CONF-970315--1-Rev.

BNL-NUREG-63663
(REV. 2/97)

ANALYZING THE ROD DROP ACCIDENT IN A BWR WITH HIGH BURNUP FUEL*

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ABSTRACT

The response of fuel in a boiling water reactor to the rod drop accident (RDA) was studied using the RAMONA-4B computer code. Calculations of this design-basis event by licensees have traditionally been done conservatively because there was margin to the fuel failure criterion of 170 cal/g. However, because high burnup fuel may fail at much lower fuel enthalpies, it is of interest to understand what might be the best-estimate of the enthalpy as well as the corresponding uncertainty. This study has several parts. In one part of the study calculations were done to assess the sensitivity to *reactor conditions* such as control rod pattern, inlet subcooling, and fuel burnup. It was shown that the fuel enthalpy at any location in the region surrounding the dropped rod depends on the rod worth, the distance from the dropped rod, and the burnup of the fuel. Another part of the study included calculations to assess the sensitivity to fundamental parameters whose *modeling* introduces significant uncertainty which may increase with burnup. These parameters are the control rod worth, Doppler reactivity coefficient, delayed neutron precursor fraction, and fuel specific heat. The results of these sensitivity studies were then used in a simple model to determine the random uncertainty in the fuel enthalpy. By using engineering judgement for the uncertainty in the fundamental parameters, the standard deviation for the calculated fuel enthalpy in this study was estimated to be 37%. Based on this analysis, the limiting bundle fuel enthalpy might be 75% (2σ) higher than calculated. The last part of this study was to investigate one source of systematic error; namely, the effect of the fuel rod enthalpy distribution within a bundle. RAMONA-4B calculates the fuel bundle average enthalpy and estimates must be made of a bundle peaking factor to determine the fuel rod enthalpy. In this study a fit of RAMONA-4B bundle powers was used to estimate the local power peaking. It was determined that the peaking factor could be 25% higher than the factor usually assumed for RDA analysis. Combining this error with the 2σ random error means that for this analysis the actual fuel rod enthalpy could be 100% larger than calculated by RAMONA-4B. This is much larger than the uncertainty in most parameters that are calculated with best-estimate methods for other design-basis events.

I. INTRODUCTION

The design-basis reactivity accident for a boiling water reactor (BWR) is the rod drop accident (RDA). The acceptance criterion for licensing is that the peak fuel enthalpy (pellet average at any axial location) be below 280 cal/g. For events starting from low power (the usual case), the criterion for determining fuel failure for the purpose of assessing radiological consequences is 170 cal/g. Conservative licensing calculations in the past have indicated margin to these criteria. However, recent experiments with high burnup fuel suggest that it may be necessary to consider fuel enthalpies as low as 30 cal/g, or increases in enthalpy of only 15 cal/g, when evaluating fuel failure. This has precipitated interest in examining the RDA from a best-estimate point of view as conservative calculations give higher fuel enthalpies for high burnup fuel.

This paper addresses one of several studies being supported by the U.S. Nuclear Regulatory Commission to understand high burnup fuel behavior. One objective of this work was to analyze the response of a BWR to the RDA and to understand the sensitivity to reactor conditions,

*Work performed under the auspices of the U.S. Nuclear Regulatory Commission.

including the presence of high burnup fuel. Another objective was to investigate the sources of uncertainty in the RDA analysis and to perform sensitivity calculations to help understand the magnitude of the uncertainty.

In Section II the response of fuel to an RDA is discussed as a function of different reactor conditions; namely, control rod pattern, core inlet subcooling, and burnup. In Section III the sources of uncertainty in modeling the RDA are discussed and an estimate is made of the uncertainty in the calculated fuel rod enthalpy. More details are found in References 1 and 2.

II. SENSITIVITY OF THE RDA TO REACTOR CONDITIONS

A. Base Case

The calculations of the RDA were done with the RAMONA-4B code and a model for a BWR/4 with a core having bundle burnups up to 30 GWd/t. The RDA calculations were also repeated for a pseudo high burnup core with bundle burnups up to 60 GWd/t. The latter model was generated as an approximation to a high burnup core. These models are explained in detail in Reference 2.

The initial control rod pattern was a checkerboard with half the control rods fully inserted. Figure 1 shows, for an RDA, the core power and fuel bundle enthalpy, at the axial position of the maximum, for three bundles. The locations of the three bundles are immediately adjacent to the dropped rod (#27), one row away (#58), and two rows away (#87). Also indicated on the figure are the burnups at the location corresponding to the enthalpy being plotted. The speed of the dropped rod is assumed to be 0.94 m/s (3.1 ft/s) and, hence, it can be seen from the results that the excursion is over before the rod has moved more than half-way out of the core.

