Reactivity-Initiated Accident Fuel Rod Codes Benchmark Phases I-III

Synthesis Report
Reactivity-Initiated Accident Fuel Rod Codes
Benchmark Phases I-III: Synthesis Report
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Foreword

The Nuclear Energy Agency (NEA) Working Group on Fuel Safety (WGFS) is tasked with advancing the understanding of nuclear fuel safety issues by assessing the technical basis for current safety criteria and their applicability to high burnup and to new fuel designs and materials. The group aims to facilitate international convergence in this area, including as regards experimental approaches and interpretation and the use of experimental data relevant for fuel safety.

One of the key areas in fuel safety is the analysis of fuel behaviour under reactivity-initiated accident (RIA) conditions for which the WGFS has led major fuel performance codes benchmarking activities over the last decade. These activities were initiated in 2010 following an NEA technical workshop on “Nuclear Fuel Behaviour during Reactivity-Initiated Accidents” which was organised in September 2009. This reference report synthesises these benchmark activities and provides recommendations regarding future research and code enhancement needs for RIA safety analysis.
Acknowledgements

Special gratitude is expressed to Vincent Georgenthum, Jean Baccou, Nicolas Tregourès (IRSN, France), Asko Arkoma (VTT, Finland) and Jinzhao Zhang (TRACTEBEL, Belgium) for drafting the report, to Marco Cherubini (NINE) for reviewing the report as well as to all the contributors of the three phases of the RIA Fuel Rod Codes Benchmark.
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<th>Description</th>
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<tbody>
<tr>
<td>BFC</td>
<td>bottom of fissile column</td>
</tr>
<tr>
<td>BWR</td>
<td>boiling water reactor</td>
</tr>
<tr>
<td>CABRI</td>
<td>test reactor in France</td>
</tr>
<tr>
<td>CEA</td>
<td>Commissariat à l’Energie Atomique et aux énergies alternatives (Alternative Energies and Atomic Energy Commission, France)</td>
</tr>
<tr>
<td>CFP</td>
<td>cladding failure prediction, defined by each participant: if CFP &lt; 1 no failure is predicted, if CFP ≥1 a failure is predicted</td>
</tr>
<tr>
<td>CSNI</td>
<td>Committee on the Safety of Nuclear Installations (NEA)</td>
</tr>
<tr>
<td>DHR</td>
<td>variation of radial average enthalpy</td>
</tr>
<tr>
<td>DNB</td>
<td>departure from nucleate boiling</td>
</tr>
<tr>
<td>ECT</td>
<td>cladding total axial elongation</td>
</tr>
<tr>
<td>ECTH</td>
<td>cladding total (thermal + elastic + plastic) hoop strain</td>
</tr>
<tr>
<td>EFT</td>
<td>fuel column total axial elongation</td>
</tr>
<tr>
<td>ENEA</td>
<td>Agenzia nazionale per le nuove tecnotologie, l’energia e lo sviluppo economico sostenibile (National Agency for New Technologies, Energy and Sustainable Economic Development, Italy)</td>
</tr>
<tr>
<td>ETZ</td>
<td>cladding residual hoop strain</td>
</tr>
<tr>
<td>FGR</td>
<td>fission gas release</td>
</tr>
<tr>
<td>FWHM</td>
<td>full width at half maximum</td>
</tr>
<tr>
<td>GAP</td>
<td>gap width</td>
</tr>
<tr>
<td>GRS</td>
<td>Gesellschaft für anlagen- und ReaktorSicherheit (Germany)</td>
</tr>
<tr>
<td>HFC</td>
<td>fuel-clad heat-exchange</td>
</tr>
<tr>
<td>INL</td>
<td>Idaho National Laboratory (United States)</td>
</tr>
<tr>
<td>IRSN</td>
<td>Institut de Radioprotection et de Sûreté Nucléaire (Institute for Radiological Protection and Nuclear Safety, France)</td>
</tr>
<tr>
<td>J</td>
<td>rice integral</td>
</tr>
<tr>
<td>JAEA</td>
<td>Japan Atomic Energy Agency</td>
</tr>
<tr>
<td>KINS</td>
<td>Korean Institute of Nuclear Safety</td>
</tr>
<tr>
<td>LUB/UUB</td>
<td>lower/upper uncertainty bound</td>
</tr>
<tr>
<td>LUB_{min}/UUB_{max}</td>
<td>minimum/maximum of the LUBs/UUBs provided by all participants</td>
</tr>
<tr>
<td>NEA</td>
<td>Nuclear Energy Agency</td>
</tr>
<tr>
<td>NINE</td>
<td>Nuclear and Industrial Engineering (Italy)</td>
</tr>
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</table>
### LIST OF ABBREVIATIONS AND ACRONYMS

<table>
<thead>
<tr>
<th>Abbreviation</th>
<th>Description</th>
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<tbody>
<tr>
<td>NRC</td>
<td>Nuclear Regulatory Commission (United States)</td>
</tr>
<tr>
<td>NSRR</td>
<td>Nuclear Safety Research Reactor (Japan)</td>
</tr>
<tr>
<td>OECD</td>
<td>Organisation for Economic Co-operation and Development</td>
</tr>
<tr>
<td>PCMI</td>
<td>pellet-cladding mechanical interaction</td>
</tr>
<tr>
<td>PDF</td>
<td>probability density function</td>
</tr>
<tr>
<td>PPN</td>
<td>peak power node</td>
</tr>
<tr>
<td>PRCC</td>
<td>partial rank correlation coefficient</td>
</tr>
<tr>
<td>PSI</td>
<td>Paul Scherrer Institute (Switzerland)</td>
</tr>
<tr>
<td>PWR</td>
<td>pressurised water reactor</td>
</tr>
<tr>
<td>q0</td>
<td>uncertainty interval width</td>
</tr>
<tr>
<td>q1</td>
<td>indicator of the position of the reference calculation within the uncertainty interval</td>
</tr>
<tr>
<td>QT</td>
<td>Quantum Technologies AB (Sweden)</td>
</tr>
<tr>
<td>RCC</td>
<td>rank (Spearman's) correlation coefficient</td>
</tr>
<tr>
<td>REF</td>
<td>reference calculation</td>
</tr>
<tr>
<td>REFmin/REFmax</td>
<td>minimum/maximum of the reference calculations provided by all participants</td>
</tr>
<tr>
<td>RFO</td>
<td>fuel outer radius</td>
</tr>
<tr>
<td>RIA</td>
<td>reactivity-initiated accident</td>
</tr>
<tr>
<td>SAPIUM</td>
<td>Systematic APproach for Input Uncertainty quantification of the physical Models in thermal-hydraulic codes</td>
</tr>
<tr>
<td>SCH</td>
<td>cladding hoop stress at outer part of the clad</td>
</tr>
<tr>
<td>SSM</td>
<td>Strålsäkerhetsmyndigheten (Swedish Radiation Safety Authority)</td>
</tr>
<tr>
<td>TCO</td>
<td>temperature of cladding outer surface</td>
</tr>
<tr>
<td>TFC</td>
<td>temperature of fuel centreline</td>
</tr>
<tr>
<td>TFO</td>
<td>temperature of fuel outer surface</td>
</tr>
<tr>
<td>UJV</td>
<td>ÚJV řež (Nuclear research institute, Czech Republic),</td>
</tr>
<tr>
<td>VTT</td>
<td>Teknologian Tutkimuskeskus VTT Oy/VTT (Technical Research Centre of Finland Ltd)</td>
</tr>
<tr>
<td>WGFS</td>
<td>Working Group on Fuel Safety (NEA/CSNI)</td>
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</table>
Executive summary

The Nuclear Energy Agency (NEA) Committee on the Safety of Nuclear Installations (CSNI) organised a technical workshop on “Nuclear Fuel Behaviour during Reactivity-Initiated Accidents” (RIAs) in September 2009. The workshop participants recommended that a benchmark between RIA fuel performance codes, later known as the RIA Fuel Rod Codes Benchmark Phase I, be organised to give a sound basis for calculation comparisons and modelling assessments. This recommendation was endorsed by the NEA/CSNI Working Group on Fuel Safety (WGFS) in 2010, and Phase I was carried out from 2010 to 2013.

RIA Fuel Rod Codes Benchmark Phase I covered modelling of four experiments on very similar highly irradiated fuel rods under different experimental conditions: the VA-1 and VA-3 tests performed in the NSRR facility in Japan in stagnant coolant water; the CIP0-1 test conducted in the CABRI test reactor in sodium coolant loop and the CIP3-1 test to be conducted in water coolant loop (pre-test calculations). The outcomes of the comparisons between participants’ calculation results can be summarised as follows. With respect to the thermal behaviour, differences in the calculated fuel temperatures remained limited, although significant in some cases. Cladding temperatures exhibited considerable scatter, in particular for the cases where water boiling occurred. With respect to mechanical behaviour, under pellet-cladding mechanical interaction (PCMI) conditions, as occurred in the CIP0-1 test, there was a factor of two between the highest and the lowest calculation predictions in cladding hoop strain. The conclusion was not as favourable for those cases where water boiling had been predicted: a factor of ten for the hoop strain between the calculations was noticed. Other mechanical results compared during Phase I were fuel stack and cladding elongations. The scatter remained limited for the fuel stack elongation, while the cladding elongation was found to be much more difficult to evaluate. Regarding the fission gas release (FGR), the ratio between maximum and minimum results was roughly two, which is considered to be relatively moderate given the complexity of FGR processes.

Since major causes of the differences between participants’ results produced in Phase I were not clearly identified, it was recommended to perform systematic sensitivity and uncertainty analyses in a new phase of the benchmark. This recommendation was endorsed by the WGFS in 2013, and Phase II was carried out in 2014-2016. It was agreed that the emphasis in RIA Fuel Rod Codes Benchmark Phase II would be placed on better understanding differences in modelling between various transient codes, in particular by defining a set of simplified cases aimed at clarifying the causes of the large scatter observed in the analysis of tests in Phase I. The cladding-to-coolant heat transfer in the case of water boiling under RIA conditions, and more specifically during the film boiling regime, was considered to be of particular interest as, firstly, large uncertainties exist in the models and, secondly, it leads to large differences in both thermal and mechanical predictions. Thus, as a first activity in Phase II, ten simplified artificial cases covering both pressurised water reactor (PWR) and boiling water reactor (BWR) conditions were defined with an increasing degree of complexity to assess various phenomena systematically. One case was mainly devoted to the thermal behaviour, three cases were focused on thermo-mechanical behaviour and the rest studied thermo-hydraulics. Besides these basic cases, there were variations regarding pellet-cladding gap state (open/closed), pellet-cladding contact conditions (slip/no-slip), thermo-mechanical model, thermo-hydraulic conditions, maximum power, and filling gas pressure. In order to avoid possible divergences due to the evaluation of the initial state of an irradiated fuel, the cases were limited to fresh fuel.
As a second activity in Phase II, uncertainty and sensitivity analysis was performed on one of the simplified cases defined for the first activity. The recommended number of simulations for the statistically based derivation of the uncertainty was 200. The uncertain input parameters and their distributions were agreed upon prior to the simulations. The selected parameters included the fuel rod manufacturing tolerances, thermo-hydraulic boundary conditions, core power boundary conditions, and physical properties and models. A sensitivity study was conducted by calculating partial rank correlation coefficients at distinct instants in the course of the transient simulations.

The uncertainty and sensitivity analysis results from the fresh fuel case cannot be transferred as such to the case of irradiated fuel. Therefore, the WGFS launched in 2017 a complementary RIA Fuel Rod Codes Benchmark Phase III, carried out in 2018-2019, with a focus on uncertainty and sensitivity analyses of irradiated fuel. The same procedure and methods applied in Phase II were used for uncertainty and sensitivity analyses. Based on the results obtained during the first two phases, which confirmed the large discrepancy in RIA thermal-hydraulics modelling, it was decided to limit the Phase III to the early phase of a RIA event, during which the PCMI is the prominent phenomenon, and for which test data are available. The CIP0-1 test performed in the CABRI sodium loop, and already calculated in Phase I, was chosen for Phase III. With this choice, the dispersion of participants' results associated with the water boiling was avoided. In particular, uncertainties regarding fission product distribution, fuel microstructure, cladding corrosion state and gap conductance were of interest.

A total of 21 organisations, representing 14 countries, participated in the RIA Fuel Rod Codes Benchmark Phases I-III, providing results for some or all the cases proposed in the different phases of the benchmark. In terms of computer codes used, the spectrum was also very large: 8 base irradiation codes, 10 transient analysis codes, 12 statistical analysis tools (codes and in-house procedure).

Nearly all the participants used transient codes relying on a simplified geometrical representation of the fuel rod, usually referred to as 1.5-dimensional codes (i.e. the thermo-mechanics of the fuel rod is modelled in the radial direction but the axial direction is handled only by the communication of pellet-cladding gap pressure and the axial coolant conditions). Although some three-dimensional calculations may be done, it appeared that a detailed geometrical description was not necessary. Rather, it was considered more important to focus efforts on physical modelling.

