

The PARAMETER test series

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ABSTRACT

The PARAMETER programme investigates phenomena associated with reflooding of a degrading Pressurized Water Reactor (both Western and VVER type) like core under postulated severe accident conditions, in an early degradation phase when the geometry is still mainly intact. The objective of the presented out-of-pile test series is both the experimental and computational investigation of 19-fuel rods bundle behaviour under severe accident conditions including the stage of low rate flooding from bundle top or high-rate flooding from bottom and top. The studies are performed according to the Work Plan of ISTC Project#3194 "Fuel assembly tests under severe accident conditions". The PARAMETER programme is being realized jointly by the leading organizations of the Federal Atomic Energy Agency of the Russian Federation: FSUE EDO "GIDROPRESS", IBRAE RAS, FSUE SRI SIA "LUCH". The project is performed under financial support by ISTC and in close cooperation with leading European R&D organizations such as FZK, GRS (Germany), CEA, EDF and IRSN (France).

The paper will provide the information on scope of work as already completed or planned in framework of PARAMETER-SF test series. Some outcomes from PARAMETER-SF1 and PARAMETER-SF2 tests will be presented for discussion. In the PARAMETER-SF1 experiment the behaviour of VVER-1000 assembly overheated up to 2000°C under top flooding conditions was studied. In another PARAMETER-SF2 experiment the efficiency of the combined top and bottom flooding was studied for the VVER-1000 assembly overheated to 1500°C. An outlook will be given for the foreseen experiment PARAMETER-SF3 (top flooding).

The experimental part of PARAMETER project is supported by intensive analytical support. For this purpose widely used codes for NPP safety assessment were involved. Considerable contribution to the success of the PARAMETER project was provided by teams from Russia (SOCRAT, PARAM-TG, ICARE/CATHARE, and RELAP codes), France (MAAP4) and Germany (ATHLET-CD). So in the paper some results of pre- and post-test modelling and outcomes from code-to-code and code-to-data comparisons will be presented for both SF1 and SF2 tests as well.

1. Introduction

The main method of the analysis of severe accidents is the numerical simulation with the use of computer codes. Complexity and mutual relation of physical processes and phenomena during the accident including the stage of temperature escalation and the stage of the core reflooding [1, 2] lead to the need of the comprehensive verification of the models and codes. Besides, to take the justified solutions on accident management and bringing the reactor into safe state it is necessary to have rather a clear concept of the change in the core state in the course of accident and of possible methods of its cooling down. Then significant efforts are taken for studying of the initial stage of severe accident wherein a possibility of the overheated core cooling is investigated as a possible measure on managing the accident.

One of the specific features of VVER reactor plant is a possibility of the core reflooding from top and bottom in a case of recovery of the Emergency Core Cooling System (ECCS) active part. Under “bottom” flooding, the pressure chamber is filled first, and only after its filling the boiling starts. The core is cooled down with steam first and with water after wetting of fuel rods. Under “top” flooding, the water entering the reactor collection chamber is distributed through the core section and moves downwards cooling the core. Therefore, the core cooling under “top” flooding can occur somewhat earlier than under “bottom” flooding. Later combined flooding conditions take place.

Top flooding conditions come in PWR when:

- steam condensed in steam generator (SG) tubes returns to the core through the hot leg;
- water injected into the hot leg from the ECCS partially enters into the core.

In this connection the experimental study of fuel assembly (FA) behaviour under the severe accident conditions, including the stage of the reflooding from top, bottom, and simultaneous reflooding is of great scientific interest.

The behaviour of the overheated assembly of PWR (twelve tests up to date) [3] and VVER (test QUENCH-12) [4] under bottom reflooding was studied in the QUENCH programme and the corresponding data are widely used for verification of the computer codes. However all experiments, performed up to now in framework of the QUENCH program, modelled only bottom flooding.

The PARAMETER-SF experiments series performed recently on the PARAMETER test facility in FSUE SRI SIA “LUCH” [5] became the essential contribution to the verification base of the computer codes. In the PARAMETER-SF1 experiment [6] the behaviour of VVER-1000 assembly overheated up to 2000°C was studied for the first time under top flooding conditions. In the other experiment PARAMETER-SF2 [7], the efficiency of simultaneous top and bottom flooding of VVER-1000 assembly overheated to 1500°C was studied. The rate of top and bottom flooding corresponded to the prototype (by water mass per fuel rod) under LOCA conditions at VVER NPP.

The studies are performed according to the Work Plan of ISTC Project#3194 “Fuel assembly tests under severe accident conditions”. The project was realized jointly by the leading organizations of the Federal Atomic Energy Agency of the Russian Federation: FSUE EDO “GIDROPRESS”; IBRAE RAS, FSUE SRI SIA “LUCH”. The project is performed under financial support by ISTC and in close cooperation with leading European R&D organizations such as FZK, GRS (Germany), CEA, EDF and IRSN (France).

The paper provides the information on scope of work as have been already completed or foreseen in framework of PARAMETER-SF test series. Some outcomes from the PARAMETER-SF1 and PARAMETER-SF2 tests are presented for discussion. An outlook will be given on the future experiment PARAMETER-SF3 (top flooding).

The experimental part of PARAMETER project is supported by intensive analytical work. For this purpose, widely used codes for NPP safety assessment were involved. Considerable contribution for success of the PARAMETER project was done by teams from Russia (SOCRAT, PARAM-TG, ICARE/CATHARE, and RELAP codes), France (MAAP4) and Germany (ATHLET-CD). So in the paper some results of pre- and post-test modelling and outcomes from code-to-code and code-to-data comparisons are presented for both PARAMETER-SF1 and -SF2 tests as well [8, 9].

2. Pre-test calculations of the PARAMETER-SF1 experiment

The experiment PARAMETER-SF1 included the following main stages:

- heat up of the bundle to the temperature of 1470 K (in the hottest zone);
- holding of the bundle at the temperature of 1470 K in flowing argon/steam environment (pre-oxidation stage);
- heat up of the bundle to the temperature of 2100 K (transient stage);
- top flooding.

The test bundle is made up of the 19 fuel rod simulators. The claddings of the fuel rods are identical to those used in VVER with respect to material and dimensions. The fuel rods are filled with UO_2 pellets. 18 fuel rods are heated electrically over a length of 1275 mm. Tantalum heaters are installed in the centre of the rods to heat up the rods. The test bundle is surrounded by a hexahedral shroud, a thick ZrO_2 fibre insulation to decrease radial heat loss, a thermoinsulation shroud, and a body with a water cooling jacket. The bundle cross section is given in [Fig. 1](#).