B. Effect of Control Rod Pattern

The most important initial core condition is the control rod pattern as that is the principal determinant of control rod worth (for a given core design). Control rod worth is the driving force that causes the power excursion and many analysts have demonstrated that the peak fuel enthalpy is strongly dependent on the dropped rod worth.[3,4,5,6,7,8] The control rod pattern is determined by the particular withdrawal sequence being used (e.g., the banked position withdrawal sequence[9]) and by the positions of any out-of-service rods. In order to consider a particular rod as the dropped rod, its drive mechanism must have moved at least halfway out of the core. Consideration must also be given to a single failure (human or machine) during the withdrawal sequence if the event is to be calculated as a design-basis accident. Since there are many patterns that could lead to a dropped rod worth sufficient to cause significant fuel enthalpy increases in the core, many calculations must be done to get to the most limiting consequences.

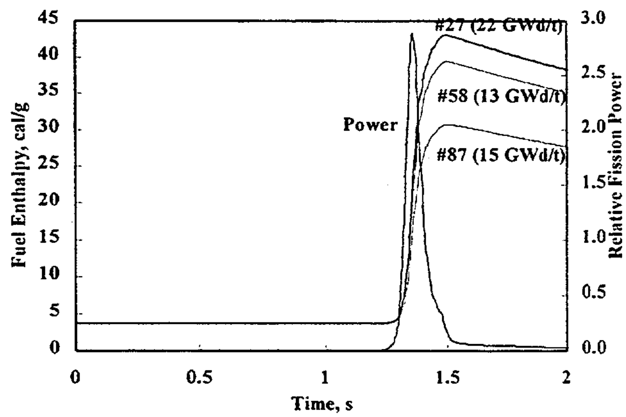

Figure 1 Maximum Fuel Bundle Enthalpy and Total Power During RDA (Low Subcooling)

Rather than consider the effect on the RDA of dozens of patterns that might be possible, the sensitivity to dropped rod worth for a single pattern was calculated. The rod worth was artificially modified from its base case static worth of approximately 950 pcm. One case with lower worth and two cases with higher worth were calculated. The results show a linear dependence between inserted rod worth and peak fuel enthalpy for the bundle adjacent to the dropped rod. Figure 2 shows these results modified to show the magnitude of the relative change (%) in peak fuel enthalpy from the base case vs the magnitude of the relative change (%) in inserted control rod reactivity (another curve on the figure will be discussed below). A straight line fit goes through the points. Note that the relative change in rod worth refers to the worth inserted during the excursion rather than the total static worth of a completely withdrawn control rod. The inserted worth was $1.32 for the base case.

C. Effect of Core Inlet Subcooling

The initial core inlet subcooling, is determined by the withdrawal sequence. Power and coolant temperature begin to rise as more rods are withdrawn. Pressure also increases, increasing the saturation temperature and operators allow the subcooling to decrease. A low initial subcooling leads to void generation soon after the initial power pulse and this limits the fuel enthalpy rise after the pulse (the second phase of the transient) as can be seen in Figure 1. Hence, a given control rod worth will result in

a higher fuel enthalpy when it occurs early during the withdrawal sequence when the subcooling is highest.

This effect was quantified by an RDA calculation with the inlet subcooling changed from the base case value of 11°C to 70°C. In going to the higher subcooling the peak fuel enthalpy increased from ~45 cal/g to ~70 cal/g. This is a result of a decrease in void generation; an effect that has been known previously[3] but can now be quantified more accurately. The increase in fuel enthalpy during the second phase of the transient accounts for approximately 45% of the total increase in enthalpy and hence, it is important to assure that the conditions used for calculations are consistent with operational procedures for a particular plant.

Although it is clear that the effect of subcooling is important, the quantification of the effect has a large uncertainty. The void generation, which is the reason that the power is shutdown when the subcooling is low, is calculated with a model in RAMONA-4B that has not been validated under the conditions of an RDA. Indeed, nor have any other such models used in codes for RDA analysis.

D. Effect of Fuel Burnup

Only the peak fuel enthalpy throughout the core has traditionally 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 RDA 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.