The main conclusions of the three benchmark phases are as follows:

- The PCMI phase phenomena are well understood and there is a relatively good agreement between most of the transient codes on results during this phase.
- With respect to the fuel thermal behaviour, the differences in the estimation of fuel enthalpies and temperatures by the different codes are rather limited, especially for the maximum values of these parameters.
- Regarding FGR, there is no consensus between participants on its modelling. Therefore, there is a rather large dispersion between participants' results, which are not fully in agreement with experimental data even when taking into account uncertainties.
- Concerning cladding temperatures and hoop strain evaluations, considerable scatter is obtained for the cases where water boiling occurs. This scatter is clearly related to the cladding-to-coolant heat transfer modelling. There is indeed no consensus on the post-departure from nucleate boiling (DNB) behaviour and in particular on the heat transfer between cladding and coolant during the boiling crisis and on the cladding mechanical behaviour at high temperature. Boiling in RIA conditions is known to be significantly different from (quasi) steady-state conditions. Some codes assume that the steady-state correlations are applicable to RIA conditions while other codes use specific fast transient correlations (for critical heat flux, heat transfer in film boiling, rewetting conditions, etc.). Given the lack of sufficient experimental investigations on boiling heat transfer in RIA conditions, no sound recommendation can be made for the heat transfer correlations.
EXECUTIVE SUMMARY

• From the cases devoted to BWR conditions, it is clear that very few (if any) of the applied computer codes are able to handle the thermal-hydraulics conditions expected in a BWR RIA with large energy injection at cold, zero power conditions. This is not simply a question of uncertainties in the cladding-to-coolant heat transfer modelling; the excessive steam generation expected in the fuel assembly at atmospheric pressure can obviously not be handled by the simple thermal-hydraulics models of the transient codes.

• Various influential input parameters are identified for fresh fuel. For instance, injected energy, fuel enthalpy, parameters related to the rod geometry (pellet and cladding roughness, cladding inside diameter), fuel thermal expansion model, and the full width at half maximum (FWHM) of the power pulse have been found to be influential regarding the maximum value of each output parameter of interest. Uncertainties cannot fully explain the scatter observed during these benchmark exercises.

• The RIA Fuel Rod Codes Benchmark Phase III, which was focused on uncertainty and sensitivity analyses on an irradiated case, confirmed the conclusions of Phase II concerning the strong dependence of the uncertainty results on the type of outputs. Input data related to rod state after base irradiation (initial pellet-cladding gap, zirconia thickness, radial power profile, roughness) are also very influential. The initial pellet-cladding gap is the most influential in terms of the cladding failure prediction.

• Some codes have been developed to perform both base irradiation and transient calculations, which ensures continuity between the two phases but makes it more difficult to perform the decoupling. Their users thus had difficulties to match the pre-transient state defined in the specifications.

• The specifics of the applied codes and the user effect could play a more important role than the uncertainties.

• The OECD RIA Fuel Rod Codes Benchmark offered an opportunity to investigate the temperature effect, i.e. whether it is realistic to use the RIA fuel codes to transpose results from experiments performed at test reactors to typical power reactor conditions, because it had results of experiments performed on similar rods but at different temperatures. The conclusion was that the extrapolation to reactor conditions should be done with caution given the scatter that existed between the predictions of various codes. Making such a transposition with two different codes can lead to results that differ significantly.

The table below summarises the scope of each of the three RIA benchmark phases.

Table ES1: Scope of each of the three RIA benchmark phases

<table>
<thead>
<tr>
<th>Benchmark phase</th>
<th>Experiments and/or calculation cases</th>
<th>Scope</th>
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<tbody>
<tr>
<td>Phase I</td>
<td>NSRR VA-1 and VA-3, stagnant coolant water&lt;br&gt;CABRI CIP0-1, sodium coolant loop&lt;br&gt;CABRI CIP3-1 (pre-test calculation), water coolant loop</td>
<td>Full transient calculations</td>
</tr>
<tr>
<td>Phase II</td>
<td>Uncertainty and sensitivity calculations on 10 simplified cases:&lt;br&gt;  - Thermal behaviour (1)&lt;br&gt;  - Thermomechanical behaviour (3)&lt;br&gt;  - Thermalhydraulic behaviour (6 cases)&lt;br&gt;Detailed uncertainty and sensitivity calculations on a selected thermalhydraulic case</td>
<td>Focus on cladding to coolant transfer in the case of water boiling</td>
</tr>
<tr>
<td>Phase III</td>
<td>CABRI CIP0-1, sodium coolant loop</td>
<td>Focus on PCMI</td>
</tr>
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</table>
The recommendations of the three phases are the following:

**Improvement and validation of the models implemented in transient codes**

- It is recommended to build up a comprehensive and robust database consisting of results from both separate effect tests and integral RIA simulation tests. This database could be defined in an international framework. In this way, both individual model validation and model integration into codes would be feasible. The database could be shared by the modellers, whenever possible, to ease the comparison of simulation results from various codes. For the purpose of model validation, it is important to include detailed information on the pre-irradiation conditions of the test rods, as well as results available from pre-test and post-test characterisation.

- Models related to the evolution of the pellet-cladding gap should be improved and validated for RIA conditions as this has been shown to have a significant effect on fuel rod response. To reach this objective, in-reactor measurements of cladding strain during RIA simulation tests should be done (or at least attempted).

- Fuel and cladding thermo-mechanical models (with the associated material properties) should be further improved and validated more extensively against a well-qualified RIA database. Efforts should be made to improve mechanical models, in particular the failure probability models and failure criteria of claddings.

- Further developments are required for the transient axial gas flow and release, with attention to their validation, which first involves gathering more high quality data.

- The cladding-to-coolant heat transfer in the case of water boiling during very fast transients is of particular interest, and capabilities related to modelling this phenomenon should be improved. Co-operation between existing experimental teams could be established to collect and share all existing data and to formulate specific proposals in order to reduce the lack of knowledge and achieve common understanding on the subjects.

**Reduction of user and code effect for best estimate simulation**

- To reduce the user effect and the code effect, the code development teams should, where possible, provide recommendations regarding the version of their codes to be used, the models, as well as the numerical parameters to be used (mesh size, time step, etc.). The availability of a RIA experimental database can help code users to assess their modelling techniques.

**Treatment of uncertainties**

- When the uncertainty evaluation is based on input uncertainty propagation, a first task to be accomplished (before uncertainty and sensitivity analyses) is the quantification of input uncertainties that was partly performed in this benchmark by expert judgement. In order to minimise the user effect, the recently outlined SAPIUM guidance [17] could be exploited for a transparent and rigorous model for input uncertainty quantification.

- Sensitivity and uncertainty analyses should be considered as an integral part of computer code applications to RIA. These analyses should be directed towards models and phenomena for which the most substantial uncertainties are known to exist.

- Safety analysis studies can require uncertainty analysis on parameters associated to the state of the rod at the end of irradiation. However, for some codes, the transient pulse and the base irradiation are not considered apart. It could be of interest to develop strategies to allow the propagation of uncertainties on input parameters associated to irradiation behaviour.
Chapter 1. Introduction and background

Reactivity-initiated accident (RIA) fuel performance codes have been developed for a significant period of time and validated against their own available databases. However, the high complexity of the scenarios dealt with has resulted in a number of different models and assumptions adopted by the code developers. In addition, databases used to develop and validate the codes have been different depending on the availability of results from experimental programmes, so each code has its own characteristics and capabilities. These differences make it difficult to find the sources of discrepancies when calculation results are compared.

The Nuclear Energy Agency (NEA) Committee on the Safety of Nuclear Installations (CSNI) organised in September 2009 a technical workshop on “Nuclear Fuel Behaviour during Reactivity-Initiated Accidents” [1]. The workshop participants recommended that a benchmark between RIA fuel performance codes be organised to give a sound basis for calculations comparison and modelling assessment. The NEA/CSNI Working Group on Fuel Safety (WGFS) endorsed this recommendation and carried out a series of three benchmark exercises between 2010 and 2019.

In the RIA workshop, it was concluded that a detailed calculation of the initial state of a high burnup fuel is essential for the RIA simulations. Whereas models for analysing the early phase of an RIA, in which the pellet-cladding mechanical interaction (PCMI) is the prominent phenomenon, had been developed for RIA fuel codes, modelling of thermal-hydraulics in fast transient conditions and transient fission gas behaviour was identified to require further work. The simulation cases for the benchmarks were selected to reflect the above-mentioned issues, i.e. PCMI, water boiling phenomena, and high burnup effects.

RIA Fuel Rod Codes Benchmark Phase I covered modelling of four experiments on very similar highly irradiated fuel rods under different experimental conditions. Comparisons between participants’ calculation results showed that with respect to the thermal behaviour, differences in the fuel temperatures remained limited, although significant in some cases. Cladding temperatures exhibited considerable scatter, in particular for the cases where water boiling occurred. With respect to mechanical behaviour, under PCMI conditions, there was a factor of two between the highest and the lowest calculation predictions in cladding hoop strain. The conclusion was not as favourable for cases where water boiling had been predicted: a factor of ten for the hoop strain between the calculations was noticed. Other mechanical results compared during 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. Regarding fission gas release (FGR), the ratio between maximum and minimum of delivered results was roughly two, which is considered relatively moderate given the complexity of FGR processes.

Since the major causes of the differences between participants’ results produced in Phase I were not clearly identified, a systematic sensitivity and uncertainty analysis was performed in a new phase of the benchmark, Phase II. The emphasis was on better understanding the differences in modelling between various transient codes. In particular, by defining simpler cases than those used in Phase I it was expected that the causes of the observed large scatter in some results would be clarified. The cladding-to-coolant heat transfer in case of water boiling under RIA conditions, and more specifically during the film boiling regime, was considered to be of particular interest as, firstly, large uncertainties exist in the models, and secondly, it leads to large differences in both thermal and mechanical predictions. Thus, as a first activity in Phase II, ten simplified artificial cases covering both pressurised water reactor (PWR) and boiling water reactor (BWR) conditions were defined with an increasing degree of complexity to assess various phenomena systematically.
As a second activity, uncertainty and sensitivity analysis was performed on one of the simplified cases defined for the first activity.

Because the uncertainty and sensitivity analysis results from the fresh fuel case cannot be applied to the case of irradiated fuel, it was decided to launch a complementary RIA Fuel Rod Codes Benchmark Phase III, with a focus on uncertainty and sensitivity analyses of irradiated fuel. In particular, uncertainties regarding fission product distribution, fuel microstructure, cladding corrosion state, and gap conductance were of interest. Based on the results obtained during the first two phases, which confirmed the large discrepancy regarding RIA thermal-hydraulics modelling, it was decided to limit the scope of Phase III to the early PCMI phase of an RIA, and for which test data are available.

The leading organisations and co-ordinators of the benchmark phases were the IRSN (Phases I, II and III), the JAEA (Phase I), Tractebel (Phases II and III and NINE (Phase II). All the participating organisations and the applied codes in the three benchmark phases are listed in Table 1-1.

In this report, each phase of the benchmark is tackled separately: Phase I in Chapter 2, Phase II in Chapter 3, and Phase III in Chapter 4. The overall conclusions and recommendations of the benchmark ensemble are provided in Chapter 5.
Table 1-1: Participating organisations and the applied codes in the three phases of the transient codes benchmark. Phase II was divided into two activities

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<tr>
<th>Organisation</th>
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1: in Phase I; 2: in Phase II; 3: in Phase III; 4: in Phase I and Phase II activity 1; 5: in Phase I and Phase II activity 1 and 2; 6: plus TRACE in Phase I; 7: plus TRABCO; 8: plus RELAPS.
Chapter 2. RIA benchmark Phase I

2.1. Objectives of Phase I

As a conclusion of the CSNI RIA workshop held in 2009 [1], participants noted that even though the various RIA fuel modelling codes show reasonably good agreement with existing experimental results, the codes use different assumptions, which raises the question of the validity of extrapolation of the results to reactor conditions. They therefore recommended that a benchmark between RIA modelling codes be organised, with a goal of laying a sound basis for the codes’ comparison and assessment. In addition, the participants pointed out that some efforts should be devoted to compare the modelling between the codes. To do so, the participants set up and filled in a questionnaire listing for each code the models used.

The common perception was that the test facilities (with the exception of the new CABRI water loop) do not provide conditions prototypical to a commercial light water reactor during a hypothetical RIA [3]. It is generally believed that these differences may mostly impact the cladding heating up and deformation. Another major objective besides code comparison was to shed some light on the cladding temperature effect.

2.2. Specifications of Phase I

Detailed and complete benchmark specifications were prepared in order to make the calculation results as comparable as possible. It was an iterative process; after comparison of the initial results, modifications were made to the benchmark specification to increase clarity, and an opportunity to re-perform the calculations was provided to all the participants. Information on the fuel rod fabrication data, on base irradiation history and conditions and on pre-test examination results was provided in the benchmark specifications. Each participant then used their base irradiation code, listed in Table 1.1, to initialise the transient calculations.