For justification of the PARAMETER-SF1 experiment scenario pre-test numerical analyses including an analysis of the effect of the physical-chemical and thermohydraulic processes on temperature behaviour of the assembly have been performed using various computer codes. Most of them were developed for reactor applications and the pre-test calculations represent a blind phase of its verification in fact.

Electric power mode supplied presented in [Fig. 2](#) applied as a boundary condition. Some users used the electric power mode specified in the Specification booklet (SOCRAT (RATEG), ATHLET-CD), the others tuned it to fit needed temperature evolution.

All used codes include the models to calculate power loss due to the electric resistance of the electrodes and leads. The value of the external resistance has an influence on the calculation of the remaining power to heat the bundle (Joule heating power), but it was unknown. The users assumed the value based on their own experience. Proposed heat loss due to external resistance varies from 0.5 to 3 kW.

Calculated Joule heating power is presented in [Fig. 3](#). Most of the codes predict that Joule heating power of 5-6 kW provides the bundle holding at 1470 K, its increasing up to 11-12 kW causes the bundle heat up to 2100 K, i.e. proposed electric power mode results in needed temperature evolution.

Surrounding structures (thermoinulation, thermoinulation shroud, and cooling jacket) models are simplified. It leads to a large scattering (~3 kW) in the calculated radial heat loss through the shroud ([Fig. 4](#)). So heat losses through shroud and due to external resistance represent the significant factor of differences in the predictions.

Power generated by the oxidation reaction becomes significant (from 10 to 50 kW) just before the top flooding onset to be equal to Joule heating power (11-12 kW) or even higher. Heat exchange between the hot claddings and the steam flowing upwards result in nonuniform temperature distribution over the heated zone. The hottest zone location should be expected at the top of the heated zone. Due to heat loss through unheated upper part of the test section the

hottest zone would move down. Most of the applied codes predict that maximum temperature should be expected at 1200-1300 mm elevation. Claddings temperature at the elevations of 1250 mm can be seen in Fig. 5.

The total hydrogen release is shown in Fig. 6. One can note a large scattering in calculated results. The reactor codes predict from 22 to 36 g of H₂ before quenching starts. At the quenching stage RELAP shows significant hydrogen generation, SOCRAT predicts moderate hydrogen release (~30% of total amount), another codes give a small hydrogen mass (from 5 to 15%). The maximum of the ZrO₂ layer thickness ranges from ~500 to 1000 μm. It can be found at the elevations near the hottest region.

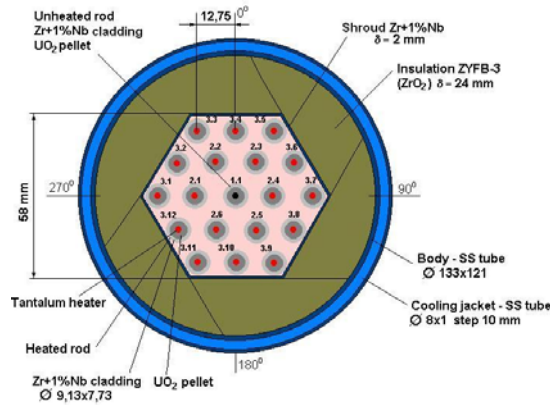


Fig. 1. Cross section of PARAMETER-SF1 bundle.

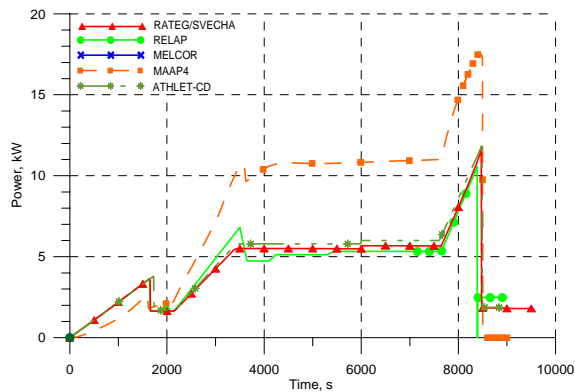


Fig. 2. Electric power supply.

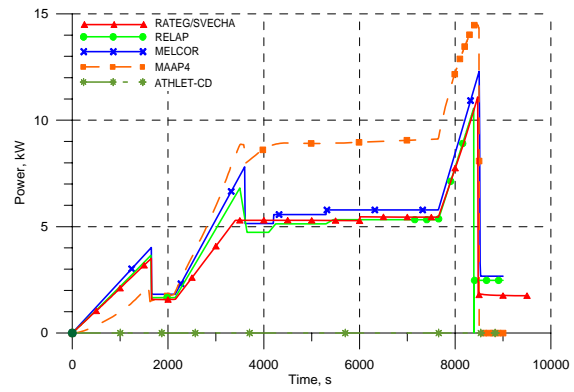


Fig. 3. Joule heating power.

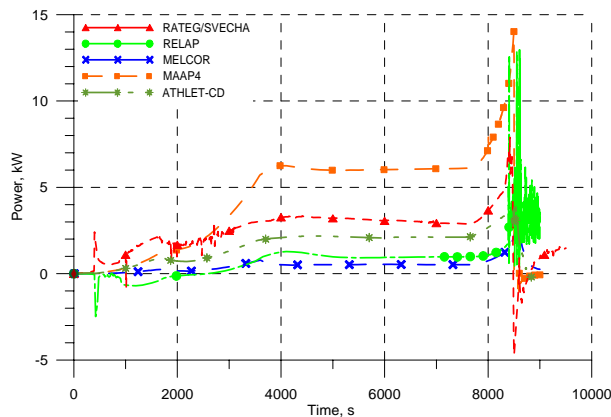


Fig. 4. Radial heat loss (through shroud).

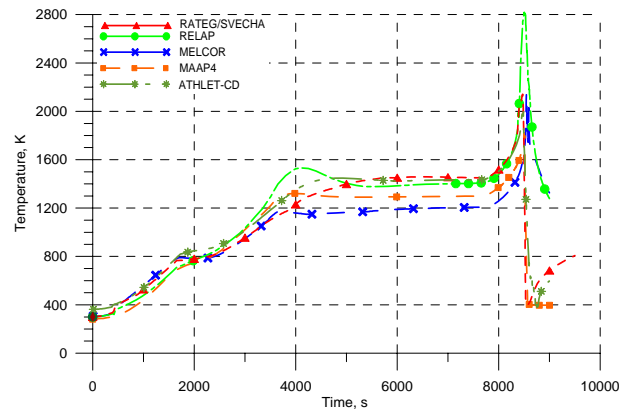


Fig. 5. Cladding temperature of fuel rods in the 3rd ring at the elevation of 1250 mm.

