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Original Article

# Uncertainty and sensitivity analysis in reactivity-initiated accident fuel modeling: synthesis of organisation for economic co-operation and development (OECD)/nuclear energy agency (NEA) benchmark on reactivity-initiated accident codes phase-II

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ABSTRACT

In the framework of OECD/NEA Working Group on Fuel Safety, a RIA fuel-rod-code Benchmark Phase I was organized in 2010–2013. It consisted of four experiments on highly irradiated fuel rodlets tested under different experimental conditions. This benchmark revealed the need to better understand the basic models incorporated in each code for realistic simulation of the complicated integral RIA tests with high burnup fuel rods. A second phase of the benchmark (Phase II) was thus launched early in 2014, which has been organized in two complementary activities: (1) comparison of the results of different simulations on simplified cases in order to provide additional bases for understanding the differences in modelling of the concerned phenomena; (2) assessment of the uncertainty of the results. The present paper provides a summary and conclusions of the second activity of the Benchmark Phase II, which is based on the input uncertainty propagation methodology. The main conclusion is that uncertainties cannot fully explain the difference between the code predictions. Finally, based on the RIA benchmark Phase-I and Phase-II conclusions, some recommendations are made.


## 1. Introduction

Reactivity-initiated accidents (RIA) are nuclear reactor accidents that involve unwanted increase in fission rate and reactor power. A rapid power excursion leads to an adiabatic heating of the fuel pellets and may lead to failure of the fuel rod. Thereon, a large part of the fuel pellet inventory could be dispersed into the coolant. Fuel-coolant interaction could cause pressure pulses or rapid steam generation, which could damage not only fuel assemblies or other core components but also the reactor pressure vessel. The fuel rod behavior during RIA has to be analyzed to verify its compliance with safety criteria [1].

RIA fuel rod codes have been developed for a significant period of time and validated against their own available database. However, the high complexity of the scenarios dealt with has resulted in a number of different models and assumptions adopted by code developers; additionally, databases used to develop and validate codes have been different depending on the availability of the results of some experimental programs. This diversity makes it difficult to find the source of estimation discrepancies, when these occur.

A technical workshop on “Nuclear Fuel Behavior During Reactivity-Initiated Accidents” was organized by the nuclear energy agency (NEA) of the organisation for economic co-operation and development (OECD) in September 2009 [2]. A major highlight from the session devoted to RIA safety criteria was that RIA fuel rod codes are now widely used, within the industry as well as the technical safety organizations, in the process of setting up and assessing revised safety criteria for RIA design basis accidents. This turns mastering the use of these codes into an outstanding milestone, particularly in safety analyses. To achieve that, a thorough understanding of code predictability is mandatory.

At the conclusion of the workshop, it was recommended that a benchmark (RIA benchmark Phase I) between these codes be organized to give a sound basis for their comparison and assessment. To maximize the benefits from this RIA benchmark Phase I exercise, it was decided to use a consistent set of four experiments using fuel

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rodlets refabricated from similar high-burnup full-length rods under different experimental conditions. A detailed and complete RIA benchmark Phase I specification was prepared to ensure, as much as possible, the comparability of the calculation results submitted [3].

The main conclusions of the RIA benchmark Phase I are the following [4]:

- With respect to the thermal behavior, the differences in the evaluation of fuel temperatures remained consistent with each other, although these differences were significant in some cases. The situation was very different for the cladding temperatures that exhibited considerable scatter, in particular for cases when water boiling occurred.
- With respect to mechanical behavior, the parameter of largest interest was the cladding hoop strain, because failure during RIA transient results from the formation of longitudinal cracks. When compared to the results of an experiment that involved only pellet clad mechanical interaction, the predictions from the different participants appeared acceptable even though there was a factor of 2 between the highest and the lowest calculations. This conclusion was not so favorable for cases in which water boiling had been predicted to appear: a factor of 10 for the hoop strain between the calculations was exhibited. Other mechanical results compared during the RIA benchmark Phase I were fuel stack and cladding elongations. The scatter remained limited for the fuel stack elongation, but the cladding elongation was found to be much more difficult to evaluate.
- The fission-gas release evaluations were also compared. The ratio of the maximum to the minimum values appeared to be roughly two, which is considered to be relatively moderate given the complexity of fission gas release processes.
- Failure predictions, which may be considered as the ultimate goal of fuel code dedicated to the behavior in RIA conditions, were compared: it appears that the failure/no failure predictions are fairly consistent between the different codes and the experimental results. However, when assessing the code qualification, one should rather look at predictions in terms of enthalpy at failure because it is a parameter that may vary significantly between different predictions (and is also of interest in practical reactor applications). In the frame of this RIA benchmark Phase I, the failure prediction levels among the different codes were within a $\pm$50% range.

