# Fuel Enthalpy Behavior Evaluations in Rod Drop Accidents of BWRs

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ABSTRACT

Analyses for fuel enthalpy behavior and the fuel integrity evaluations in rod drop accidents (RDAs) for BWRs are presented. A three-dimensional simulation code was used to analyze core transient during RDAs. The code predicts a small amount of cladding failure in the cold start-up core RDA, but predicts no failure in the hot-standby core RDA. The code also predicts no mechanical energy release in either core RDA.

## 1 INTRODUCTION

In BWRs, RDA is a typical reactivity initiated accident, which is analyzed as one of design basis accidents. However the possibility of its occurrence is considerably small. In this accident, rapid and great core thermal power increase may occur, due to the movement of a control rod, and might lead to a significant increase in nuclear fuel temperature and its enthalpy. Power deposition on nuclear fuel is the severest in the vicinity of the drop control rod.

In such a rapid power burst, the degree of fuel damage can be related to fuel enthalpy rise, based on NSRR experiments. Therefore, it is possible to evaluate fuel rod integrity by analyzing its enthalpy.

## 2 ANALYSIS CODE

To evaluate RDAs appropriately, present analysis code contains three-dimensional time-dependent neutronics model and multiple channel thermal-hydraulics model, which is applied to analyzing BWR core transient behaviors.

In the neutronic model, the code solves a modified one group diffusion equation with an improved quasi-static method to reflect three dimensional effects precisely. The basic equations for neutronics are as follows,

$$\frac{1}{v}\frac{\partial \Phi}{\partial t} = \nabla \cdot D \nabla \Phi - \Sigma_r \Phi - (1-\beta)\frac{F}{k_0}\Phi + Sum\,\lambda_i C_i$$

$$\frac{\partial C_i}{\partial t} = -\lambda_i C_i + \beta_i \frac{F}{k_0}\Phi$$

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where F is a macroscopic cross section of neutron production.

In the thermal-hydraulic model, it solves a one dimensional equation set with non-equilibrium 5 or 6 conservation equations for two phase flow. The model consists of many parallel one dimensional flow channels in the axial direction, whose treatment is based on the independence of individual BWR channel flow characteristics. Flow characteristics for each channel are solved, whose pressure drops across the channels are equalized over the entire core.

The neutronic and the thermal-hydraulic models which improved from those of the ARIES code[1][2].

Fuel and cladding properties are solved in each axial location. The code solves a one dimensional heat transfer model for fuel and cladding. A fuel rod is radially discretized into fine meshes whose components can be arbitrary specified. Fuel-cladding gap conductance and heat transfer rate between cladding and coolant are also evaluated in the code.

## 3 FUEL DAMAGE BEHAVIOR IN RDAS

Based on many reactivity initiated experiments, the NSRR experiments[3],[4] and so on, the behavior of fuel damage under power bursts have been widely investigated.

If a large amount of reactivity is initiated in a reactor core, such as a rod drop accident, fuel power increases very rapidly especially in the vicinity of the uncontrolled region which emerged as a result of the control rod drop. Such a rapid power increase leads to great energy deposition, increasing fuel temperature and fuel enthalpy swiftly.

If the degree of fuel temperature surge is sufficiently large, heat flux from cladding wall to coolant becomes huge enough to induce film boiling. Film boiling raise cladding surface temperature greatly, so as to promote metal-water reaction and oxidation. These surface exothermic reactions and temperature jump deteriorate cladding mechanical intensity. If the temperature were to rise extremely, the cladding would be damaged.

If the degree of fuel enthalpy rise is larger, the fuel pellet melts instantly, before the cladding surface integrity is deteriorated by the film boiling, there will be a possibility of mechanical energy release, resulting from molten fuel burst due to excessive inner gas pressure.

It should be emphasized that there are two kinds of failure threshold to fuel enthalpy, which distinguish the degree of fuel damage. One is a threshold for fuel damage, due to the loss of cladding integrity. The other is a threshold for the generation of mechanical energy, which impacts on core structures. According to the experimental results, the threshold enthalpy for fuel damage contrasts with whether or not coolant liquid, water, is included within the cladding. If there is no water within it, applying a very conservative estimation, the threshold value for cladding damage is defined as 92 cal/g-$UO_2$ while the threshold for mechanical energy release is defined as 230 cal/g-$UO_2$. However, in case of a water-logged fuel rod, the value goes down considerably to 65 cal/g-$UO_2$. In this case, the mechanical energy releases before the cladding failure mechanism works. However, since the possibility of the existence of a water-logged fuel rod is very low, it has been analyzed that no serious mechanical effect would occur to harm core structures during an RDA.