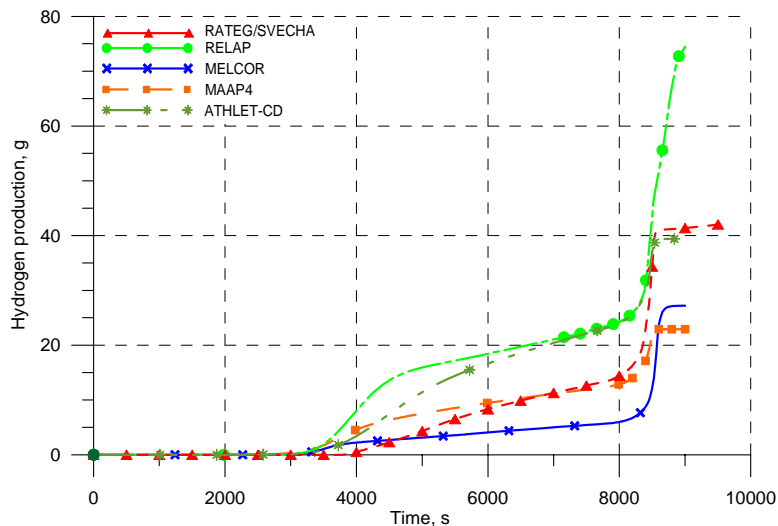


Fig. 6. Hydrogen release.

3. Main results of the PARAMETER-SF1 experiment. Post-test numerical analysis of the PARAMETER-SF1 experiment

After the test, it was found a burnout on the thermoinsulation shroud surface (Fig. 7) at the elevation of ~ 800 mm. The shroud is strongly oxidized, partially damaged at the elevations of 700-1000 mm (Fig. 8). Maximum temperature exceeded 2270 K. Total amount of the generated hydrogen was ~ 91 g and $\sim 59\%$ of this amount was produced during the flooding stage. The maximum rate of hydrogen generation was $\sim 0,19$ g/s. Fig. 9 provides the pictures of the cross-sections at elevations 652 mm (local melt), 793 mm (total melting) and 1263 mm (the bundle oxidized but remained undamaged).



Fig. 7. View of the thermoinsulation shroud at the elevations of 700 – 900 mm.



Fig. 8. Post-test appearance of the shroud (elevations of 700 – 1000 mm).

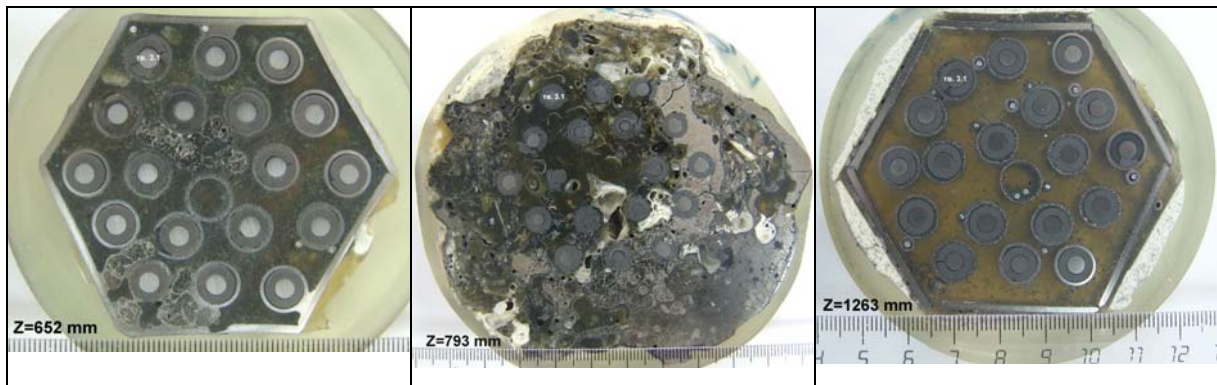


Fig. 9. PARAMETER-SF1 cross-sections.

In the post-test calculations of the PARAMETER-SF1 test one should take into account processes of steam condensation and bypassing which occur during the PARAMETER-SF1 experiment due to design peculiarities of the test section. Steam condensation can be treated by models of condensation heat transfer in presence of non-condensable gas. Steam bypassing via thermoinsulation can be modelled by additional flow path in nodalization scheme but the time of bypass formation, sizes of unsealing zones are unknown. It only allows approximated description of the bypassing phenomenon. In Fig. 10, the calculated data of the flow rate of steam and argon through the assembly are presented. For comparison, the indications of the test facility flow metering devices reflecting the flow rate of steam and argon at the assembly inlet are given.

All the computer codes give satisfactory description of fuel rod cladding temperature behaviour for first 10000 seconds of the test. For example, in Fig. 11 the calculated and experimental data on the cladding temperatures of the fuel rods in the 2nd ring at the elevation of 1250 mm are presented. The calculated data are in a good agreement with each other, as well as with thermocouples readings. Beginning from 10000 second the results of the calculation of cladding temperature behaviour at the elevations of 1100-1300 mm become different. It should be noted that this zone is the hottest part of the assembly, the cladding temperature here is sensitive to steam flow rate due to considerable contribution of heat exchange between cladding and steam in heat balance. At the pre-oxidation stage differences in calculated temperature reach 600 degrees (at the elevation of 1300 mm). Cause of the differences is supposed to be simplified modelling of condensation and bypassing phenomena. The difference becomes still greater at the end of the transient phase just before flooding. Some codes (RATEG/SVECHA, ICARE) predict that melting of metal zirconium occurs in the assembly – the calculated temperatures exceed 2000°C.

Fig. 12 gives the hydrogen release. We can see a qualitative agreement of the calculated and experimental data. In spite of noted scattering in calculated temperature at the pre-oxidation phase, the amount of hydrogen generated by the end of the pre-oxidation phase was predicted by the codes with an accuracy of 30-40%. The main difference appears at the end of the transient stage and during the flooding stage. At the transient and the flooding stages, RELAP code predicts no extra hydrogen generation although during these phases melt formation and its strong oxidation occurred. Other codes simulating these processes give significant hydrogen release. However, all the codes underestimate total hydrogen production.

