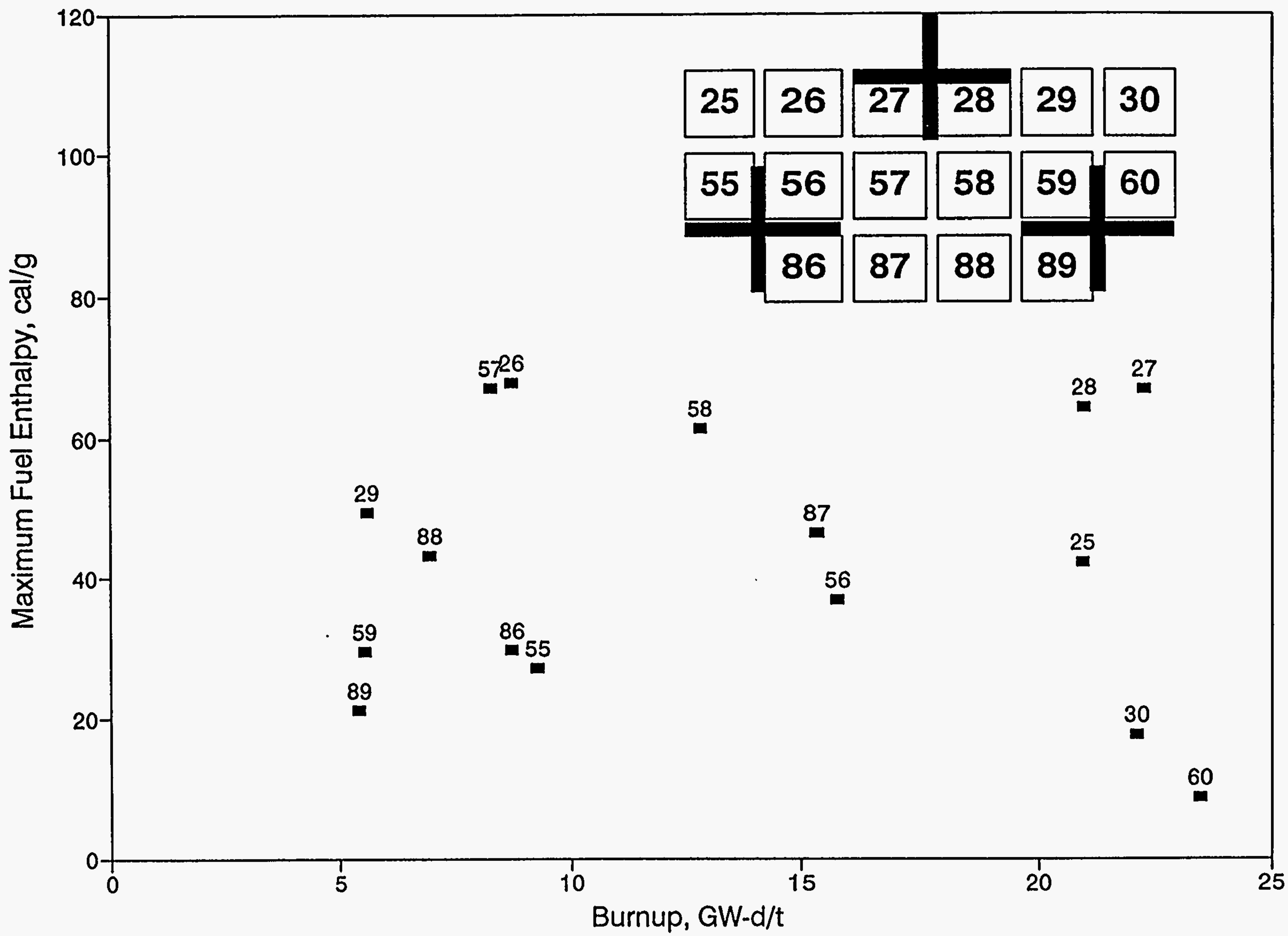

Figure 5 Maximum Fuel Bundle Enthalpy vs. Burnup (Medium Burnup Case)