A consistent set of four experiments on sibling high burnup fuel rods tested under different experimental conditions was selected for the exercise. The rods, fabricated by Enusa, were standard pressurised water reactor (PWR) 17x17 UO₂ rods with ZIRLO™ cladding, and were irradiated in the Vandellos-2 reactor up to the maximum local burnup close to 75 GWd/t. Each test segment experienced essentially the same pre-test irradiation history.

The selected experiments were:

- Low temperature, low pressure, stagnant water coolant, very narrow power pulse (NSRR VA-1).
  
  The VA-1 experiment was conducted in the NSRR reactor (the pool test reactor in Japan) at room temperature and atmospheric pressure. The maximum injected energy at peak power node was 140 cal/g (i.e. 585 J/g, evaluated from the specifications by integration of power between 0 and 0.5 s) and the pulse full width at half maximum (FWHM) was 4.4 ms. The fuel rod failed during this experiment at injected energy of 66 cal/g (i.e. 276 J/g, evaluated from the specifications by integration of power between zero seconds and the rupture time). Initial rodlet fill pressure was 0.1 MPa (at 20°C).

- High temperature, medium pressure, stagnant water coolant, very narrow power pulse (NSRR VA-3).
  
  The VA-3 experiment was performed in the NSRR at -280°C and 7 MPa. The maximum injected energy at peak power node was 115 cal/g (i.e. 481 J/g, evaluated from the specifications by integration of power between 0 and 0.5 s) and the pulse FWHM was 4.4 ms.
The fuel rod failed during this experiment at injected energy of 86 cal/g (i.e. 360 J/g, evaluated from the specifications by integration of power between zero seconds and the rupture time). The initial rodlet fill pressure was 0.1 MPa (at 20°C).

- High temperature, low pressure, flowing sodium coolant, larger power pulse (CABRI CIP0-1).

The CIP0-1 experiment was performed in the CABRI sodium loop facility in France at 280°C and low pressure (~0.3 MPa). The maximum-deposited energy at peak power node was equal to 99 cal/g (i.e. 414 J/g) and the pulse FWHM was 32.4 ms. The fuel rod did not fail and the initial rodlet fill pressure was 0.3 MPa (at 20°C).

- High temperature, high pressure, flowing water coolant, medium width power pulse (CABRI CIP3-1).

The CIP3-1 experiment is planned to be done in the CABRI water loop facility, at 280°C and 15.5 MPa. The benchmark simulation was a blind calculation. The anticipated maximum-deposited energy at peak power node is 115 cal/g (i.e. 481 J/g) and the pulse FWHM is expected to be about 8.8 ms. The initial rodlet fill pressure was set to 5 MPa (at 20°C).

After the base irradiation, the cladding outer surface oxidation was significant: it varied between 50 and 120 µm over the lengths of the four segments (about 50-110, 50-120, 60-105, and 65-110 µm in CIP0-1, CIP3-1, VA-1 and VA-3, respectively). The cladding was highly hydrided, with a mean hydride concentration of about 1000 ppm in the CIP0-1 test segment. It is also worth noting the very tight bonding between the fuel and inner zirconia layer seen in adjacent span to the CIP0-1 segment.

As an example of post-test examination results, the CIP0-1 rodlet showed substantial outer surface oxide spallation along all angular orientations with many spalled areas (see in Figure 2-1, to the left, the white area) of large dimensions (several millimetres wide and often several centimetres long). The oxide-spalling phenomenon was also noticeable on the cladding diameter and zirconia thickness measurements that were performed along four and eight azimuths, respectively (see Figure 2-1, middle and right). The cladding residual hoop strain ranged from 0% at both ends of the CIP0-1 segment, to 0.5 ± 0.1% at the peak power node (PPN) location. According to the rodlet puncturing, the fission gas release was estimated to be 13-16% of the total fission gas produced.

![Figure 2-1: Visual examination of the rod after CIP0-1 test (left), zirconia thickness (middle) and cladding diameter measurements (right)](source: Courtesy of V. Geogenthum from [15].)

One of the main difficulties in calculating cladding temperatures is the possible onset of boiling in experiments performed with water coolant. In order to ease the comparison between the codes and to determine the areas of agreement and disagreement of the models, an additional hypothetical case with boiling inhibited was computed for CIP3-1, VA-1 and VA-3. It was also requested to run additional CIP3-1 calculations in which the cladding outer temperature was prescribed, and at the same time using a flat axial power profile. The temperature trace for this case was issued from a SCANAIR calculation in which boiling was predicted.
Due to the scatter in the calculation results of the above-mentioned cases, a simpler case was proposed in order to be able to check basic modelling assumptions in various codes. The modelling group selected the NH-562-12 experiment performed with fresh fuel in the NSRR reactor. However, when the predictions for this case were compared, it appeared that the scatter was even more pronounced for this test than for the previous ones. The reason was that as there is no pellet-cladding contact in a fresh fuel rod prior to RIA, the assumptions relative to the contact conditions, e.g. gap width and friction coefficient, have a larger impact on the mechanical results than for high burnup fuel. Due to a lack of time available to investigate this case sufficiently, it was decided not to include it in the scope of Phase I.

2.3. Summary of results of Phase I

During Phase I, it was not possible to assess the influence of the initial state resulting from the base irradiation. Evidence of the differences in the base irradiation calculation results was apparent for instance in the scatter of initial gap width values between the participants’ results. Another boundary condition causing scattered results was the power pulse: although the power pulses were precisely defined, it unexpectedly appeared that various codes interpreted this information differently, resulting in differences in the injected energy. For instance, in CIP0-1 simulations, there was a difference of about 10% between the minimum and maximum injected energies between the results (see Figure 2.2, upper figure). In the case of narrow pulses in NSRR tests, in addition to the scatter, the time evolutions of injected energies showed differences.

With respect to the thermal behaviour, the differences in interpretation of power pulse resulted in differences in the maximum values of enthalpy variation, i.e. variation of mean enthalpy with respect to the initial conditions of the transient. In the longer term, the slopes of the curves could differ as a result of deviating heat transfer models in the codes (see Figure 2.2, lower figure). The description of coolant boiling requires specific models with additional uncertainties that amplify this deviation. When boiling was artificially inhibited, differences on the thermal behaviour were less pronounced.

Differences in evaluation of the fuel temperatures remained limited overall (see Figure 2.3, upper figure), although significant in some cases. Differences in fuel centreline temperatures are essentially due to differences in the use of the input data (power pulse). Regarding the maximum fuel temperature situated near the pellet periphery in high burnup fuel RIA, most of the models had consistent predictions but some predicted values were much higher than the majority. This may be a result of specific modelling assumptions.

On the contrary, the cladding temperatures exhibited considerable scatter, in particular in cases where water boiling occurred, as shown in the NSRR VA-1 simulations in Figure 2.3 (lower figure). This was explained by the very low thermal inertia of the cladding compared to that of the fuel. Thus, even limited differences in the heat transfer conditions have a large impact on cladding temperatures whereas they are hardly noticeable on fuel temperatures. In VA-1 simulations with boiling allowed (see Figure 2-3, lower figure), two groups of predictions can be seen. In the first group, temperatures remained below 1 100°C, decreased quite rapidly after 0.6 s, and most of them showed rewetting, though rewetting was predicted at different points in time. In the second group, temperatures higher than 1 100°C were predicted and they showed no or only little decrease during the studied period. The latter group is typical of codes in which steady-state cladding-to-coolant heat transfer correlations are applied in transient conditions.

Indeed, boiling under fast transient conditions is known to be significantly different from that under steady-state conditions. Especially the film boiling heat transfer model was responsible for the large differences between the calculations. In the first group, differences in temperature levels and rewetting time were essentially due to the lack of validation of the corresponding models. Even with boiling artificially inhibited in VA-1 simulations, there was large scatter in cladding temperatures with a maximum ranging from about 350°C to about 900°C (at peak power node at metal-oxide interface).
Figure 2-2: Injected energy in the whole rodlet in CABRI CIP0-1 (upper figure) and variation of mean enthalpy at peak power node in NSRR VA-1 (lower figure)

Source: [2].
Figure 2-3: Temperature evolutions of fuel centreline in the NSRR VA-3 (upper figure) and cladding at metal-oxide interface in VA-1 (lower figure) tests at peak power node

Source: [2].
With respect to mechanical behaviour, when compared to the experimental results that involved only pellet-cladding mechanical interaction (PCMI), the cladding hoop strain predictions appeared acceptable, even though there was a factor of two between the highest and the lowest calculation results. Simulated and measured cladding permanent hoop strain profiles after the CIP0-1 test are shown in Figure 2.4 (upper figure). Due to the scatter in the experimental results, it was not possible to conclude that one of the calculations or one of the codes could provide more accurate evaluations than the others. The outcome was worse for a case in which both the experimental results were unknown and water boiling was predicted: then there was a factor of ten between the calculation results on the hoop strains (see Figure 2.4, lower figure). Most of the scatter was interpreted to be due to the differences in cladding temperatures, with some potential impact also from the use of models for the cladding mechanical behaviour, e.g. whether viscoplastic behaviour is modelled or not.

The scatter remained limited for the fuel stack elongation (see Figure 2.5, upper figure). However, the scatter in fuel radial deformation (see Figure 2.6, upper figure) was more significant than the scatter in fuel temperature evaluations due to the differences in modelling, e.g. fuel thermal expansion, cracking, and fuel mechanical behaviour. In conjunction with the differences in evaluation of the initial state of the rod, this leads to differences in timing of the gap closure in terms of variation of mean enthalpy. For instance, gap closure in CIP0-1 occurs between 0 to 30 cal/g (i.e. 125 J/g). This again leads to significant discrepancy in the cladding hoop strain, up to a ratio of three between the upper and the lower value at the end of the PCMI stage, as seen in Figure 2.6 (lower figure).

Compared to fuel elongation, the cladding elongation was found to be more difficult to evaluate (Figure 2.5, lower figure). This was anticipated as the cladding elongation is a result of the complex conditions of the contact between the fuel pellets and the cladding. Cladding temperature, the timing of the gap closure, and the assumptions describing the pellet-cladding contact (sliding, friction, sticking) all affect the calculated cladding elongation. As seen in Figure 2.5 (lower figure), even though the calculated values are generally comparable to the measurement in CIP0-1 test, it is obvious that the computed values should be used with caution and be associated with proper uncertainty bounds.

The ratio of maximum and minimum values of fission gas release was roughly two, which is relatively moderate given the complexity of fission gas release processes.

Failure predictions in terms of failure/non-failure and the time of failure were relatively consistent between the various codes and with experimental results. However, when looking at predictions in terms of enthalpy at failure, which is of interest in practical reactor applications, typical variations between calculations were found to be within a +/- 50% range. For instance in VA-1 simulations, the calculated enthalpy variation at the time of failure ranged from 47 to 121 cal/g (i.e. 196 to 506 J/g). The variation could be explained by various factors, such as differences in defining the initial state of the rod prior to an RIA, differences in the gap width evolutions, and differences in the failure criteria. Indeed, different approaches were used to evaluate failure, e.g. based on the strain limit, strain energy density limit, or fracture mechanics approach. Moreover, even when using the same code and the same approach, different users may have had different predictions as the failure criterion had been based on different parameters.

As there were multiple FRAPTRAN and SCANAIR users, the so-called user effect could be estimated within the users of these codes. For both codes, the user effect was found to be very limited in the cases selected for this benchmark, and nearly negligible if compared to the differences between the results of different codes. In order to generalise this result, however, more cases would need to be analysed.
Figure 2-4: **Cladding permanent hoop strain profiles in the CABRI CIP0-1 (upper figure) and CIP3-1 (lower figure) tests**

Source: [2].
Figure 2-5: Evolutions of fuel stack (upper figure) and cladding (lower figure) total axial elongation in the CABRI CIP0-1 test

Source: [2].
Figure 2-6: Fuel radial deformation (upper figure) and cladding mechanical hoop strain at metal-oxide interface (lower figure) at peak power node vs. variation of mean enthalpy in the CABRI CIP0-1 test.

Source: [2].
2.4. Conclusions and recommendations of Phase I

Nearly all the participants in Phase I used a code that relies on a simplified geometrical representation of the fuel modelling, usually referred to as 1.5-dimensional codes (i.e. the thermo-mechanics of the fuel rod is modelled in the radial direction but the axial direction is handled only by the communication of pellet-cladding gap pressure and the axial coolant conditions). Although some three-dimensional calculations may be done as shown by one participant, it appeared that the detailed geometrical description would not be a priority. Rather, it was considered more important to focus efforts on physical modelling.