A special attention was focused on assessing the reliability and self-consistency of the PARAMETER-SF1 experimental data. Using RATEG/SVECHA code, one can manage to reproduce the set of experimental data on temperature (Fig. 13), oxide layer thickness (Fig. 14) and fuel pellet dissolution (Fig. 15).

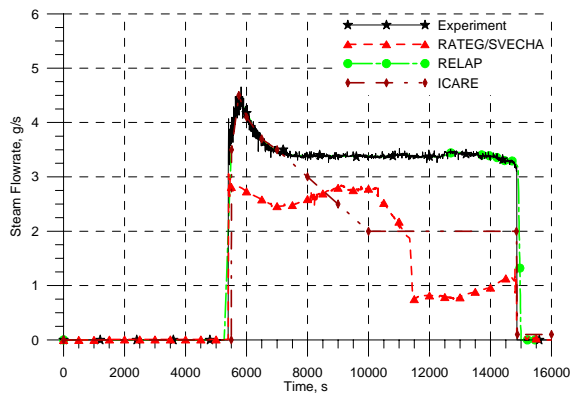


Fig. 10. Steam flow rate (at the assembly inlet) and its calculated approximations (through the assembly).

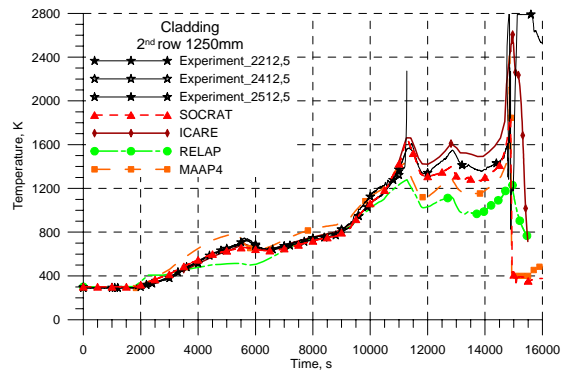


Fig. 11. Claddings temperature of the fuel rods in the 2nd ring at the elevation of 1250 mm.

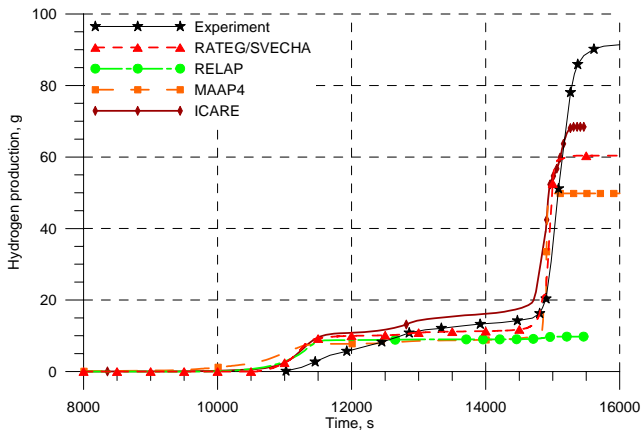


Fig. 12. Hydrogen production.

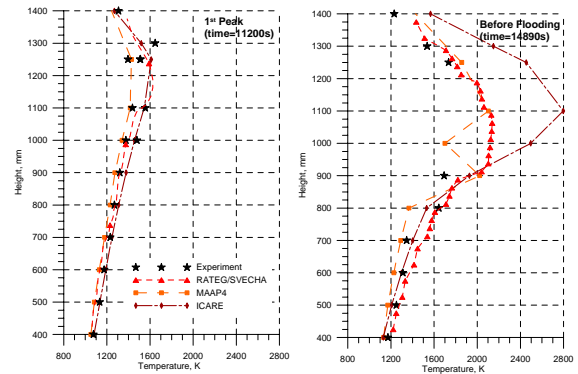


Fig. 13. Temperature axial profile for claddings in the 2nd ring rods.

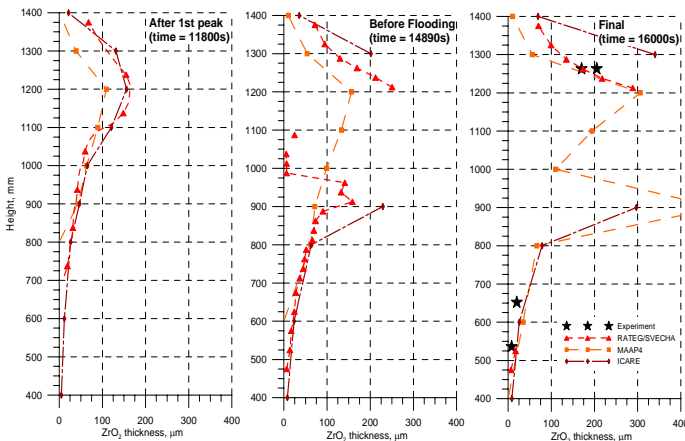


Fig. 14. Claddings oxide scale thickness axial profile for the 2nd ring rods.

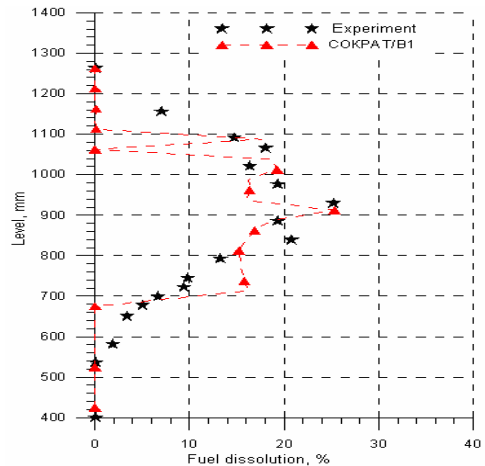


Fig. 15. Dissolution of UO₂ pellets by the end of the experiment.

4. Pre-test calculations of the PARAMETER-SF2 experiment

The scenario of the PARAMETER-SF2 experiment was developed to assess the adequacy of modelling the flooding and oxidization processes by the SFD codes and to obtain the data for improving the thermo-hydraulic models. The main goal to be solved by pre-test calculations was to determine supplied electric power to fit needed temperature scenario and time delay between onsets of top and bottom flooding (Fig. 16). Input data decks were derived from the

PARAMETER-SF1 post-test calculation and was adjusted to the special initial and boundary conditions of the PARAMETER-SF2 test and also to the changed geometry.