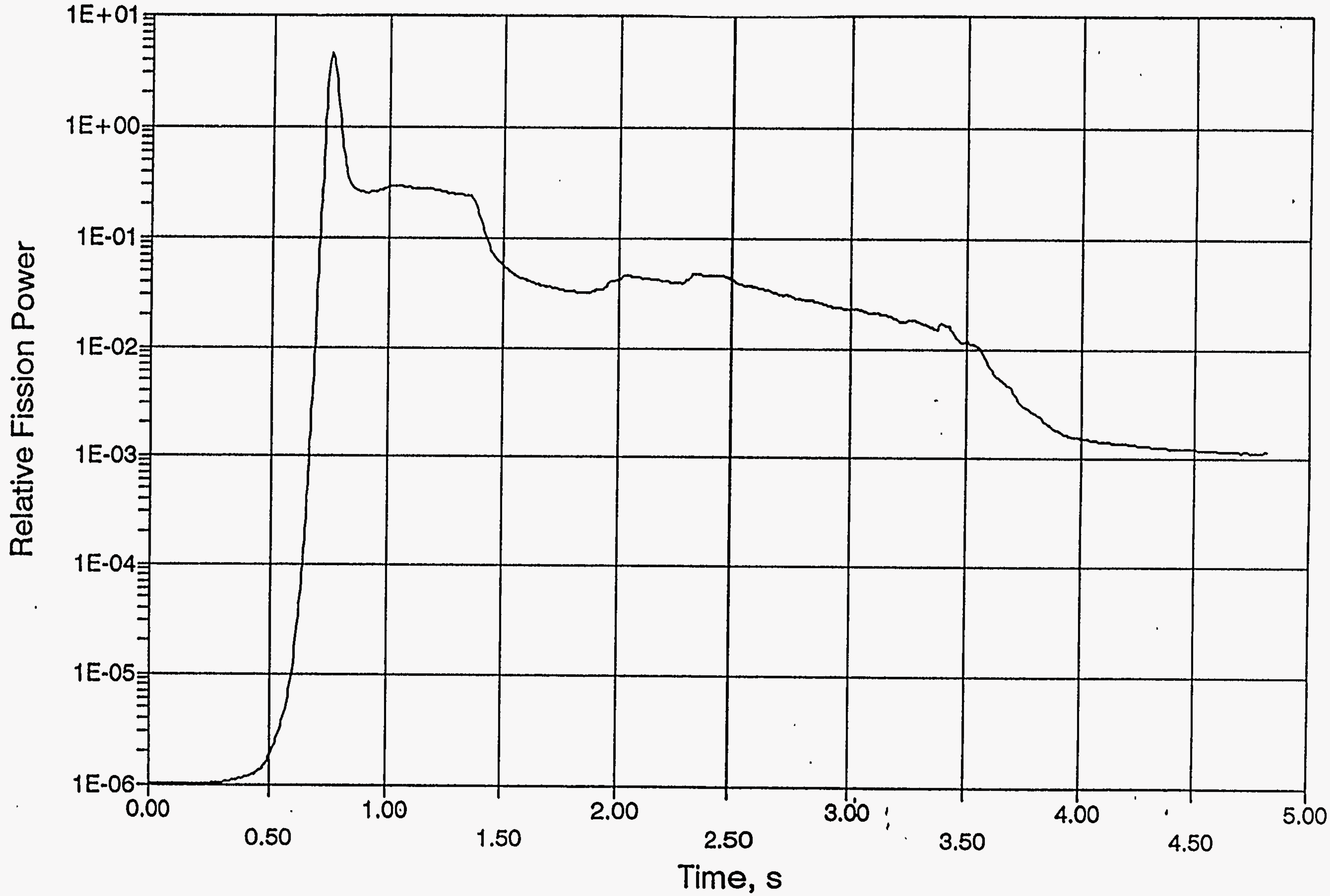

Figure 6 Power During RDA (Pseudo High Burnup Case)