The mechanical effect, which arises from explosive evaporation due to the interaction between coolant liquid and the burst fragments of melted fuel, can divide into two major effects. One is an impulse pressure loading to the core structures, with propagating shock wave from the rupture point. The other is a water slug motion to the reactor pressure vessel (RPV) head. It is assumed conservatively that the slug is formed with a water above rupture level, whose cross section is equivalent to that for a fuel channel.

## 4 APPLICATION RESULTS

In every reactor operating condition, the magnitude of peak power during an RDA is the severest, when core thermal power level is the lowest, such as at core start-up or in the standby mode. In the following, some results of application to typical BWR RDAs, both in cold start-up and hot standby core conditions, are presented. Figure 1 shows the core configuration and devision into thermal-hydraulic channels for the analyses against cold RDA. The reactor is assumed to be initially critical. At this time, the drop control rod is fully inserted and is positioned at the core center.

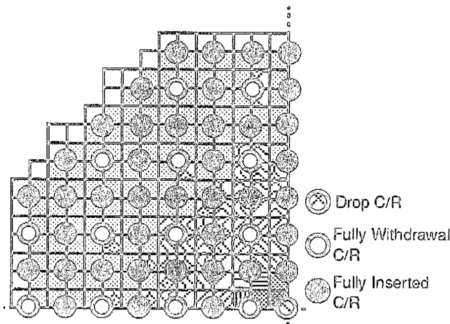



Figure 1

Core Configuration and Division into Channels (1/4 core)

The analysis conditions are determined very conservatively. Such conditions are drop control rod worth, its drop speed and initial core thermal power level, which are tabulated below. They are defined so that they lead to a severer transient core state with more reactivity initiation, faster initiation rate and less reactivity feedback. Since the mechanical effect from a water-logged fuel rod is able to be neglected, as mentioned previously, every fuel rod is treated as being a waterless rod.

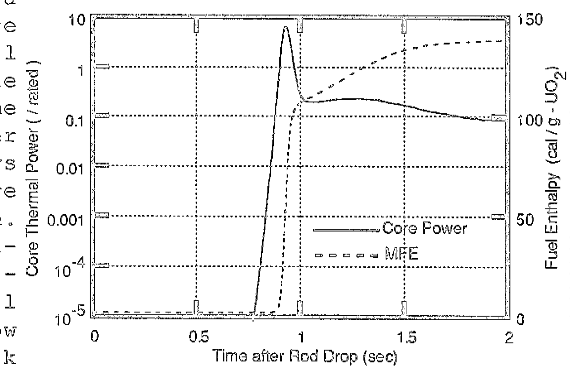
Table 1. Major Conditions for Analyses

| Items | Conditions |
|---|---|
| Initiated Reactivity(%Δk) | 1.5 |
| C/R Drop Speed (cm/s) | 95 |
| Initial Core Thermal Power (rated) | $10^{-8}$ ( cold start-up )<br>$10^{-6}$ ( hot stand-by ) |

Analyzed results are illustrated in Figs. 2 and 3 for a cold RDA and in Figs. 4 and 5 for a hot standby RDA. They demonstrate time histories about core thermal power, reactivity components and the maximum fuel enthalpy in the core.

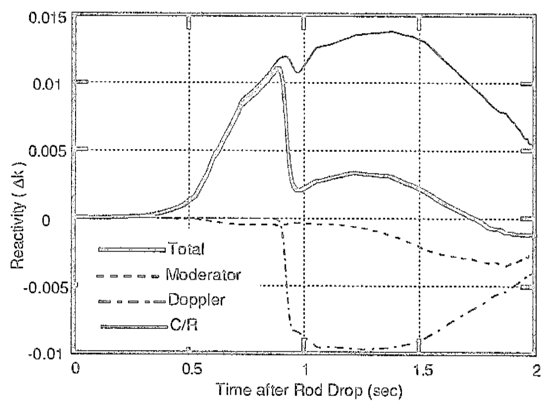
According to the analyses by the code, core power excursions are simultaneously suppressed by inherent neutronic feedback mechanisms, Doppler and moderator density feedback, for both core conditions.