Uncertainties in the external resistance (leads and electrodes) lead to acceptable (about 1 kW) uncertainties in Joule heating power (Fig. 17). Another objective cause which will result in differences between calculated results is connected with the uncertainties in heat loss through shroud. The users used their own experience of PARAMETER-SF1 modelling but in PARAMETER-SF2 the geometry of the shroud and the cooling jacket was changed. In the PARAMETER-SF2 cylindrical shroud is used instead of hexahedral in PARAMETER-SF1, coil cooling jacket was replaced by additional cylinder to allow cooling water to flow upwards between the body and the external cylinder.

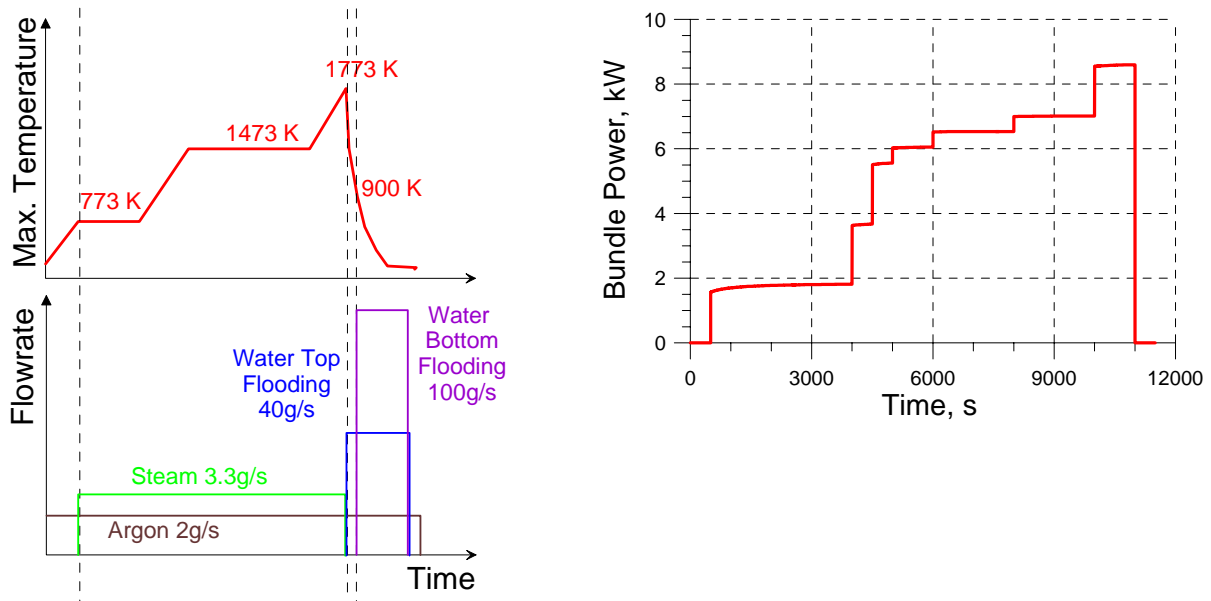


Fig. 16. Proposed boundary conditions.

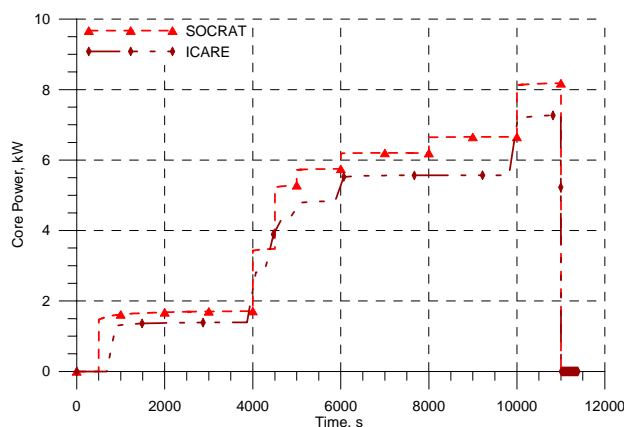


Fig. 17. Joule heating power.

Codes show that the maximum temperatures in the assembly should be expected at 1200-1300 mm elevation (Fig. 18), maximum thickness of the oxide layer on the fuel rod claddings is about 250 μm (Fig. 19).

To choose the scheme of water supply (sequence and time delay) multi-variant calculations were performed. The numerical analysis reveals that if the “bottom” water reaches -300 mm elevation, where the temperature of the body and the fuel rod claddings are high enough (~ 750 K), intensive water boiling starts. In Fig. 20 one can see that at time ~ 11070 s the

claddings at the 1250 mm elevation, already cooled with the top flooding water to the temperature of ~ 400 K, begin to heat up again. Evidently, this phenomenon is mainly due to the entrainment of the top flooding water by the steam generated in the bottom part of the bundle.

It was agreed to start the bottom flooding after the top flooding water flows downwards to the 1250 mm elevation (to the upper spacer grid) and cools the fuel rod claddings to the temperature of 900 K or below (close to the threshold of wetting). According to SOCRAT predictions, the time delay between the top and bottom start is ~ 50 s.

During the flooding phase, one should expect a few amount of hydrogen production of 3 g, total hydrogen release is estimated to be 23 g (Fig. 21).

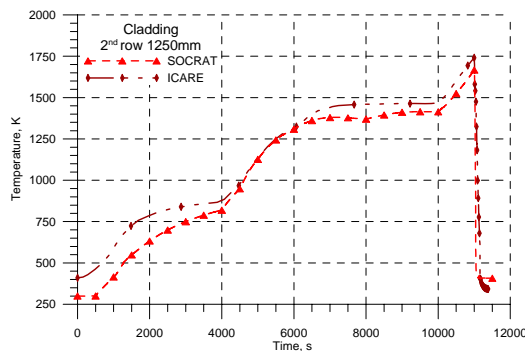


Fig. 18. The 2nd row rods claddings temperature at the elevation of 1250 mm.