During the benchmark Phase I, one source of differences between the results was identified in the way boundary conditions are provided to the code, in particular how the power pulse was interpreted within various codes. It was recommended that the code users examine carefully the way the input data are used because this source of difference, which appeared to be significant, should be completely removed.

Based on the questionnaire on code models, or on the outputs of the codes, it was not possible to assess the sources of the modelling differences. This was due to the fact that the physical phenomena (thermal, mechanical, thermo-hydraulics, fission gas release) are closely coupled in RIA transients. It was concluded that in order to better understand the differences in modelling of various codes, comparisons of each model would have to be made separately. This could be achieved, for instance, by looking for simpler cases than those used in Phase I. Analysis of one simpler case was attempted in Phase I but it remained unsuccessful.

Although major causes for the divergent results were identified, it was recommended to perform more systematic uncertainty and sensitivity analyses to further assess the significance of the results produced. In particular, it was recommended to investigate the impact of the initial state on transient results. The cladding-to-coolant heat transfer in case of water boiling under RIA conditions, and more specifically during the film boiling regime, was considered also of particular interest. For film boiling, on the one hand, large uncertainties exist in the cladding-to-coolant heat transfer models and, on the other hand, large differences in the thermal as well as in the mechanical predictions were observed. Finally, in order to identify and assess the strengths and weaknesses of the physical models in the codes, the calculated results should be compared with reliable experimental data whenever possible.

The broader objective of the benchmark was to assess the possibility to evaluate the temperature effect, i.e. whether it is realistic to use the RIA fuel codes to transpose results, in particular enthalpy at failure, from experiments performed in test reactors to typical power reactor conditions. The benchmark offered an opportunity to investigate this question because it included results of experiments performed on similar rods but at different temperatures. The conclusion was that it was not possible to resolve the temperature effect. Therefore, it appears obvious that the extrapolation to reactor conditions should be done with caution given the scatter that existed between the predictions of various codes. If such a transposition is made with two different codes, the results may differ significantly.
Chapter 3. **RIA benchmark Phase II**

### 3.1. Objectives of Phase II

As a conclusion of the RIA benchmark Phase I [2], it was recommended to launch a second-phase exercise with the following specific guidelines:

- The emphasis needed to be on a deeper understanding of the differences in modelling of the different codes; in particular, it was expected that analysing simpler cases than those used in the first exercise would reveal the main reasons for the observed large scatter in some conditions, such as coolant boiling.

- Due to the large scatter between calculations shown in the RIA benchmark Phase I, an assessment of the uncertainty of the results was required for the different codes. This activity was 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 outcome of the power pulse.

The NEA Working Group on Fuel Safety endorsed these recommendations and a second phase of the RIA fuel-rod-code benchmark (RIA benchmark Phase II) was launched early in 2014. This RIA benchmark Phase II was organised as two complementary activities:

- The first activity was to compare the results of different simulations on simplified cases in order to provide additional bases for understanding the differences in modelling of the concerned phenomena.

- The second activity was focused on the assessment of the uncertainty of the results and sensitivity study. In particular, the impact of the initial state and key models on the results of the transient were to be investigated and the most influential input uncertainties were to be identified.

### 3.2. Specifications of Phase II

#### 3.2.1. First activity: Simplified cases

The detailed specifications of the simplified cases can be found in the NEA report *Reactivity-Initiated Accident (RIA) Fuel Codes Benchmark Phase-II – Volume 2: Task No. 1 Specifications* published in 2016 [5].

To limit the differences linked to the initial state of the fuel, the cases are limited to a fresh 17x17 pressurised water reactor (PWR)-type fuel rodlet with standard UO$_2$ fuel and Zircaloy-4 cladding, as shown in Figure 3.1.

Two different values for the cladding inner radius (RCI) are used to impose the presence or absence of an initial gap between the fuel and the clad. In most of the cases, the fuel and the cladding are considered bonded (no slipping between the fuel and the cladding is assumed) except for one case where perfect slipping between the fuel and the cladding is assumed as contact condition.
The thermal-hydraulic conditions during transient for each case could be:

- Water coolant in nominal PWR hot zero power (HZP) conditions (coolant inlet conditions: \( P_{\text{cool}} = 155 \text{ bar}, T_{\text{cool}} = 280^\circ \text{C} \) at \( V_{\text{cool}} = 4 \text{ m/s} \)), referred to as "PWR conditions".

- Water coolant in boiling water reactor (BWR) cold zero power (CZP) conditions (coolant inlet conditions: \( P_{\text{cool}} = 1 \text{ bar}, T_{\text{cool}} = 20^\circ \text{C} \) at \( V_{\text{cool}} = 0.0 \text{ m/s} \)), referred to as "BWR conditions".

- Imposed coolant bulk temperature (\( T_{\text{bulk}} = 300^\circ \text{C} \) during the first 5 seconds, then \( T_{\text{bulk}} = T_{\text{cool}} = 280^\circ \text{C} \) until the end of transient); imposed cladding-to-coolant heat transfer coefficient (\( H_{\text{trans}} = 4000 \text{ W/m}^2/\text{K} \) during the first 5 seconds, then \( H_{\text{trans}} = H_{\text{steady}} = 40000 \text{ W/m}^2/\text{K} \) until the end of transient) and external pressure at 155 bar (\( P_{\text{cool}} \)), referred to as "imposed conditions".

- Imposed external cladding temperature at 280°C (\( T_{\text{cool}} \)) and external pressure at 155 bar (\( P_{\text{cool}} \)), referred to as "fixed conditions".

The pulse starts from zero power and is considered to have a triangular shape, with 30 ms of full width at half maximum (FWHM) and two values for the rod maximal power in the fuel. These values comprise:

- a low value to avoid departure from nucleate boiling (DNB) occurrence;
- a high value to provoke DNB occurrence;

The axial and radial power profiles in the fuel are assumed to be flat.

Finally, the initial helium pressure in free volume is increased in one case.

A total of ten cases were defined in Table 3-1, with an increasing degree of complexity to assess the different phenomena gradually. Case 1 is mainly devoted to the thermal behaviour. Cases 2 and 3 are focused on the thermo-mechanical behaviour, and in Cases 4-9 the thermal-hydraulics behaviour aspect is added.

In all cases, starting from ambient conditions, a stabilisation phase is simulated before the real transient phase in order to reach the foreseen initial state of the rod.

For each code, it is recommended to use the standard options for all models except for the failure model, fuel relocation model, and high temperature cladding oxidation model, which must be disabled (considering the proposed problems). In Case 10, thermal and thermo-mechanical properties/models for cladding and fuel should be as close as possible to those of FRAPTRAN.
### Table 3-1: Summary of simplified cases

<table>
<thead>
<tr>
<th>Geometry</th>
<th>Contact conditions</th>
<th>Thermal mechanical models</th>
<th>Thermal-hydraulics conditions</th>
<th>Pmax</th>
<th>Helium pressure</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>No gap</td>
<td>Open gap</td>
<td>Bonding</td>
<td>Slipping</td>
<td>Standard</td>
</tr>
<tr>
<td>Thermal</td>
<td>Case 1</td>
<td>X</td>
<td>X</td>
<td>X</td>
<td>X</td>
</tr>
<tr>
<td></td>
<td>Case 2</td>
<td>X</td>
<td>X</td>
<td>X</td>
<td>X</td>
</tr>
<tr>
<td></td>
<td>Case 3</td>
<td>X</td>
<td>X</td>
<td>X</td>
<td>X</td>
</tr>
<tr>
<td></td>
<td>Case 10</td>
<td>X</td>
<td>X</td>
<td>X</td>
<td>X</td>
</tr>
<tr>
<td>Mechanical</td>
<td>Case 6</td>
<td>X</td>
<td>X</td>
<td>X</td>
<td>X</td>
</tr>
<tr>
<td></td>
<td>Case 7</td>
<td>X</td>
<td>X</td>
<td>X</td>
<td>X</td>
</tr>
<tr>
<td></td>
<td>Case 4</td>
<td>X</td>
<td>X</td>
<td>X</td>
<td>X</td>
</tr>
<tr>
<td></td>
<td>Case 5</td>
<td>X</td>
<td>X</td>
<td>X</td>
<td>X</td>
</tr>
<tr>
<td></td>
<td>Case 8</td>
<td>X</td>
<td>X</td>
<td>X</td>
<td>X</td>
</tr>
<tr>
<td></td>
<td>Case 9</td>
<td>X</td>
<td>X</td>
<td>X</td>
<td>X</td>
</tr>
</tbody>
</table>

#### 3.2.2. Second activity: Uncertainty and sensitivity analysis

Case 5 of the first activity was chosen as the numerical reference case for uncertainty and sensitivity analysis. The detailed specifications can be found in [5].

First, all relevant input uncertainties were identified and classified into four categories:

- uncertainties in the fuel rod design, bounded by allowable manufacturing tolerances;
- thermal-hydraulics boundary conditions;
- core power boundary conditions;
- physical properties/key models.

As shown in Table 3-2, for each input parameter, a mean value, a standard deviation, and a type of distribution was specified. To avoid unphysical numerical values, a range of variation (lower and upper bounds) was also provided. To simplify the current benchmark application, a normal distribution was assigned to all the considered input parameters. The sampling was performed between the upper and lower bounds, i.e. the probability density functions (PDFs) were truncated. Note that all the input uncertainties were considered as statistical or random. The identification and treatment of epistemic uncertainties, if any, was beyond the scope of the current project.
## Table 3-2: Summary of input parameter uncertainty and distribution

<table>
<thead>
<tr>
<th>Input uncertainty parameter</th>
<th>Distribution</th>
<th>Mean</th>
<th>Standard deviation</th>
<th>Type</th>
<th>Lower bound</th>
<th>Upper bound</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>1. Fuel rod manufacturing tolerances</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Cladding outside diameter (mm)</td>
<td></td>
<td>9.40</td>
<td>0.01</td>
<td>Normal</td>
<td>9.38</td>
<td>9.42</td>
</tr>
<tr>
<td>Cladding inside diameter (mm)</td>
<td></td>
<td>8.26</td>
<td>0.01</td>
<td>Normal</td>
<td>8.24</td>
<td>8.28</td>
</tr>
<tr>
<td>Fuel theoretical density (kg/m³ at 20°C)</td>
<td></td>
<td>10.970</td>
<td>50</td>
<td>Normal</td>
<td>10.870</td>
<td>11.070</td>
</tr>
<tr>
<td>Fuel porosity (%)</td>
<td></td>
<td>4</td>
<td>0.5</td>
<td>Normal</td>
<td>3</td>
<td>5</td>
</tr>
<tr>
<td>Cladding roughness (µm)</td>
<td></td>
<td>0.1</td>
<td>1</td>
<td>Normal</td>
<td>10⁻⁶</td>
<td>2</td>
</tr>
<tr>
<td>Pellet roughness (µm)</td>
<td></td>
<td>0.1</td>
<td>1</td>
<td>Normal</td>
<td>10⁻⁶</td>
<td>2</td>
</tr>
<tr>
<td>Filling gas pressure (MPa)</td>
<td></td>
<td>2.0</td>
<td>0.05</td>
<td>Normal</td>
<td>1.9</td>
<td>2.1</td>
</tr>
<tr>
<td><strong>2. Thermal-hydraulics boundary conditions</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Coolant pressure (MPa)</td>
<td></td>
<td>15.500</td>
<td>0.075</td>
<td>Normal</td>
<td>15.350</td>
<td>15.650</td>
</tr>
<tr>
<td>Coolant inlet temperature (°C)</td>
<td></td>
<td>280</td>
<td>1.5</td>
<td>Normal</td>
<td>277</td>
<td>283</td>
</tr>
<tr>
<td>Coolant velocity (m/s)</td>
<td></td>
<td>4.00</td>
<td>0.04</td>
<td>Normal</td>
<td>3.92</td>
<td>4.08</td>
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<tr>
<td><strong>3. Core power boundary conditions</strong></td>
<td></td>
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<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Injected energy in the rod (Joule)</td>
<td></td>
<td>30 000</td>
<td>1 500</td>
<td>Normal</td>
<td>27 000</td>
<td>33 000</td>
</tr>
<tr>
<td>Full width at half maximum (ms)</td>
<td></td>
<td>30</td>
<td>5</td>
<td>Normal</td>
<td>20</td>
<td>40</td>
</tr>
<tr>
<td><strong>4. Physical Properties/Key models</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Fuel thermal conductivity model (Mult. Coef.)</td>
<td></td>
<td>1.00</td>
<td>5%</td>
<td>Normal</td>
<td>0.90</td>
<td>1.10</td>
</tr>
<tr>
<td>Cladding thermal conductivity model (Mult. Coef.)</td>
<td></td>
<td>1.00</td>
<td>5%</td>
<td>Normal</td>
<td>0.90</td>
<td>1.10</td>
</tr>
<tr>
<td>Fuel thermal expansion model (Mult. Coef.)</td>
<td></td>
<td>1.00</td>
<td>5%</td>
<td>Normal</td>
<td>0.90</td>
<td>1.10</td>
</tr>
<tr>
<td>Cladding thermal expansion model (Mult. Coef.)</td>
<td></td>
<td>1.00</td>
<td>5%</td>
<td>Normal</td>
<td>0.90</td>
<td>1.10</td>
</tr>
<tr>
<td>Cladding Yield stress (Mult. Coef.)</td>
<td></td>
<td>1.00</td>
<td>5%</td>
<td>Normal</td>
<td>0.90</td>
<td>1.10</td>
</tr>
<tr>
<td>Fuel enthalpy/heat capacity (Mult. Coef.)</td>
<td></td>
<td>1.00</td>
<td>1.5%</td>
<td>Normal</td>
<td>0.97</td>
<td>1.03</td>
</tr>
<tr>
<td>Cladding-to-coolant heat transfer (Mult. Coef. - Same Coef. applied for all flow regimes)</td>
<td></td>
<td>1.00</td>
<td>12.5%</td>
<td>Normal</td>
<td>0.75</td>
<td>1.25</td>
</tr>
</tbody>
</table>
The probabilistic input uncertainty propagation method was selected for the uncertainty analysis due to its simplicity, robustness and transparency ([8], [9]). The sample size was set to 200. That is, each input parameter was sampled 200 times by Simple Random Sampling (SRS) according to the selected PDFs (normal distribution in this benchmark) and assuming independence between input parameters [10]. Two hundred code input models were constructed using the sampled input uncertainty values, and two hundred code simulations were performed, leading to a sample of the same size for each specified output quantity of interest.