As a conclusion of the RIA benchmark Phase I, it was recommended to launch a second phase exercise with the following specific guidelines:

- The emphasis should be put on deeper understanding of the differences in modeling of the different codes; in particular, looking for simpler cases than those used in the first exercise was expected to reveal the main reasons for the observed large scatter in some conditions such as coolant boiling.
- Owing to the large scatter between the calculations that was shown in the RIA benchmark Phase I, it appears that an assessment of the uncertainty of the results should be performed for the different codes. This should be based on a well-established and shared methodology. This also entailed performing a sensitivity study of results to input parameters to assess the impact of initial state of the rod on the final outcome of the power pulse.

The second phase of the RIA fuel rod code benchmark (RIA benchmark Phase II) was launched early in 2014. This RIA benchmark Phase II has been organized as two complementary activities [5,6]:

- The first activity is to compare the results of different simulations on simplified cases to provide additional bases for understanding the differences in modeling of the concerned phenomena.
- The second activity is focused on the assessment of the uncertainty of the results. In particular, the impact of the initial states and key models on the results of the transient are investigated.

The present article provides the specification (§2), the participants and their adopted codes (§3), a detailed comparison of the results (§4), and conclusions and recommendations from the second activity (§5).

## 2. Benchmark specification

The objective of this second activity of the RIA benchmark Phase II [7] was to assess the uncertainty of the results. In particular, the impact of the initial states and key models on the results of the transient behavior of fuel rods was investigated. All uncertainties were considered as either statistical or random ones. The identification and treatment of epistemic uncertainties, if any, was beyond the scope of the project. In addition, a sensitivity study was performed to identify or confirm the most influential input uncertainties.

### 2.1. Description of the reference case

Considering the feedback from Phase I of the RIA benchmark, the uncertainty analysis was initially intended to be performed on the foreseen CABRI international program test, CIP3-1, on an irradiated ZIRLO-cladded $UO_2$ fuel rodlet in pressurized water reactor (PWR) representative conditions. This case was also considered in Phase I of the benchmark and resulted in the largest differences in the predictions from the different codes [4]. However, in the first activity of Phase II, it appeared that, despite simplifications in the defined cases, a significant spread of results was still present. The original thought of using the CIP3-1 case (interesting due to its high burnup) seemed too ambitious for the reference case due to the complex initial rod state evaluation, the large effort required from participants and the risk of nonconclusive outcomes.

Therefore, it was agreed that the numerical reference case should be "Case 5" for the first activity (see Refs. [5] and [6]). To limit the differences linked to the initial state of the fuel, the case is limited to a fresh 17 $\times$ 17 PWR-type fuel rodlet, as described in Fig. 1, with standard $UO_2$ fuel pellet without dish and chamfer and Zircloy-4 cladding. It is also assumed that there is no initial gap between the fuel and the clad; these are considered perfectly bonded from the mechanical point of view. The upper plenum is pressurized with helium at a typical pressure of a PWR rod (2 MPa at 20°C).

The thermal-hydraulic conditions during the transient are representative of water coolant in nominal PWR hot zero power conditions (coolant inlet conditions: $P_{cool}$ = 15.5 MPa, $T_{cool}$ = 280°C and $V_{cool}$ = 4 m/s). These conditions are established by letting the coolant pressure and temperature increase linearly from ambient conditions over 50 seconds, after which a 50 seconds pretransient hold time is postulated to establish steady-state conditions. The reference pulse starts from zero power at t-100 seconds. It is considered to have a triangular shape, with 30 ms of full width at half maximum (FWHM). A high value for the rod maximal power in the fuel is considered to lead to a specific injected energy of 127 cal/g. This value should provoke departure from nucleate boiling.

All parameters of rod design and boundary conditions are specified in Table 1.