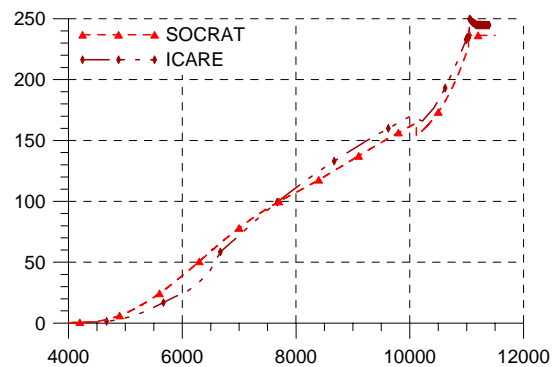


Fig. 19. Oxide scale thickness at the elevation of 1200 mm.

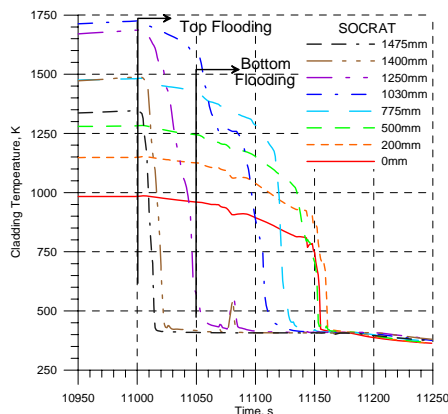


Fig. 20. Cladding temperature at the flooding stage calculated by SOCRAT.

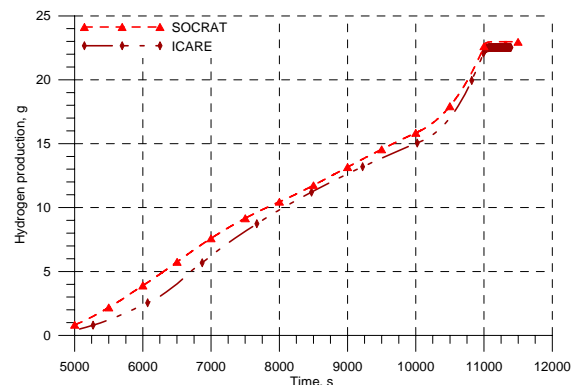


Fig. 21. Hydrogen release.

5. Main results of the PARAMETER-SF2 experiment. Post-test analysis of the PARAMETER-SF2 experiment

During the experiment, the bundle was preoxidized for ~ 3400 s at the peak temperature of fuel rod claddings of 1470 K in the flowing steam-argon environment, steam flow rate through the assembly was ~ 3.64 g/s. At the moment of switching the power before quenching onset the cladding temperature in the hottest zone at the elevation of 1250 mm was ~ 1750 K.

Water supply into top part of the test section resulted in cooling the fuel rod claddings at the elevation of 1250 mm and upper. The effect of top flooding on the cladding temperature at 1100 mm and below was not recorded by thermocouples. Water supply into the bottom part resulted in successive cooling of claddings at the levels of 0 – 1100 mm.

Bottom flooding resulted in intensive water evaporation and entrainment of the top flooding water that, in its turn, resulted in the claddings reheating.

After the tests the dismantling, preservation and cutting of the model assembly were performed. Analysis of the shroud state showed that it kept its integrity and leak tightness. Visual examination of the state of the assembly structural materials at the elevations of 702, 1103 and 1250 mm are presented in [Fig. 22](#). The fuel rod claddings were oxidized considerably and partially fragmented, there is no melt in the space between fuel rods.

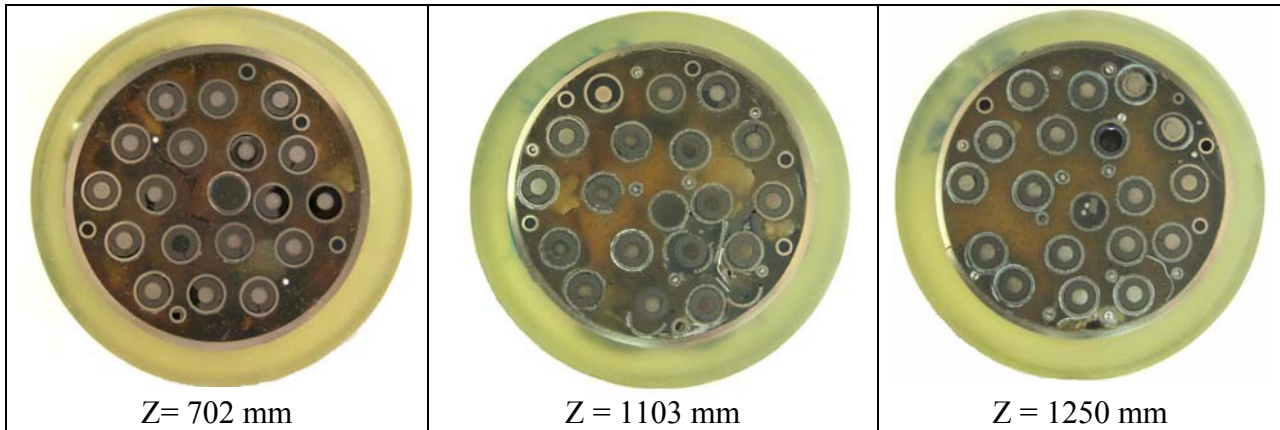


Fig. 22. PARAMETER-SF2 cross-sections.

Comparison of the PARAMETER-SF2 pre-test calculated results with the experimental data reveals that experimental electric power at the pre-oxidation and the transient phase higher than calculated one. In the course of the post-test calculations heat loss in the external resistances (leads and electrodes) and radial heat loss through the shroud unknown exactly were improved.

Post-test calculations were performed on the basis of the PARAMETER-SF2 experimental data on power history, mass flow rates, inlet temperature ([Figs. 23, 24](#)). In RELAP code Joule heating power instead electric power was applied as a boundary condition.

Heat loss in the external resistance is assessed to be 1.5 kW at the pre-oxidation and the transient phases. Scattering in calculated Joule heating power doesn't exceed 0.5 kW. Most part of Joule heating power is transferred to the coolant at these phases.

From ~10000 s oxidation reaction energy starts to release. Its contribution in the heat balance is small at the pre-oxidation phase (up to 1 kW), by the end of the transient phase that value might be 2.3 kW, i.e. no more than 30% of the Joule heating power.

All codes well reproduce temperature evolution of the bundle heated part besides the "hump" in experimental temperature at ~12000 s ([Fig. 25](#)). Calculated maximal temperatures before flooding onset correspond to experimental one (~1770 K). RELAP code demonstrates too fast runaway at the transient phase, and early electric power switching off is needed to fit experimental maximum temperature.