The uncertainty analysis step was focused on a percentile estimation of each output quantity of interest. More precisely, due to the limited size (200), the output sample was used to derive a lower bound of the 5% percentile and an upper bound of the 95% percentile. The order statistics [16], which is a well-established and shared statistical technique in the nuclear community ([8], [9]), was recommended for this analysis. It makes it possible to determine the bounds from the sorted values of the output sample (5th and 196th for a sample of 200), without any assumption on the output distribution.

Finally, a sensitivity analysis was performed using the 200 code runs previously obtained. The (Spearman) partial rank correlation coefficients (PRCC) were chosen for this activity with multiple regressors (input parameters) which might have non-linear but monotonic relation with the output quantity of interest [11]. Note that in this case, for a given regressor, the correlation is not evaluated with the response, but with the response from which the part explained by all the other regressors has been removed.

After the correlation coefficients were computed, the identification of the most influential input parameters was based on the introduction of a significance threshold. In this benchmark, the significance threshold is set to 0.25, which corresponds to a high confidence level, that is >99.9%, or, in other words, to a very low probability of rejecting the assumption of linear independence (i.e. \( P(\text{R}>0.25) < 0.001 \), where \( \text{R} \) denotes the correlation coefficient). An input parameter is considered influential as soon as its correlation coefficient is in absolute value larger than that threshold [6].

The participants were asked to provide the lower and upper bounds associated with all specified output parameters for uncertainty analysis, as well as the PRCCs associated to each uncertain input for each specified output parameter at each specified time and for their maximum values for sensitivity analysis.

### 3.3. Summary of results of Phase II

#### 3.3.1. First activity: Simplified cases

The objective of the first activity of RIA benchmark Phase II was to compare the results of different simulations on simplified cases, in order to better understand the differences in modelling of the concerned specific phenomena. Nonetheless, this understanding is limited by the semi-empiric nature of many codes, which rely on parameters determined when compared to integral tests.

As shown in Table 1.1, 15 organisations representing 12 countries have provided analyses for some or all the cases that were defined, using a wide spectrum of computer codes including ALCYONE, BISON, FRAPTRAN, RANNS, SCANAIR, TESPAROD, and TRANSURANUS.
The results were compared and discussed in detail in the NEA report [4]. An example of the radial average enthalpy variations is shown in Figure 3-2 for PWR Case 4 (upper figure) and Case 5 (lower figure). The agreement is very good for Case 4 (PWR Case without DNB occurrence) during the whole transient, but significant differences appear during the cooling phase for Case 5 (PWR case with DNB occurrence).

Figure 3-2: **Radial average enthalpy variations for PWR Case 4 (upper figure) and 5 (lower figure)**

![Case_4 - Variation of Radial Average Enthalpy (DHR)](image1)

![Case_5 - Variation of Radial Average Enthalpy (DHR)](image2)

Source: [4].
The same conclusions can be drawn about the fuel centreline temperature variations, and the maximum values for fuel enthalpy [4].

The variations in the cladding outer temperature are shown in Figure 3-3 for PWR Case 4 (upper figure) and Case 5 (lower figure). The agreement is very good for Case 4 but not so good for Case 5. In Case 5, most of the calculations show DNB occurrence, followed by a post-DNB phase with very high cladding temperatures that ends with a quenching phase. If the physical trend is very similar for all calculations, the maximum temperature reached by the cladding varies from ~500°C to ~1 040°C and boiling duration ranges from 0.3 s to ~10 s.

Figure 3-3: Variation of the cladding outer temperature for Case 4 (upper figure) and Case 5 (lower figure)

Source: [4].
The lower, upper, and mean value (extracted from all simulations) of maximum values for cladding temperature and boiling duration are presented for all cases in Figure 3-4. The scatter is very high for cases where coolant boiling occurs. In the worst case, the maximum cladding temperature ranges from 669°C to 1311°C while the boiling duration ranges from almost 0 s to 28 s. This last value can reach almost 100 s if one takes into account simulations that failed.

Figure 3-4: The lower, upper and mean value (extracted from all simulations) of maximum values for cladding temperature (upper figure) and boiling duration (lower figure)

As discussed in the NEA report [4], boiling under RIA conditions is known to be significantly different from boiling under steady-state conditions. Some codes assume that the steady-state correlations are applicable to RIA conditions while other codes use specific fast transient correlations (for critical heat flux, heat transfer in film boiling, rewetting conditions, etc.). In addition, as boiling in RIA conditions have not been extensively studied until now, specific fast transient correlations have still to be developed and validated for BWR and PWR conditions.

In conclusion, the agreement between all simulations for cladding temperature is rather good for cases with no boiling crisis, and very poor when two-phase flow conditions are met. Simulations with codes having specific fast transient correlations seem to provide more credible results, but those codes still have to be validated in real reactor conditions.
Finally, the poor agreement obtained in some cases on cladding temperature can partially explain the discrepancies in the fuel temperature, cladding hoop strain, and fuel/cladding elongation observed in those cases (see below).

Examples of cladding total hoop strain evolution are given in Figure 3-5 for Case 1 (upper figure) and Case 5 (lower figure). They include, respectively, similar cases with and without thermal conditions imposed.

Figure 3-5: **Cladding total hoop strain evolution for Case 1 (upper figure) and Case 5 (lower figure)**

![Figure 3-5: Cladding total hoop strain evolution for Case 1 (upper figure) and Case 5 (lower figure)](image)

The behaviour for these two cases is qualitatively similar in all simulations with the different codes: a maximum value is reached during the beginning of the transient when the gap is closed and a residual hoop strain is reached after the gap opening. The long-term behaviour displays
some differences because the gap opening time changes a lot among the benchmarked codes. It seems that the scatter in the long-term behaviour is more significant in Case 5 (compared to Case 1): in this case, as the thermal behaviour of the cladding is not imposed, larger cladding temperature differences lead to divergent cladding mechanical behaviours.

The fuel total elongations for Case 1 and Case 5 are given in Figure 3.6. The fuel elongation evolutions are very similar for all cases because the main contributing factor is the thermal expansion driven by fuel temperature evolutions, which are very similar in all simulations. The difference between lower and upper values for the maximal fuel elongation estimation is between 15% and 75% (of the mean value).

Figure 3-6: Evolution of fuel stack total axial elongation for Case 1 (upper figure) and Case 5 (lower figure)

Source: [4].
The cladding total elongation evolution for Case 1 is given in Figure 3.7 (upper figure), and the maximum values for all cases in Figure 3.7 (lower figure). For cases where no slipping between the fuel and the cladding is assumed (when the gap is closed), the cladding elongation follows the fuel one, and the maximum values for cladding elongation and fuel elongation are relatively similar. Lower values for cladding elongation are observed for Cases 2, 3 and 10 where an initial gap is assumed. As a consequence, the difference between lower and upper values for the maximal cladding total elongation is between 20% and 75% (of the mean value) except for Case 7.

Figure 3-7: **Evolution of cladding total axial elongation for Case 1 (upper figure) and the maximum values for all cases (lower figure)**

Source: [4].
3.3.2. Second activity: Uncertainty and sensitivity analysis

The second activity of the NEA Working Group on Fuel Safety (WGFS) RIA fuel codes benchmark Phase II was focused on the uncertainty assessment of the Case 5 calculation results. In particular, the impact of the initial state and key models on the results of the transient was investigated. In addition, a sensitivity study was performed to identify or confirm the most influential input parameters.

As shown in Table 1.1, 14 organisations representing 12 countries provided: the lower and upper bounds associated with all specified output parameters for uncertainty analysis; the partial rank correlation coefficients associated to each uncertain input for each specified output parameter at each specified time; and their maximum values for sensitivity analysis.

The simulations and analyses were performed using various computer codes, including ALCYONE, BISON, FRAPTRAN, RANNS, SCANAIR, TESPA-ROD, and TRANSURANUS, and various statistical uncertainty and sensitivity analysis tools (DAKOTA, SUNSET, SUSA, URANIE/ROOT, Python & Scipy).

The results supplied by all participants were used in synthesis analyses performed by the IRSN according to the commonly agreed methodology.

During the power deposit, there is a very good agreement between all the participants on fuel thermal behaviour. For example, concerning the reference calculations, the discrepancy for the maximal fuel centreline temperature (TFC) is lower than 40°C (see Figure 3.8, upper figure). The TFC uncertainty assessment with respect to time is also quite consistent between all the participants. During the power deposit, the uncertainty band width can be very high, around 800°C (see Figure 3.8, lower). This result is linked to the fact that the uncertainty on the pulse width is very large (+/-10 ms). Thus, even with the same final injected energy, the injected energy 40 ms after the beginning of the transient can vary by a factor of two in the two extreme cases of pulse half width (20 ms and 40 ms) that can lead to large thermal differences during the transient. Just after the power deposit, the TFC uncertainty is significantly lower, 200-300°C, which is 10-15% of the TFC reference value (~1 850°C).

The same “pulse width uncertainty effect” can be seen on other parameters during the transient: very large during the power deposit, and much smaller after the pulse (see Figure 3.9, upper figure, for the cladding hoop strain, ECTH). One can notice that the uncertainty is rather significant for mechanical parameters during the pulse (see Figure 3.9, lower figure).

After the power deposit, all participants calculate boiling at the cladding-to-coolant interface. The thermo-mechanical behaviour of the rod will mainly depend on the cladding-to-coolant heat transfer coefficient and on the quenching time. The uncertainty evolutions of the cladding thermal and mechanical results are represented in Figure 3-10. The uncertainty is still large after the power deposit for all the participants, both for thermal (cladding outer temperature, TCO) and mechanical output (cladding hoop stress, SCH). The discrepancies between participants are significant, mainly due to different modelling of the heat transfer during the boiling phase and to significant difference in quenching time. SCH uncertainty is also affected by the mechanical approach (i.e. different yield stress laws).

Figure 3.11 shows the reference calculation dispersion and the global uncertainty width for fuel enthalpy variation (DHR) and TCO. The global uncertainty width is close to the reference calculation dispersion. The code effect is very significant here. As illustrated for TCO on Figure 3.11 (lower figure), for several outputs the reference calculation dispersion is larger than the typical uncertainty resulting from input parameter variations calculated by each participant individually.
Figure 3-8: Temperature of fuel centreline (TFC) reference calculations (upper figure) and uncertainty bandwidths (lower figure)

Source: [6].
Figure 3-9: Cladding hoop strain (ECTH) reference calculations (upper figure) and uncertainty bandwidths (lower figure)

Source: [6].
Figure 3-10: **Cladding outer temperature (TCO, upper figure) and cladding hoop stress (SCH, lower figure) uncertainty bandwidths**

Source: [6].
Figure 3.11: Fuel enthalpy variation (DHR, upper figure) and cladding outer temperature (TCO, lower figure) uncertainty widths

Source: [6].

Figure 3.12 displays the relative uncertainty interval associated with each scalar output parameter related to maximum values and boiling duration taking account of the contributions of all participants. The bounds of this interval have been obtained by dividing the minimum (resp. maximum) of all lower (resp. upper) bounds given by participants by the average of all reference calculations. The results show that the relative uncertainty interval width (called relative uncertainty for simplicity) depends on the type of parameter under consideration. More precisely, the narrowest relative uncertainty intervals are obtained for fuel thermal behaviour.