When the high subcooling calculations were repeated for the pseudo high burnup core the control rod worth increased and the peak fuel bundle enthalpy went from ~70 cal/g to ~100 cal/g. More importantly, the distribution of peak fuel bundle enthalpy remained similar to the case with medium burnup. Some information on the distribution of fuel enthalpy can be gleaned from Figure 1 which shows the results for three bundles, although a more thorough analysis would show all peak fuel enthalpies as a function of fuel burnup (e.g., see References 2 and 5).

The results for both the medium and pseudo high burnup cases 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 depends on both the distance from the dropped rod and burnup. Equivalently, for a given burnup, the fuel enthalpy will depend on the dropped rod worth and the distance from the dropped rod. An implication of this is that high burnup in a fuel bundle does not inherently limit the enthalpy rise.

III. UNCERTAINTY ANALYSIS

A. The Approach

The initial power pulse during the RDA is the result of a simple reactivity balance. Reactivity is added because of the worth of the dropped rod and then reactivity is subtracted because of the fuel temperature (primarily Doppler) feedback. There are secondary effects, but these generally become important after the initial pulse (in the second phase of the transient). The calculation of the control rod worth and fuel temperature feedback is based on reactor physics models that increase in uncertainty as the burnup increases. One reason for this is that the conditions within the pellet begin to change more dramatically as burnup increases into the upper allowable (by licensing) range. One contributor to this growing uncertainty is the rim effect. This is the buildup of Pu-239 at the edge of the pellet which in turn increases the fission density at the pellet edge.[10] The increase in neutron flux at the edge then increases the production of Pu-239 and the feedback continues so that pellet radial power distributions become even more peaked. Physics models make assumptions about the temperature and nuclide concentrations in the pellet and these must be consistent with the rim effect. Furthermore, there is less validation of models for high burnup fuel just because there is less data at high burnups.

The parameters in the space-dependent neutron kinetics model within RAMONA-4B which determine the control rod and fuel temperature reactivities are the coefficients of the polynomial expressions for the cross sections. In a typical RAMONA-4B reactor model (e.g., the one used for this study) there are more than 3000 numbers defining the cross sections throughout the core, and a large fraction of these determine the reactivity changes during the RDA. Since it is impractical to calculate the sensitivity to these coefficients, and it is more physically meaningful to talk in terms of reactivity components, this study considered the effect of the uncertainty in the more simplistic, but nevertheless relevant, parameters, control rod worth and Doppler reactivity coefficient.

The control rod and fuel temperature reactivities in RAMONA-4B or in any neutron kinetics model are determined not only by the absolute reactivities but also by the delayed neutron precursor fraction, β, as this parameter enters into the basic equations as a divisor of the reactivities. If the sensitivity to reactivity in absolute units is investigated so must also the sensitivity to β be investigated.

One additional parameter which must be considered is the specific heat of the fuel, C. It is what converts integrated power, or energy, to temperature and the Doppler coefficient expresses the change in reactivity per unit change in fuel temperature. Hence, the effect of uncertainties in C on the fuel enthalpy need to be considered.

The approach in the following subsections is to first consider the sensitivity of the fuel enthalpy to these parameters. Sensitivity is the change in enthalpy for a given change in the parameter. The uncertainty in fuel enthalpy as a result of the uncertainty in the parameter can then be calculated if the latter is known. This stage in uncertainty analysis is frequently difficult for transient calculations and is for the RDA because of the lack of information on the uncertainty in control rod worth, Doppler reactivity coefficient, etc. Nevertheless, it will be shown below that estimates can be made so that the uncertainty in fuel enthalpy can be obtained for a single parameter and for all important parameters combined.

B. Sensitivity Calculations

The sensitivity of fuel enthalpy calculated by RAMONA-4B to control rod worth (ρ_0) is straightforward and has been discussed above in the context of the importance of the initial control rod pattern (see Figure 2).

For the Doppler coefficient (α) the same approach is not possible as the Doppler coefficient is not calculated by the code but rather it is the Doppler reactivity which is calculated. To extract a change in Doppler coefficient for an arbitrary change in cross section one would have to separate out the change in fuel temperature and this is a more complicated analysis that has not been attempted in this study.

The sensitivity to the delayed neutron precursor fraction is shown in Figure 2. The plot shows the relative change in fuel enthalpy (absolute value) versus the relative change in β where the base case is $\beta = 0.006$ and the two parametric cases are for $\beta = 0.005$ and $\beta = 0.0065$. Also shown on the graph is a straight line representing a proportionality with a coefficient of unity.