Significant differences between calculated and experimental hydrogen flowrate for 12000-13000 s time period ([Fig. 27](#)) for SOCRAT, ICARE, ATHLET would follow from unreproduced "hump" in temperature behaviour. RELAP code overestimates both hydrogen flowrate and total hydrogen besides time period with "hump" due to higher calculated temperature in the upper unheated part of the test section but a good fit with experimental data was achieved by turning off the electric power earlier.

At the flooding stage all codes predict practically no hydrogen release. Some hydrogen increasing in SOV-3 system indications at that phase might be result of time delay connected with hydrogen transport from source to hydrogen measurement device.

The best coincidence between experimental hydrogen mass and calculated is achieved in calculations by ICARE code for the entire experiment (Fig. 26). 28 g of hydrogen mass was measured at the end of the experiment; ICARE demonstrates 25.5 g. ATHLET and SOCRAT codes show lower hydrogen release. One of the possible ways to improve the calculated results is based on the clarification of “hump” causes.

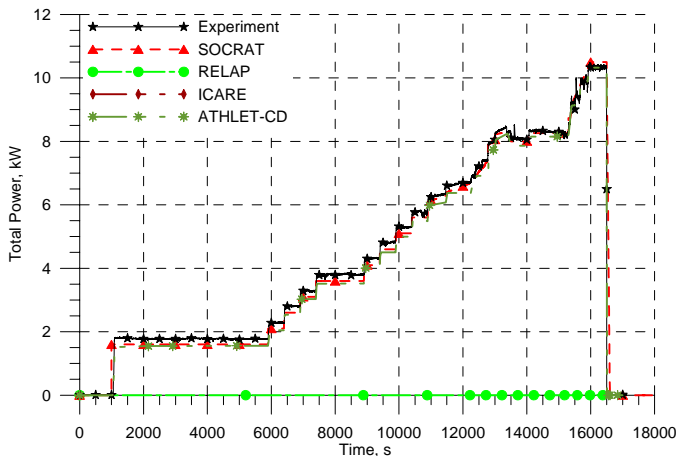


Fig. 23. Power supply.

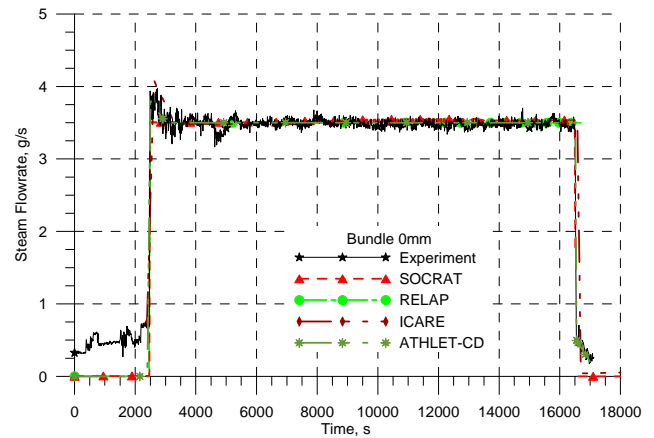


Fig. 24. Steam flow rate.

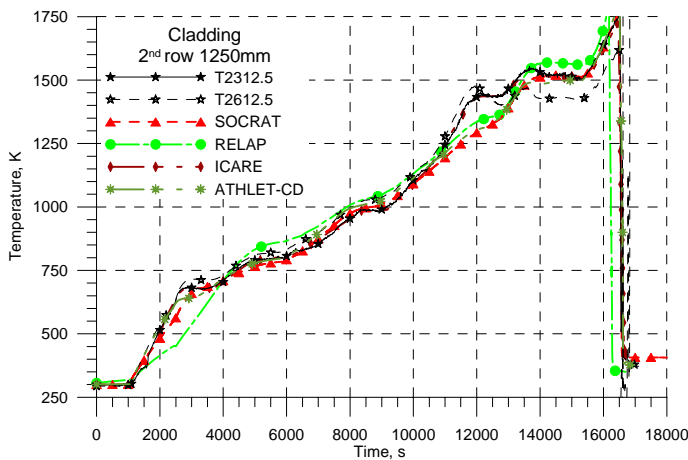


Fig. 25. Cladding temperature.

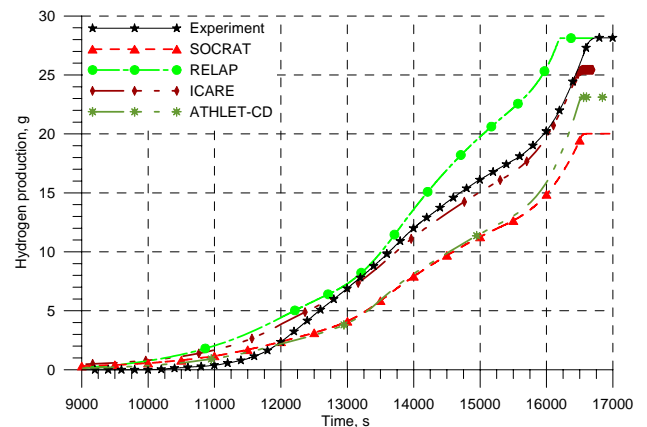


Fig. 26. Hydrogen release.

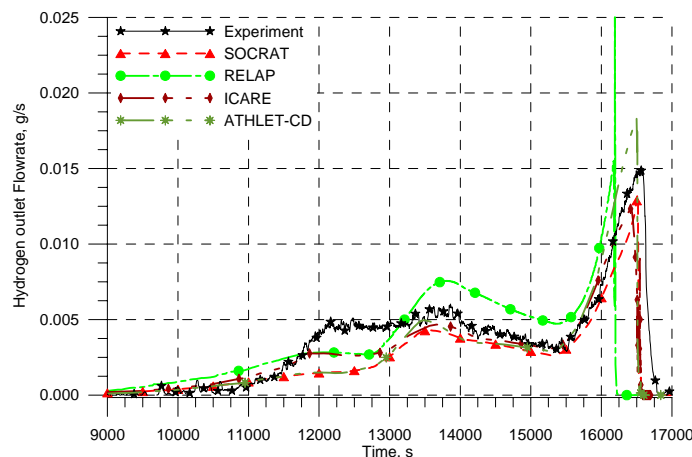


Fig. 27. Hydrogen flow rate.