1. Note that, by construction, this quantity takes into account the dispersion between all participants’ contributions and therefore includes results obtained by codes with different degrees of maturity.
outputs (DHR, TFC and TFO, with a factor less than 1.5 between upper and lower bounds). The uncertainty interval width increases slightly for fuel mechanical outputs and cladding elongation (RFO, EFT1 and ECT, with a factor between 2 and 2.5) and more significantly for other cladding mechanical outputs and cladding temperature (ECTH, SCH and TCO). Concerning this last group, SCH exhibits a larger uncertainty (a factor ~5) than ECTH (a factor ~2.5). Finally, the largest uncertainty interval is observed for the boiling duration (a factor ~30).

Figure 3-12: **Relative global uncertainty interval associated with each output associated with maximum values and boiling duration**

![Relative global uncertainty interval](image)

*Source: [6]*.

In order to evaluate the level of agreement between participants' contributions, a conflict indicator for each output was built. A large value of this indicator means a strong disagreement between at least two participants. It appears that the results are highly conflicting, except for fuel thermal behaviour (see Figure 3.13).

Figure 3-13: **Conflict indicator associated with each output (maximum values and boiling duration)**

![Conflict indicator](image)

*Source: [6]*.
However, by construction, this conflict indicator is equal to one as soon as two participants fully disagree (i.e. empty intersection between their uncertainty intervals). It was therefore interesting to go deeper in the analysis by investigating the reasons that may explain this lack of coherence. Three levels of coherence were identified:

- **High coherence**: it includes participants’ results that are in strong agreement for uncertainty intervals and reference calculations. It corresponds to fuel thermal behaviour outputs (DHR, TC, and TFO).

- **Low coherence**: it concerns participants’ results with a conflict indicator equal to 1 but exhibiting coherent reference calculations and uncertainty intervals for a large majority of participants, (i.e. the empty intersection is due to few participants that do not agree with the others). It is the case for fuel and cladding mechanical behaviour outputs except for SCH.

- **No coherence**: it concerns participants’ results with a conflict indicator equal to 1 where the incoherence cannot be reduced by removing just a few contributions. It corresponds to thermal-hydraulics behaviour outputs (TCO, Boiling duration) and SCH. This strong lack of coherence can be explained by the combination of a large dispersion of reference calculations and narrow uncertainty bands, which is particularly noticeable for TCO and SCH.

In conclusion, the uncertainty analysis has resulted in the following main observations:

- The pulse width uncertainty has a strong effect on the uncertainty results during the transient.

- The uncertainty bandwidth is similar for all codes in the case of fuel thermal behaviour outputs and a high coherence is obtained for those parameters (participants’ results are in strong agreement for uncertainty intervals and reference calculations).

- Large uncertainty bandwidth is observed for cladding mechanical and thermal behaviour outputs and fuel mechanical behaviour outputs.

- A low coherence is observed for fuel and cladding mechanical behaviour outputs (except for stresses in the clad), but coherent reference calculations and uncertainty intervals are obtained for a large majority of participants.

- No coherence is observed for cladding stress and thermal-hydraulics behaviour outputs due to the combination of a large dispersion of reference calculations and narrow uncertainty bands for those parameters.

Table 3-3 summarises the sensitivity analysis performed for the maximum values of the main output parameters of interest. A coloured bar means that more than 50% of the participants have identified the corresponding input parameter as influential for the type of behaviour.

The following observations were made:

- The injected energy and fuel enthalpy have been identified as influential for all types of behaviours (cladding and fuel thermal and mechanical behaviours) by a majority of participants.

- Input parameters related to the rod geometry (pellet and cladding roughness, cladding inner diameter) as well as fuel thermal expansion model and full width at half maximum were also identified as influential by this study.

- On the contrary, several input parameters (cladding thermal expansion and conductivity model, coolant velocity and pressure, filling gas pressure, fuel porosity, fuel theoretical density and cladding outside diameter) have been considered as influential by a very low percentage of participants.

- There are more influential parameters for cladding mechanical behaviour than for the other types of behaviour.
Table 3-3: Influential input parameters with respect to the type of behaviour when focusing on maximum values

<table>
<thead>
<tr>
<th></th>
<th>Fuel thermal (DHR, TFC)</th>
<th>Clad thermal (TCO)</th>
<th>Clad mechanical (ECTH, SCH)</th>
<th>Fuel mechanical (EFTI, RFO)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cladding outside diameter</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Cladding inside diameter</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Fuel theoretical density</td>
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<tr>
<td>Fuel porosity</td>
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<td></td>
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<tr>
<td>Cladding roughness</td>
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</tr>
<tr>
<td>Fuel roughness</td>
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<tr>
<td>Filling gas pressure</td>
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</tr>
<tr>
<td>Coolant pressure</td>
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<tr>
<td>Coolant inlet temperature</td>
<td></td>
<td></td>
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<td></td>
</tr>
<tr>
<td>Coolant velocity</td>
<td></td>
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<tr>
<td>Injected energy in the rod</td>
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<tr>
<td>Full mid height width</td>
<td></td>
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</tr>
<tr>
<td>Fuel thermal conductivity model</td>
<td></td>
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<tr>
<td>Clad thermal conductivity model</td>
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<tr>
<td>Fuel thermal expansion model</td>
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<tr>
<td>Clad thermal expansion model</td>
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<tr>
<td>Clad Yield stress</td>
<td></td>
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<tr>
<td>Fuel enthalpy</td>
<td></td>
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<tr>
<td>Clad to coolant heat transfer</td>
<td></td>
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</tr>
</tbody>
</table>

3.4. Conclusions and recommendations of Phase II

By comparing the results provided by participants, it has been possible to draw the following conclusions:

- Using simplified cases with fresh fuel leads to very close evaluations of the initial state of the rod (just before the pulse), which was not the case previously during the RIA benchmark Phase I.

- With respect to the fuel thermal behaviour, the differences in the estimation of fuel enthalpies and temperatures are rather limited, especially for the maximum values of these parameters.

- However, the agreement is worse for BWR thermal-hydraulics conditions than PWR thermal-hydraulics conditions that lead to water boiling. This seems to be mainly driven by uncertainty in the cladding-to-coolant heat transfer.

- Concerning cladding temperatures, considerable scatter is obtained for the cases where water boiling occurs. This scatter is clearly related to the cladding-to-coolant heat transfer modelling.

- Boiling in RIA conditions is known to be significantly different from in steady-state conditions. Some codes assume that the steady-state correlations are applicable to RIA conditions while other codes use specific fast transient correlations (for critical heat flux, heat transfer in film boiling, rewetting conditions, etc.). Given the lack of sufficient experimental investigation on boiling in RIA conditions, no sound recommendation can be made as for which correlations are the most suitable ones to use.
From cases devoted to BWR conditions, it is clear that very few (if any) of the applied computer codes are able to handle the thermal-hydraulics conditions expected in a BWR RIA with large energy injection at cold zero power conditions. This is not simply a question of uncertainties in the cladding-to-coolant heat transfer modelling; the excessive steam generation expected in the fuel assembly at atmospheric pressure can obviously not be handled by the simple thermal-hydraulics models in the codes.

With respect to mechanical behaviour, the loading mode of the cladding considered during this benchmark exercise is limited to the pellet-cladding mechanical interaction (PCMI) one.

Although the general behaviour is similar from one case to another, and although the agreement between predictions is reasonable during the heating phase, significant discrepancies are obtained for the maxima of different variables of interest (namely cladding hoop strain, fuel and cladding elongation).

The first activity of the WGFS RIA Fuel Rod Codes Benchmark Phase II (simplified cases) led to the following recommendations:

- Fuel and cladding thermo-mechanical models (with the associated material properties) should be further improved and validated more extensively against a sound RIA database.
- A comprehensive and robust database consisting of both separate effect tests and integral tests should be built up in the short term. In this way, both individual model validation and model integration into codes would be feasible. The database could be shared by the modellers, whenever possible, to ease the comparison of simulation results from various codes.
- The cladding-to-coolant heat transfer in the case of water boiling during very fast transients is of particular interest, and capabilities related to modelling this phenomenon should be improved. To achieve this target, more separate effect tests and experiments seem necessary.
- Models related to the evolution of the pellet-cladding gap should be improved and validated for RIA conditions as this has been shown to have a significant effect on fuel rod response. To reach this objective, in-reactor measurements of cladding strain during RIA simulation tests should be done (or at least attempted).

The second activity (uncertainty and sensitivity analysis) confirmed the conclusions from the first activity but also offered some new insights:

- Regarding fast transient thermal-hydraulics post-DNB behaviour, there are major differences between the modelling approaches, resulting in significant deviations between simulations. Unfortunately, there are currently no simple and representative experimental results for the assessment of the different approaches.
- The models of fuel and cladding thermo-mechanical behaviour and the associated materials properties should be improved and validated in RIA conditions.
- Different influential input parameters were identified for fresh fuel. For instance, injected energy, fuel enthalpy, parameters related to the rod geometry (pellet and cladding roughness, cladding inside diameter), fuel thermal expansion model and the pulse full width at half maximum come out as influential regarding the maximum value of each output parameter of interest. However, the parameters with significant influence on the results for irradiated fuel could be different.
- Uncertainties cannot fully explain the scatter observed in the first activity results and during Phase I of this benchmark exercise.
- The specifics of the applied codes and the user effects could play a more important role than the uncertainties.
Based on the conclusions summed up above, the following recommendations were made:

- To reduce user effect and code effect, the code development teams should provide recommendations regarding the version of their codes to be used, the models and also the numerical parameters to be used (mesh size, time step, etc.) as much as possible.

- A complement of the RIA benchmark should be launched. This activity should be limited in time and should be focused on uncertainty and sensitivity analyses on an irradiated case, in order to identify the corresponding influential input parameters. In particular, uncertainties regarding fission gases distribution, fuel microstructure, cladding corrosion state, and gap conductance should be investigated.

- This information on the most influential input parameters for an irradiated case will be useful when possibly establishing the Phenomena Identification and Ranking Table (PIRT) for RIA and could guide future RIA tests and code improvements.

- Co-operation between existing experimental teams should be established in the area of cladding-to-coolant heat transfer during very fast transients as well as in the areas relevant to fuel and cladding thermo-mechanical modelling. The main objectives of this co-operation should be to collect and share all existing data and to formulate specific proposals in order to reduce the lack of knowledge and achieve common understanding on the subjects. This activity should address both out-of-pile and in-pile tests.
4.1. Objectives of the Phase III

One recommendation from the benchmark Phase II was to launch a complementary phase focused on uncertainty and sensitivity analyses on an irradiated case, in order to identify the corresponding influential input parameters, which may be different from those identified for fresh fuel. In particular, uncertainties regarding fission products distribution, fuel microstructure, cladding corrosion state, and gap conductance should be investigated.

Considering the results obtained during the first two phases, mainly the large discrepancy regarding RIA thermal-hydraulics modelling, it was chosen to limit the exercise to the "PCMI phase", for which data are available from in-pile tests. Therefore, it was proposed to use the CIP0-1 test for this exercise. This test was performed in the CABRI facility (Cadarache, France) in 2002 in the sodium loop (for which it should be easier to reach a consensus between the codes in relation to cladding-coolant heat transfer) on an irradiated UO\textsubscript{2} fuel with ZIRLO\textsuperscript{TM} cladding. As this test case was also analysed in Phase I, the main test characteristics are found in Section 2.2.

This phase was focused on the assessment of the uncertainty of the results. In particular, the impact of the initial state and key models on the results of the transient were investigated. In addition, a sensitivity study was performed to identify or confirm the most influential input uncertainties.

4.2. Specifications of Phase III

The assessment of the uncertainty associated with the test prediction is affected by the simulation of the base irradiation, which determines the rodlet pre-test conditions. In order to focus this activity on the transient codes, the state of the rod at the end of the base irradiation, before the transient, was evaluated with the FRAPCON V4.0 Patch 1 code, whose input data and results were distributed to all participants. As an example, Figure 4-1 shows some FRAPCON results provided.

Figure 4-1: Examples of FRAPCON radial profile output at PPN location (power and total gas retained)

The input parameters were classified into three categories:

- end-of-life fuel state;
- core power boundary conditions;
- physical properties/key models.
Table 4-1 provides the 12 specified input parameters as well as the information related to their uncertainty. In most cases, a normal distribution has been assigned to input parameters for simplicity’s sake. Their standard deviation has been taken as the half of the maximum of the absolute value of the difference between their nominal value and their upper or lower bound.

After probabilistic input uncertainty propagation, each participant gave lower and upper bounds associated with all time/height trended output parameters. Similarly to Phase II, these bounds, denoted LUB and UUB, are associated to the estimation of a lower, resp. upper, bound of the 5%, resp. 95%, percentiles at confidence level higher than 95%.

In addition, the results of the calculation with the nominal value of the input parameters, also called reference calculation, were provided. The reference, lower, and upper bound values following the previous method, for the maximum of each output parameter, were given.