Sensitivity calculations were also done for the specific heat, C. Two cases were carried out with C increased by 10% and 20%. The changes in fuel enthalpy were insignificant and hence, the conclusion is that fuel specific heat is not an important parameter.

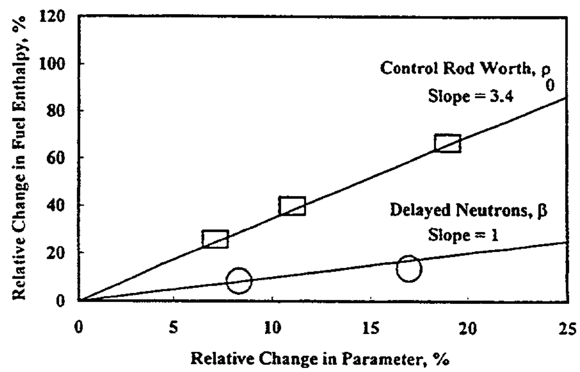

Figure 2 Relative Change in Fuel Enthalpy vs. Relative Change in Parameters

C. Uncertainty Analysis

Once the sensitivity to different parameters is determined the next step is to use this sensitivity to determine the uncertainty in the calculated fuel enthalpy. This requires knowing the uncertainty or error distribution for the key parameters. In lieu of knowing these basic uncertainties, engineering judgements can be used in order to estimate the uncertainty in fuel enthalpy. This is one of the incentives for using the global (point kinetics) parameters such as control rod worth and Doppler coefficient; analysts are familiar with these quantities and can offer estimates of their uncertainty.

The uncertainty in fuel bundle enthalpy at any location can be assumed to be modeled as:

$$\sigma_{h_f} = \left[\gamma^2\sigma_{\rho_0}^{\ 2} + \sigma_\beta^{\ 2} + \sigma_\alpha^{\ 2}\right]^{1/2} \qquad (1)$$

where each of the basic uncertainties refers to a relative error and it is assumed that this is equivalent to the standard deviation of the error distribution. Equation 1 is the result of noting that the sensitivity of peak fuel bundle enthalpy to ρ_0 and β, is approximately linear (see Figure 2) and for the rod worth the sensitivity coefficient $\gamma = 3.4$. The same information is not available for the Doppler coefficient. However, the assumption is made that the same linearity is true for α with a proportionality constant of unity. Also it is assumed that there is no correlation between the individual effects, and the uncertainties can be combined statistically. The sensitivity calculations above showed that it is not necessary to consider the effect of uncertainty in pellet specific heat.

With Equation 1 and engineering experience with the basic uncertainties, an estimate can be made of the uncertainty in fuel enthalpy. Figure 3 shows the result of a parametric study of the fuel enthalpy with ranges for the uncertainty in dropped rod worth and Doppler coefficient and with the uncertainty for β assumed to be 5%. A best estimate point, also shown on the graph, is obtained with the following reasoning.

The uncertainty in the calculated rod worth for a BWR is not easy to assess because measurements of rod worth are not routinely made as they are in pressurized water reactors (PWRs). The uncertainty in PWR rod worth can be estimated for a given reactor using measurements made during zero and low power physics tests. Rod bank worths are always measured[3] and in some instances measurements of individual rod worths are performed. Large perturbations such as entire banks can be calculated easier than individual rods. In general, calculations are in agreement with measured bank worths in the range of 0-10%. Anything over 15% would be unacceptable.[11] For BWRs at zero power conditions the accuracy should be similar. However, since it is an individual rod moving only approximately half-way out of the core that is of interest, it is expected that the uncertainty in a dynamic calculation may be at least 10%.

Trying to follow the same approach with the Doppler coefficient is problematic. There are no measurements of the Doppler coefficient (even in PWRs) making it more difficult to get an estimate of the uncertainty in the coefficient (although still easier than getting the uncertainty in cross-section coefficients). Another approach is to use a lattice physics calculation of a fuel bundle with different assumptions about uncertainties in lattice conditions (e.g., with different models accounting for the rim effect). The effect on the Doppler coefficient for the bundle with these changes could be used to infer an uncertainty. This was not done for this study but rather an estimate of the uncertainty was made using engineering judgment. That estimate is that the uncertainty is at least 15%.

The error in the delayed neutron precursor fraction can also be estimated using engineering judgement. The uncertainty in this parameter is based not only on the measurement error for β for different fissionable nuclides, but also the uncertainties in the concentrations of these nuclides--uncertainties which increase with increasing burnup. It is also dependent on how the weighting is done to either obtain bundle average values or a core average value. For the purposes herein the uncertainty will be assumed to be at least 5%.