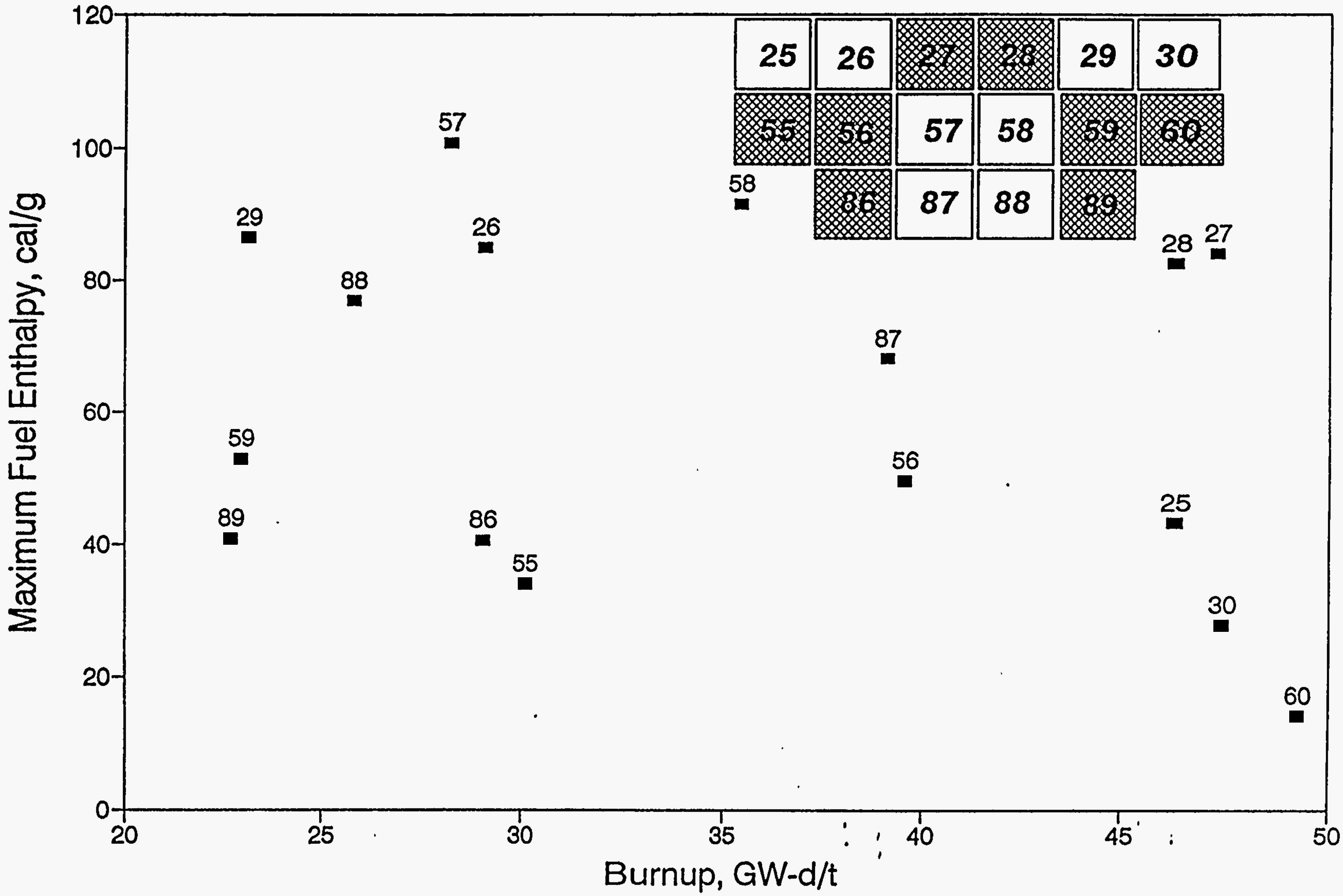

Figure 7 Maximum Fuel Bundle Enthalpy vs. Burnup (Pseudo High Burnup Case)