Finally, regarding sensitivity analysis, participants evaluated the partial rank correlation coefficients (or Spearman’s if PRCC were not available) associated to each uncertain input parameter and for each output of interest.

Table 4-1: List of input parameter uncertainty for statistical uncertainty analysis

<table>
<thead>
<tr>
<th>Input parameter uncertainty</th>
<th>Distribution</th>
<th>Reference/mean value</th>
<th>Standard Deviation</th>
<th>Type</th>
<th>Lower bound</th>
<th>Upper bound</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>1. End-of-life fuel state</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Pellet-cladding radial gap (at 20 °C) (µm)</td>
<td>10.</td>
<td>uniform</td>
<td>0.1</td>
<td></td>
<td>20</td>
<td></td>
</tr>
<tr>
<td>Cladding roughness (µm)</td>
<td>0.1</td>
<td>uniform</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Pellet roughness (µm)</td>
<td>0.1</td>
<td>uniform</td>
<td>10^{-6}</td>
<td></td>
<td>2.</td>
<td></td>
</tr>
<tr>
<td>Zirconia thickness (multiplying coefficient C)</td>
<td>Half of the upper value ( C =1/2)</td>
<td>uniform</td>
<td>Total spalling (C=0.)</td>
<td></td>
<td>Before test measurement (C=1.)</td>
<td></td>
</tr>
<tr>
<td><strong>2. Core power boundary conditions</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Injected energy in the rod at PPN location (cal/g)</td>
<td>99</td>
<td>3</td>
<td>Normal</td>
<td>93</td>
<td>105</td>
<td></td>
</tr>
<tr>
<td>Radial power profile (<em>peaking factor</em> PF)</td>
<td>FRAPCON results (i.e. 0)</td>
<td>0.125</td>
<td>Normal</td>
<td>-0.25</td>
<td>0.25</td>
<td></td>
</tr>
<tr>
<td>Pulse width (ms)</td>
<td>Measurement 32.</td>
<td>0.75</td>
<td>Normal</td>
<td>30.5</td>
<td>33.5</td>
<td></td>
</tr>
<tr>
<td><strong>3. Physical Properties/Key models</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Fuel thermal conductivity model (multiplying coefficient)</td>
<td>1.00</td>
<td>5%</td>
<td>Normal</td>
<td>0.90</td>
<td>1.10</td>
<td></td>
</tr>
<tr>
<td>Fuel thermal expansion model (multiplying coefficient)</td>
<td>1.00</td>
<td>5%</td>
<td>Normal</td>
<td>0.90</td>
<td>1.10</td>
<td></td>
</tr>
<tr>
<td>Fuel enthalpy / heat capacity (multiplying coefficient)</td>
<td>1.00</td>
<td>1.5%</td>
<td>Normal</td>
<td>0.97</td>
<td>1.03</td>
<td></td>
</tr>
<tr>
<td>Cladding thermal expansion model (multiplying coefficient)</td>
<td>1.00</td>
<td>5%</td>
<td>Normal</td>
<td>0.90</td>
<td>1.10</td>
<td></td>
</tr>
<tr>
<td>Cladding yield stress (multiplying coefficient)</td>
<td>0.9</td>
<td>5%</td>
<td>Normal</td>
<td>0.81</td>
<td>0.99</td>
<td></td>
</tr>
</tbody>
</table>

Source: [7].

4.3. Summary of results of Phase III

Detailed analyses of the results can be found in the report [7]. The main results are summarised here.

4.3.1. Rod state initialisation

The participants in the RIA benchmark Phase III, which represented nine countries, are listed in Table 1-1 as well as the codes they used for base irradiation, transient simulation, and sensitivity/uncertainty analyses. In terms of computer fuel rod codes used, the spectrum was large for base irradiation and transient simulations.
Some participants fully initialised their CIP0-1 transient test analysis with the output from the FRAPCON results. For the other data not provided by FRAPCON and needed for RIA calculations, hypotheses were provided.

Those participants who could not fully initialise their CIP0-1 input data with FRAPCON output used their own means of calculating the non-initialisable parameters, and tried to match as much as possible the FRAPCON results, notably fission gas distribution.

As an illustration, the intragranular and grain boundary gas distributions, as given by seven participants, are plotted in Figure 4-2 and compared to the FRAPCON output data. While the grain boundary gas distributions are quite similar for all participants and consistent with the FRAPCON output, this is not the case for intragranular gas distributions.

**Figure 4-2: Intragranular (upper figure) and grain boundary (lower figure) gas radial distribution for different participants**

Source: [7].
4.3.2. **Time/height trend results**

This analysis is only focused on the outputs for which experimental information is available. Among the outputs, experimental information is available for the sodium coolant temperature (TNa1, TNa2), cladding elongation (ECT), cladding residual hoop strain (ETZ) and fission gas release (FGR).

The comparison of participants’ contributions concerns both reference calculations and uncertainty results. For this last type of results, the evolution with respect to time of each uncertainty interval width defined by $q_0 = UUB - LUB$ is studied.

The coolant temperature calculations and measurements during the transient are represented in Figure 4-3.

Figure 4-3: **Sodium temperature TNa1 (z=25 cm/BFC) reference calculation vs measurements (upper figure) and uncertainty interval width (lower figure)**

![Sodium temperature TNa1 reference calculations](image1)

![Sodium temperature TNa1 uncertainty interval width](image2)

Source: [7].
The experimental measurement general trend is well captured by all participants even if the calculated coolant temperature increase is slightly too rapid and over-predicted for a majority of participants.

The dispersion of reference calculations at the beginning of the transient and on the peak temperature (~90°C) is large, especially for TNa2, which is measured at a higher axial location: the forced convection in sodium coolant is probably not correctly computed for some codes. In the second phase (time >1 s), the dispersion is rather low (<20°C).

As shown in Figure 4-4, there is some dispersion of the cladding permanent hoop strain calculations ETZ with a ratio of two between highest (~0.5%) and lowest value (~0.25%).

Nevertheless, even if participants’ calculations slightly underestimated the measurements, the calculations are consistent and close to the lower bound of the measurements.

**Figure 4-4: Cladding permanent hoop strain at the end of transient as a function of axial location**

Moreover when taking uncertainties into account, the average measurement has lower and upper uncertainty bounds for almost all participants along the entire axial height.

Concerning the cladding elongation (ECT), there is a good agreement between calculations and measurements (Figure 4.5, upper figure): the calculations are dispersed but remain in the uncertainty band for all participants almost all along the transient. The effect of input uncertainties on cladding elongation (ECT) is also shown in Figure 4.5, in the lower figure. Globally, when taking into account the uncertainty on input data for the calculations and the experimental measurement uncertainties, there is a good agreement between calculations and measurements.
Figure 4-5: Cladding elongation as a function of time, reference calculations (upper figure) and uncertainty band widths (lower figure).

Source: [7].
4.3.3. Scalar output

Figure 4.6 provides participants’ results for some scalar outputs.

Figure 4-6: Comparison of participants’ uncertainty results and reference calculation values for scalar outputs

Source: [7].
The outputs for which experimental information is available are analysed first.

Concerning sodium temperatures, a dispersion between reference calculations is noticeable (~90°C for TNa2). At most half of the participants have reference calculation that fall inside the experimental interval for these outputs. Taking uncertainty into account allows improving the results with a non-empty intersection for a majority of participants. However, the uncertainty interval widths are also dispersed (from 10°C to 70°C for TNa2).

When focusing on cladding mechanical behaviour, ECT and ETZ, the experimental uncertainty interval encompasses all cladding elongation reference calculations but less than 50% of the residual hoop strain reference calculations. When adding the uncertainty result, the intersection with the experimental interval is non-empty for both outputs leading to a higher coherence between participants’ contributions compared to sodium temperatures. However, it is important to mention that, contrarily to ETZ, the uncertainty interval width for ECT is always smaller than the width of the experimental one.

Finally, there is a very low coherence for FGR. This can be explained by the dispersion of reference calculations and narrow uncertainty intervals for a majority of participants. As already observed in Phase I, the fission gas release calculations are strongly dispersed, with one order of magnitude between the highest and lowest evaluation. Only one participant calculated a final fission gas release inside the experimental uncertainty band.
It is also interesting to analyse the coherence between participants’ contributions taking all outputs into account. It appears that the coherence between participants depends on the type of outputs. This point had already been observed in the RIA benchmark Phase II. More precisely, the uncertainty results exhibit four main levels of coherence:

- **High coherence**: it includes participants’ results that are, for a large majority, in agreement as regards to the uncertainty interval width and reference calculation. It corresponds to fuel thermal behaviour outputs (TFM, TFC and DHR) except the fuel outer temperature (TFO).

- **Medium coherence**: it concerns participants’ results that, for a large majority, exhibit a coherent uncertainty interval width but with dispersion in the reference calculations due to few participants’ contributions. It is the case for fuel and cladding mechanical behaviour output (ECT, ETZ), except for SCH.

- **Low coherence**: it corresponds to participants’ results exhibiting dispersion for both reference calculation and uncertainty interval width. It is the case for TFO, SCH, GAP as well as for cladding and fluid thermal outputs.

- **No coherence**: this last level is associated with large dispersion for both reference calculation and uncertainty interval width. It includes the remaining outputs: fuel-clad heat-exchange (HFC), FGR and clad failure prediction (CFP).

In the Phase II benchmark, only three levels of coherence were introduced. In Phase III, four levels are defined in order to take account of the results associated to the extra outputs HFC, FGR and CFP that exhibit strong disagreement. However, for the outputs considered in both phases, the same coherence ranking has been found.

Finally, the relative uncertainty interval for each output is compared in Figure 4-7.

**Figure 4-7: Relative uncertainty interval for each output**

![Relative uncertainty interval for each output](Source:[7]).
The results show that the relative uncertainty also depends on the type of parameter observed. More precisely, the narrowest intervals are obtained for fuel and fluid thermal behaviour outputs (DHR, TFC, TFM, TNa1, TNa2, and TFO to a less extent). The uncertainty interval width increases slightly for cladding thermal outputs (TCI, TCO), then more significantly for fuel and cladding mechanical ones (EFT, RFO, ECT, ECTH, SCH). Among cladding mechanical outputs, a large relative uncertainty can be observed for ETZ and a similar conclusion also holds for gap width (GAP), fission gas release (FGR), HFC and clad failure prediction (CFP).

The relative uncertainty was also evaluated in Phase II (see Figure 3-12). Even if some outputs are different, it appears that, as for fresh fuel, the relative uncertainty is lower for fuel thermal outputs. Moreover, there are globally larger relative uncertainties for irradiated fuels.

4.3.4. **Sensitivity analysis**

The most influential input parameters have been identified for each participant, based on the correlation coefficients they calculated and using a fixed significance threshold of 0.25.

The synthesis is achieved by evaluating, for each output, the percentage of participants that consider a given parameter as influential. This short calculation has been repeated during the benchmark for the four fixed times when the correlation coefficients were evaluated and for maximum values, as required in the specifications.

In order to draw conclusions, for practical issues, the results have been summarised with respect to groups of outputs corresponding to one type of behaviour (fuel thermal and mechanical behaviour, cladding thermal and mechanical behaviour, gap size, pellet-cladding heat transfer, fission gas release, and cladding failure), following the rule that if an input parameter is influential for an output associated with a given behaviour, it is considered as influential for the whole group. They are given in Table 4.2.