If Equation 1 is used with these engineering judgements as to the basic uncertainties, the result is an uncertainty (one standard deviation) in the fuel enthalpy of 37% (see Figure 3). This is meant to be the random component and is based on engineering judgement. This judgement could be further refined by expert elicitation but it is probably more important to look for systematic errors which could also affect the calculated fuel enthalpy.

Ideally one should search for all possible systematic errors as some may show that the calculation is non-conservative but others may show that the calculation is conservative. No attempt was made to do this in the present study. However, one systematic error resulting in non-conservative results is quite obvious and could increase the uncertainty in the results considerably. It is due to the fact that most neutron kinetics codes used for RDA analysis calculate the bundle fuel enthalpy rather than the fuel rod enthalpy and it is the latter which is the result of interest. Indeed, the sensitivity calculations with RAMONA-4B given herein are for the calculation of fuel bundle enthalpy.

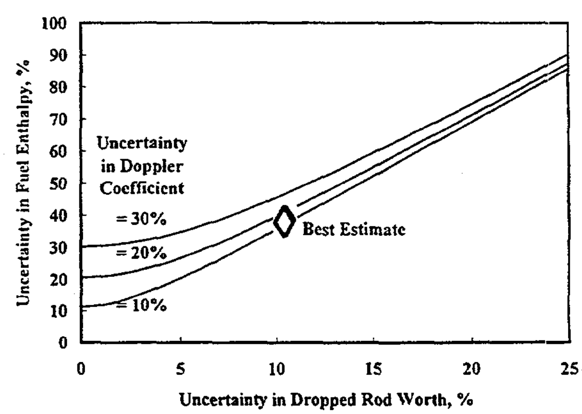

Figure 3 Effect of Basic Uncertainties on Fuel Enthalpy Uncertainty (Random Component)

When translating from the fuel bundle enthalpy to the fuel rod enthalpy it is necessary to apply a bundle peaking factor which is the peak-to-average fuel rod power within a bundle. This peaking factor is not a constant and changes during the RDA. It has been common practice in the past to assume that this peaking factor can be obtained from a post-accident calculation, i.e., the bundle power distribution is calculated for an isolated bundle with the adjacent control rod removed representing the conditions after the event. A typical value for the peaking factor is 1.1 at a burnup of 50 GWd/t.[5] This approach, however, does not take into account the global peaking as a result of the dynamic nature of the event and this can be quite large.

This peaking can be accurately calculated with a flux reconstruction model which generates the rod-by-rod power during the event. It can also be estimated using some simple models in order to understand its magnitude. This is the approach taken in this study. To calculate the bundle peaking during an RDA, a simple fit in the radial direction was carried out. This was done by assuming that the power in a bundle can be fit with a two-dimensional (x,y) quadratic polynomial.

The unknown coefficients are obtained by assuming that the fit extends into the surrounding eight bundles in an average sense. Since the bundle average power is known (from an RDA calculation) in these nine bundles, the nine coefficients can be obtained by integrating the polynomial over each bundle so that nine equations result for the nine bundle-average powers. These nine equations are used to obtain the coefficients and once they are known they can be used to calculate the peak-to-average power within the bundle of interest. This approach only uses the fit in one bundle with the eight surrounding bundles providing boundary conditions and must be repeated for each bundle of interest. The model is a mathematical artifice rather than a physical model and is meant to get an estimate of the gradient within the central bundle.

Based on using this approach it was determined that the error introduced by not accounting for the change in peaking factor during the transient is 25%, i.e., the fuel rod enthalpy can be 25% higher than calculated for this study using RAMONA-4B and assuming a peaking factor of 1.1.

The maximum error in the fuel enthalpy, i.e., the number which should be added to the calculation to come up with a bounding value for the fuel rod enthalpy can be defined in different ways. One way is to add the systematic error and the bounding error due to the random uncertainty. The latter error is taken as 2σ or 75%. Combining this with the 25% overestimate due to the systematic error leads to a bounding error in the fuel rod enthalpy of approximately 100%. Other methodologies are expected to give similar results. The results for the current study can be refined with extra effort and results can be obtained for other methods. However, without those additional studies it is immediately clear that the error for the RDA is large relative to the uncertainty expected for many other variables which are calculated to compare against licensing acceptance criteria. This becomes important if BWRs submit best-estimate calculations for the RDA as part of their licensing submittals as best-estimate calculations are usually required to be known with a 95% uncertainty at a 95% confidence level.

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