In the cold start-up core condition, the increase in initiated control rod reactivity causes a dramatic rise in core thermal power and fuel enthalpy, until the maximum core power time point is reached. Doppler reactivity feedback plays a dominant role for core thermal power suppression. As a result of in-sufficient power suppression, however, fuel enthalpy continues to grow gradually to its peak level, about 140 cal/g-$UO_2$, for a half a second, then declines barely. It runs over to the cladding failure threshold, which indicates fuel damage. This is mainly because moderator density feedback is in-sufficient, due to time delay owing to large subcooling and due to its weak work, because of limited steam voids generation. However, the peak enthalpy doesn't exceed the failure threshold value for generating mechanical energy. It turns down from a level much below the threshold. Because most of the energy deposits into the core center region after the control rod started to drop, the locations of fuel rods, where enthalpy exceeds the threshold of cladding failure, are restricted within the vicinity of the control rod. Hence, the number of damaged fuel rods is quite small. Also, no mechanical energy release can be concluded to occur.




Figue 2

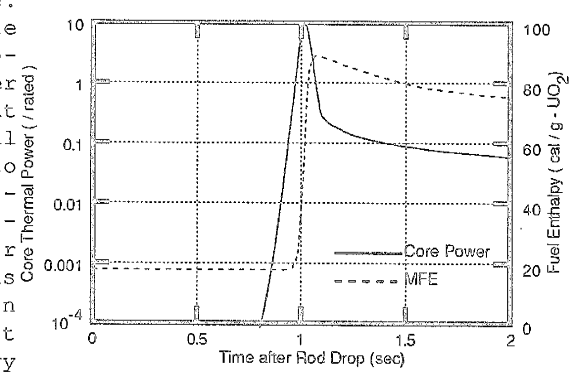
Time Histories of Core Thermal Power and Maximum Fuel Enthalpy (MFE) during a Cold RDA




Figue 3

Time Histories of Reactivity Components during a Cold RDA

In the hot stand-by core RDA, no fuel damage is analyzed. Fuel enthalpy also grows instantly to its peak value, about 90 cal/g-$UO_2$, shortly after core thermal power reaches its peak level. However, after that, it descends gradually in contrast to the cold core RDA case. This is due to the effective power suppression after peak power time, which is the result of a sudden drop in total reactivity, mainly due to moderator density reactivity feedback, in co-operation with Doppler feedback. Steam voids explosively generate in HSB RDAs. It is apparent that the mechanical energy



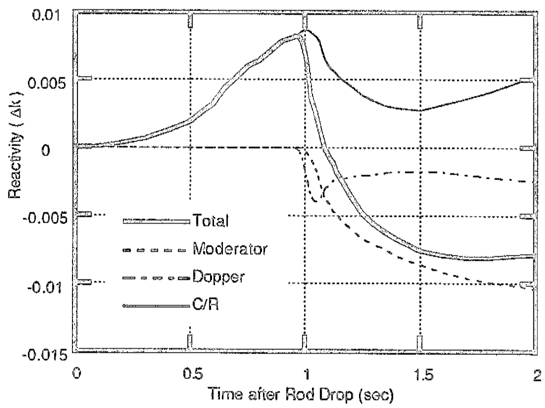

Figue 4

Time Histories of Core Thermal Power and Maximum Fuel Enthalpy (MFE) during a HSB RDA is not released in this case.

5 CONCLUSION

Analyses for fuel enthalpy behavior and evaluations for fuel integrity in RDAs for typical BWRs were carried out. Judging from the enthalpy criteria for fuel damage, based on NSRR experiments, the three-dimensional simulation code predicts a small amount of cladding failure in the cold start-up core RDA, but predicts no failure in the hot-standby core RDA. The code also predicts no mechanical energy release in either core RDA.

The conditions applied to the RDA analyses are so conservative, that it can be concluded that no mechanical energy generates in BWR cores.




Figue 5

Time Histories of Reactivity Components during a HSB RDA

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