**Table 4-2: Influential input parameters with respect to the type of behaviour when focusing on the maximum value of each output**

<table>
<thead>
<tr>
<th></th>
<th>Fuel thermal (DHR, TFC, TFM, TFO)</th>
<th>Cladding thermal (TCI, TCO)</th>
<th>Fluid thermal (TNa1, TNa2)</th>
<th>Fuel mechanical (EFT, RFO)</th>
<th>Cladding mechanical (ECT, ECTH, SCH)</th>
<th>GAP</th>
<th>HFC</th>
<th>FGR</th>
<th>CFP</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pellet-cladding radial gap</td>
<td>37.5% 25.0% 14.3%</td>
<td>71.4% 87.5% 25.0%</td>
<td>60.0% 55.6% 30.0%</td>
<td>12.5% 12.5% 0.0%</td>
<td>66.7% 11.1% 22.2%</td>
<td>50.0% 25.0% 100.0%</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Cladding roughness</td>
<td>60.0% 60.0%</td>
<td>75.0% 62.5% 88.9%</td>
<td>50.0% 50.0% 12.5%</td>
<td>88.9% 66.7% 11.1%</td>
<td>25.0% 25.0% 50.0%</td>
<td>60.0% 10.0% 50.0%</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Pellet roughness</td>
<td>50.0% 50.0%</td>
<td>77.8% 75.0% 62.5%</td>
<td>50.0% 50.0% 12.5%</td>
<td>88.9% 66.7% 11.1%</td>
<td>25.0% 25.0% 50.0%</td>
<td>60.0% 10.0% 50.0%</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Zirconia thickness</td>
<td>55.6% 77.8% 75.0%</td>
<td>62.5% 88.9% 66.7%</td>
<td>60.0% 62.5% 88.9%</td>
<td>11.1% 11.1% 44.4%</td>
<td>25.0% 25.0% 50.0%</td>
<td>60.0% 10.0% 50.0%</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Injected energy in the rod</td>
<td>100.0% 100.0%</td>
<td>88.9% 100.0% 90.0%</td>
<td>100.0% 100.0% 100.0%</td>
<td>100.0% 100.0% 100.0%</td>
<td>80.0% 80.0% 83.3%</td>
<td>83.3% 83.3% 83.3%</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Radial power profile</td>
<td>87.5% 75.0% 85.7%</td>
<td>71.4% 37.5% 25.0%</td>
<td>71.4% 37.5% 25.0%</td>
<td>25.0% 25.0% 37.5%</td>
<td>60.0% 60.0% 60.0%</td>
<td>60.0% 60.0% 60.0%</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Power pulse width</td>
<td>77.8% 44.4%</td>
<td>50.0% 33.3% 11.1%</td>
<td>77.8% 44.4% 50.0%</td>
<td>33.3% 33.3% 11.1%</td>
<td>11.1% 11.1% 20.0%</td>
<td>20.0% 20.0% 20.0%</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Fuel thermal conductivity model</td>
<td>70.0% 70.0%</td>
<td>88.9% 10.0% 20.0%</td>
<td>70.0% 70.0% 88.9%</td>
<td>10.0% 20.0% 0.0%</td>
<td>10.0% 30.0% 66.7%</td>
<td>66.7% 66.7% 66.7%</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Fuel thermal expansion model</td>
<td>0.0% 0.0%</td>
<td>0.0% 0.0% 0.0%</td>
<td>0.0% 0.0% 0.0%</td>
<td>100.0% 100.0% 100.0%</td>
<td>30.0% 30.0% 83.3%</td>
<td>83.3% 83.3% 83.3%</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Fuel enthalpy</td>
<td>100.0% 40.0%</td>
<td>50.0% 80.0% 80.0%</td>
<td>100.0% 100.0% 100.0%</td>
<td>100.0% 100.0% 100.0%</td>
<td>30.0% 30.0% 83.3%</td>
<td>83.3% 83.3% 83.3%</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Cladding thermal expansion</td>
<td>0.0% 0.0%</td>
<td>0.0% 11.1% 30.0%</td>
<td>0.0% 0.0% 0.0%</td>
<td>40.0% 40.0% 10.0%</td>
<td>0.0% 0.0% 0.0%</td>
<td>50.0% 50.0% 50.0%</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Cladding Yield stress</td>
<td>0.0% 0.0%</td>
<td>0.0% 33.3% 66.7%</td>
<td>0.0% 0.0% 0.0%</td>
<td>44.4% 44.4% 0.0%</td>
<td>40.0% 40.0% 0.0%</td>
<td>40.0% 40.0% 0.0%</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Note: A red coloured cell means that more than 50% of the participants have identified the corresponding input parameter as influential for the type of behaviour.

Source: [7].
An analysis of the results leads to the following conclusions:

- The injected energy is the most influential input parameter: it is considered influential by a large majority of participants for 16 out of the 17 outputs.
- Input data related to the end-of-life state (zirconia thickness, radial power profile, gap size, roughness) are also very influential. The initial pellet-cladding gap is the most influential in terms of cladding failure prediction.
- Fuel physical properties, thermal expansion and thermal conductivity models have a significant impact respectively on the rod mechanical and thermal behaviours.
- Regarding the fission gas release, besides injected energy and fuel enthalpy, participants have not identified any common influential input parameter.
- The cladding failure prediction is challenging because it is sensitive to many input data (9 influential input parameters out of 12 studied here).
- The cladding physical properties (thermal expansion and yield stress) have little impact on the outputs.
- It is worth noting that some outputs classified under the same behaviour type do not always have the same influential parameters (for instance, DHR and TFC).

### 4.4. Conclusions and recommendations of Phase III

Phase III confirmed the conclusions of Phase II concerning the strong dependence of the uncertainty results on the type of behaviour (in terms of uncertainty band width and coherence between participants). Moreover, it made it possible to extend the list of influential input parameters, with some due to the irradiation period. Based on the main outcomes of the analysis, the recommendations for further work are the following:

- Safety analysis studies can require uncertainty analysis on parameters associated to the state of the rod at the end of irradiation. However, for some codes, the transient pulse and the base irradiation are not considered apart. It could be interesting to develop strategies to allow the propagation of uncertainties on input parameters associated to irradiation behaviour.
- Further developments are required with a view to have validated FGR and CFP models. It first involves gathering more high quality data.
- Mechanical models are to be improved, including cladding stress behaviour and the cladding failure criteria.
- A first task to undertake before uncertainty and sensitivity analyses is the quantification of input uncertainties that was partly performed in this benchmark by expert judgement. The recently outlined SAPIUM guidance [17] could be exploited for a transparent and rigorous model for input uncertainty quantification in order to minimise the user effect.
Chapter 5. **Conclusions and recommendations**

Since the Reactivity-Initiated Accident workshop held in 2009 [1], the NEA Working Group on Fuel Safety organised a series of three phases of RIA Fuel Rod Codes Benchmark to assess the performance of the fuel rod codes for predicting the fuel behaviour under RIA and to identify the most influential input parameters:

- **Phase I (2010-2013)** was based on RIA tests with irradiated high burnup fuels (VA-1 and VA-3 tests, CIP0-1 test and the future CIP3-1 test) [2].
- **Phase II (2014-2016)** was based on simplified cases with a fresh fuel rod [4] including uncertainty and sensitivity analysis [6].
- **Phase III (2018-2019)** was based on the CIP0-1 test including uncertainty and sensitivity analysis [7].

Participation in the RIA benchmark was significant: 21 organisations representing 14 countries provided solutions for some or all the cases that were defined. In terms of computer codes used, the spectrum was also very large: 8 base irradiation codes, 10 transient analysis codes and 12 statistical analysis tools (codes and in-house procedures).

Nearly all the participants used a transient code relying on a simplified geometrical representation of the fuel modelling, usually referred to as 1.5-dimensional codes (i.e. the thermomechanics of the fuel rod is modelled in the radial direction but the axial direction is handled only by the communication of pellet-cladding gap pressure and the axial coolant conditions). Although some three-dimensional calculations may be done, it appeared that the detailed geometrical description was not necessary. Rather, it was considered more important to focus the efforts on physical modelling.

**The main conclusions of the three phases were the following:**

- The pellet-cladding mechanical interaction (PCMI) phenomena are well understood and most of the transient codes showed a relatively good agreement during this phase.
- With respect to the fuel thermal behaviour, the differences in the estimation of fuel enthalpies and temperatures by the different codes are rather limited, especially for the maximum values of these parameters.
- Concerning cladding temperatures and hoop strain evaluations, considerable scatter is obtained for the cases where water boiling occurs. This scatter is clearly related to the cladding-to-coolant heat transfer modelling. There is indeed no consensus on the post-departure from nucleate boiling (DNB) behaviour and in particular on the cladding mechanical behaviour at high temperature and on the heat transfer between cladding and coolant during the boiling crisis. Boiling in RIA conditions is known to be significantly different from that in steady-state conditions. Some codes assume that the steady-state correlations are applicable to RIA conditions while other codes use specific fast transient correlations (for critical heat flux, heat transfer in film boiling, rewetting conditions, etc.). Given the lack of sufficient experimental investigation on boiling under RIA conditions, no sound recommendation can be made to suggest the most suitable heat transfer correlations.
CONCLUSIONS AND RECOMMENDATIONS

- From cases devoted to boiling water reactor (BWR) conditions, it is clear that very few (if any) of the applied computer codes are able to handle the thermal-hydraulics conditions expected in a BWR RIA with large energy injection at cold, zero power conditions. This is not simply a question of uncertainties in the cladding-to-coolant heat transfer modelling; the excessive steam generation expected in the fuel assembly at atmospheric pressure can obviously not be handled by the simplified models used in the codes for the evaluation of thermal-hydraulics boundary conditions.

- Various influential input parameters are identified for fresh fuel. For instance, injected energy, fuel enthalpy, parameters related to the rod geometry (pellet and cladding roughness, cladding inner diameter), fuel thermal expansion model and full width at half maximum (FWHM) of the power pulse have been found to be influential regarding the maximum value of each output parameter of interest. Uncertainties cannot fully explain the scatter observed during these benchmark exercises.

- The RIA Fuel Rod Codes Benchmark Phase III, which was focused on uncertainty and sensitivity analyses on an irradiated case, confirmed the conclusions of Phase II concerning the strong dependence of the uncertainty results on the type of output. Input data related to rod state after base irradiation (initial pellet-cladding gap, zirconia thickness, radial power profile, roughness) are also very influential. The initial pellet-cladding gap is the most influential parameter in terms of the cladding failure prediction.

- Some codes have been developed to perform both base irradiation and transient calculations, which ensures continuity between the two phases, but makes it more difficult to perform the decoupling. Their users therefore had difficulties to match the pre-transient state defined in the specifications.

- The specifics of the applied codes and the user effect could play a more important role than the uncertainties.

The OECD RIA Fuel Rod Codes Benchmark offered an opportunity to investigate the temperature effect (i.e. whether it is realistic to use the RIA fuel codes to transpose results from experiments performed at test reactors to typical power reactor conditions) because it contained results of experiments performed on similar rods but at different temperatures. The conclusion was that the extrapolation to reactor conditions should be done with caution given the scatter that existed between the predictions of various codes. Making such a transposition with two different codes could lead to results that differ significantly.

Furthermore, it is recommended:

**Improvement and validation of the models implemented in RIA fuel rod codes**

- It is recommended to build up a comprehensive and robust database consisting of results from both separate effect tests and integral RIA simulation tests. This database could be defined in an international framework. In this way, both individual model validation and model integration into codes would be feasible. The database could be shared by the modellers, whenever possible, to ease the comparison of simulation results from various codes. For the purpose of model validation, it is important to include detailed information on the pre-irradiation conditions of the test rods, as well as results available from pre-test and post-test characterisation.

- Models related to the evolution of the pellet-cladding gap should be improved and validated for RIA conditions as this has been shown to have a significant effect on fuel rod response. To reach this objective, in-reactor measurements of cladding strain during RIA simulation tests should be done (or at least attempted).

- Fuel and cladding thermo-mechanical models (with the associated material properties) should be further improved and validated more extensively against a well-qualified RIA database. With regard to the improvements of mechanical models, efforts should be focused in particular on the failure probability models and failure criteria of claddings.
• Further developments are required for the transient axial gas flow and release, with attention to their validation, which first involves gathering more high-quality data.

• The cladding-to-coolant heat transfer in the case of water boiling during very fast transients is of particular interest, and capabilities related to modelling this phenomenon should be improved. Co-operation between existing experimental teams could be established to collect and share all existing data and to formulate specific proposals in order to reduce the lack of knowledge and achieve common understanding on the topics.

**Reduction of user and code effect for best estimate simulation**

• To reduce user effect and code effect, the code development teams should provide, where possible, recommendations regarding the code versions, models, and numerical parameters to be used (mesh size, time step, etc.). The availability of a RIA experimental database can help code users assess their modelling techniques.

**Treatment of uncertainties**

• When the uncertainty evaluation is based on input uncertainty propagation, a first task to be accomplished (before uncertainty and sensitivity analyses) is the quantification of input uncertainties that was partly performed in this benchmark by expert judgement. To minimise the user effect, the recently outlined SAPIUM guidance [17] could be exploited for a transparent and rigorous model for input uncertainty. However, this methodology should be applied cautiously because of the complexity in conducting it and interpreting the results properly.

• Sensitivity and uncertainty analyses should be considered an integral part of computer code applications to RIA. These analyses should be directed towards models and phenomena for which the most substantial uncertainties are known to exist.

• Safety analysis studies can require uncertainty analysis on parameters associated to the state of the rod at the end of irradiation. However, for some RIA fuel rod codes, the transient pulse and the base irradiation are not considered apart. It could be of interest to develop strategies to allow the propagation of uncertainties on input parameters associated to irradiation behaviour.
References


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Reactivity-Initiated Accident Fuel Rod Codes
Benchmark Phases I-III: Synthesis Report

One of the key areas in fuel safety is the analysis of fuel behaviour under reactivity-initiated accident conditions. Reactivity-initiated accident fuel rod codes have been developed for a significant period of time and they all have shown their ability to reproduce some experimental results with a certain degree of adequacy. However, they sometimes rely on different specific modelling assumptions whose influence on the final results of the calculations is difficult to evaluate. This report summarises three phases of benchmark conducted by the NEA between 2010 and 2019 with codes for calculating fuel behaviour in reactivity initiated accidents. Building on previous NEA reports, it provides recommendations for future research and code enhancements for safety analysis regarding reactivity accidents.