On the basis of the PARAMETER-SF2 experimental results some oxidation models were checked. So post-test analysis using ATHLET code shows that oxidation with Sokolov's kinetic data leads to underestimation of hydrogen mass. Hydrogen generation is simulated in a good agreement with test data (besides period at ~12000s) with Leistikow/Prater/Courtright' kinetic data. But calculated hydrogen mass was slightly less than the experimental one.

Special efforts were aimed to thermohydraulic models assessment. It was found using ATHLET code that after the start of top-flooding the injected liquid dropped too fast down to the bottom of the bundle with the effect that the pre-cooling of the rods up to a bundle elevation of about 700 mm was too strong. The cooling-fronts derived from measured temperatures and the calculated quench-fronts are in good agreement within the heated region; so this calculation simulates very well the average behaviour of the cool down of the bundle (Fig. 28). Time point of total cooling (~16620 s) and elevation of cooling fronts intersection (1250 mm) were well predicted. Water discharge during quench (4.3 kg) and liquid evaporation (2.3 kg) agree well with the collected liquid mass in tank 5 (6.5 kg) Measured pressure increase after start of bottom flooding is not calculated in spite of evaporation rates comparable to the test (20 – 14 g/s), since the condenser was not simulated and a constant pressure has been input as boundary condition.

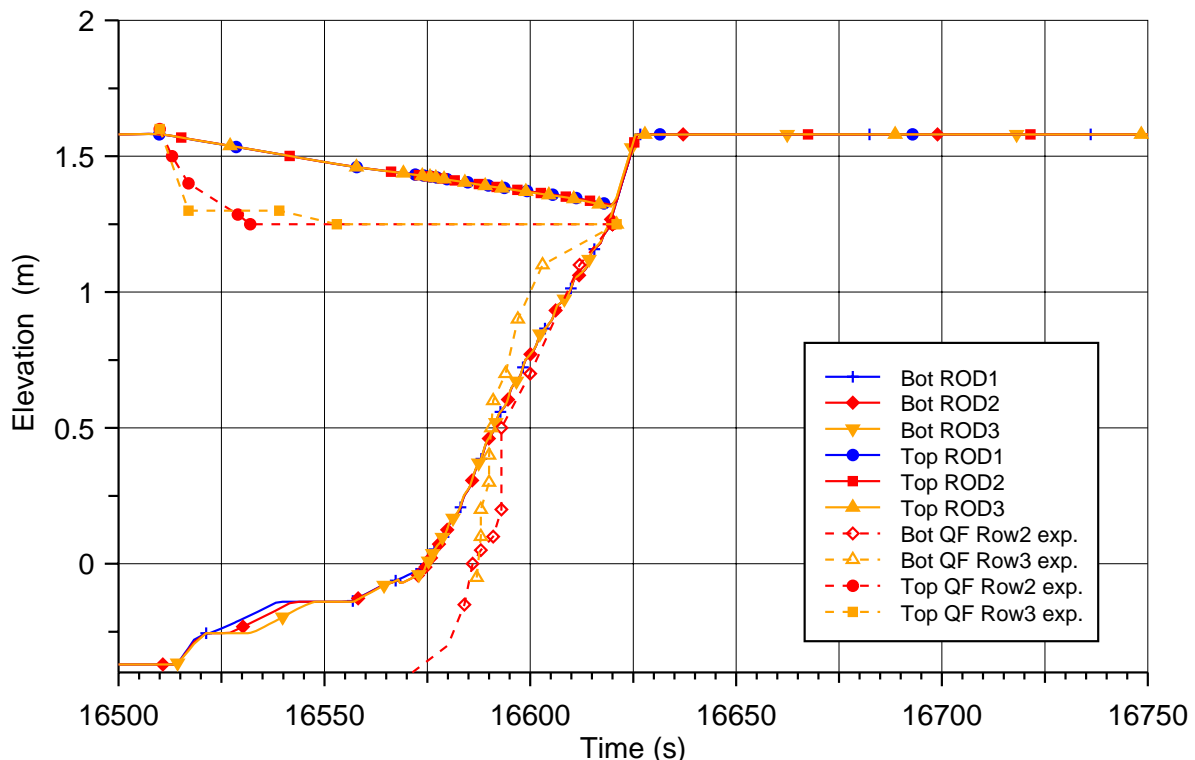


Fig. 28. Cooling-fronts derived from measured temperatures and the calculated quench-fronts.

6. Conclusion

In the course of performing the pre-test and post-test calculations of the PARAMETER-SF1 experiment by various reactor codes, the qualitative and quantitative agreements between the calculated and experimental data on temperature, oxide layers distribution, dissolution of uranium dioxide were obtained. A number of calculations were performed for sensitivities evaluation (insulation heat conductivity, bypassing, cladding temperatures at the elevation where no reliable indications of thermocouples are available, etc).

In spite of the uncertainty in PARAMETER-SF1 experimental data (decrease of coolant flow rate through the assembly under the conditions of damage of fuel rods and the surrounding structures, damage of the shroud and thermal insulation and failures of thermocouples at high temperatures), the set of experimental data was reproduced numerically. The agreement of calculated and experimental data on many processes and phenomena confirm the reliability of the SF1 experimental data and allow applying them for verification of the computer codes.

All codes underestimate hydrogen release in the PARAMETER-SF1 experiment (measured amount 91 g; calculated values less than 70 g). Main differences are revealed at quenching phase. It indicates that Counter-Current Flow Limitation (CCFL) models and melt oxidation models need improving.

Due to the melting pool formation and the shroud damaging in the PARAMETER-SF1 experiment CCFL process at the flooding stage is not singular. The foreseen PARAMETER-SF3 experiment with assumed moderate target temperature shall provide adequate experimental data to CCFL model checking.

Pre-test calculations of the PARAMETER-SF2 experiment predicted lower power supplied than in the experiment. Main cause could be referred to uncertainties in heat loss through the shroud. It couldn't be derived from modelling experience of PARAMETER-SF1 with high accuracy due to different geometry of the test sections. The phenomenon of reheating of the cladding at upper elevation after bottom flooding start, predicted by SOCRAT code, was observed in the test at the flooding stage. The time delay between onsets of top and bottom flooding as predicted by SOCRAT code is in a good agreement with experimental one. Hydrogen amount calculated for PARAMETER-SF2 by reactor codes (23 g) agrees with each other and underestimates experimental data (28 g).

In general a good fit between the post-test runs of the PARAMETER-SF2 experiment and experimental data was obtained. The obtained reliable experimental data allow users and codes developers to check and improve oxidation and CCFL models.

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