21st International Conference Nuclear Energy for New Europe
LJUBLJANA 2012
September 5 – 7

Post-Test Calculation of Air Ingress Experiment QUENCH-16 Using Thermal Hydraulic and Severe Accident Code SOCRAT/V3

Alexander Vasiliev
Nuclear Safety Institute (IBRAE)
B.Tulskaya 52,
115191 Moscow, Russia
vasil@ibrae.ac.ru

Juri Stuckert
Karlsruhe Institute of Technology (KIT)
Hermann-von-Helmholtz-Platz 1
76344 Eggenstein-Leopoldshafen, Germany
juri.stuckert@kit.edu

ABSTRACT

The thermal hydraulic and SFD (Severe Fuel Damage) best estimate computer modelling code SOCRAT/V3 has been used for the calculation of QUENCH-16 experiment which was performed in the frame of the EC supported LACOMECO programme.

The QUENCH-16 test conditions simulated a representative scenario of LOCA (Loss of Coolant Accident) nuclear power plant accident sequence in which the overheated up to 1800K core would be reflooded from the bottom by ECCS (Emergency Core Cooling System). The QUENCH-16 experiment included the following phases:

- First heat-up phase;
- Pre-oxidation phase;
- Slow cooldown phase (preparatory to air ingress);
- Air ingress phase;
- Bottom water flooding phase.

The test QUENCH-16 was successfully conducted at the KIT, Karlsruhe, Germany, on July 27, 2011. The primary objective of this test was to investigate the oxidation of Zircaloy in the air following a limited pre-oxidation in the steam and to achieve a long period of oxygen starvation to promote the interaction with the nitrogen.

QUENCH facility is designed for studies of the PWR fuel assemblies behaviour under conditions simulating design basis, beyond design basis and severe accidents.

SOCRAT/V3 computer modelling code was used for estimation of basic thermal hydraulic, oxidation and air ingress parameters in QUENCH-16.

The calculated results are in a reasonable agreement with experimental data which justifies the adequacy of modeling capabilities of SOCRAT code for application to such a complicated test as QUENCH-16.

1 INTRODUCTION

The lessons learned from severe nuclear accidents at Three Mile Island (Broughton *et al*., 1989), US, 1979, Chernobyl, USSR, 1986, and Fukushima, Japan, 2011, showed the very high influence of severe accident processes on beyond design basis accident dynamics. The deep understanding of hydraulic, mechanical and chemical processes taking place under NPP accident conditions is necessary, in particular, during air ingress conditions. It is clear now that a realistic description of accident processes is necessary for adequate accident evolution prediction instead of a conservative one.

Regarding this, the experimental and computational investigation of LOCA representative scenario with air ingress and water flooding as accident control measure will help in more thorough understanding of processes and phenomena relevant of severe accident sequences and improving of models implemented to computer modelling reactor accident codes.

At present, the experiments are underway in KIT, Karlsruhe, Germany. Those tests are under the QUENCH experimental program [1-11] aimed at studying mechanical and physical and chemical behaviour of overheated fuel rod cladding with quenching from bottom.

The QUENCH experiments contributed to the database on PWR (pressurized water reactor) severe accident phenomena. Thanks to QUENCH tests, a good understanding of *beyond design basis* accident processes and phenomena has been achieved.

The experiment QUENCH-16 was successfully conducted at the KIT on 27 July, 2011, with the aim to investigate the Zircaloy oxidation in air after a limited pre-oxidation in vapour and to achieve a long period of oxygen starvation to promote the interaction with nitrogen.

The investigation included the study of:

- thermo-hydraulic phenomena including flooding;
- physico-chemical phenomena (hydrogen generation and secondary hydrogen uptake);
- air ingress phenomena including oxidation in air and total oxygen starvation;
- the study of the behaviour of structural components of 21-rod model FA of PWR (pellets and claddings, shroud, spacing grids);
- the study of the oxidizing degree of the structural components of 21-rod model FA of PWR.

In a first transient, the bundle was heated up by power increase to about 1260 K at approximately 3000 s. Then, the power was controlled from 10 kW to 11.5 kW with the aim to maintain constant temperature. At 6300 s the power was reduced to 4 kW which resulted in cooling of FA to about 930 K. This phase lasted 1000 s until 7300 s and it was a preparation to air ingress phase. In the next phase namely the air ingress phase, the steam mass flow was replaced by the air with mass flow rate of 0.2 g/s. At the time 11340 s from the beginning of the test, the bottom quench water injection was initiated, the water flow rate was ~ 50 g/s.

The best estimate computer modelling code SOCRAT/V3 has been used for the calculation of QUENCH-LOCA-0 experiment. SOCRAT/V3 was verified on many severe accident experiments, in particular, on test series PARAMETER [12-14]. SOCRAT code consists of two major modules: RATEG – thermal hydraulics calculation, SVECHA – severe fuel damage phenomena description.

The two-phase water-steam thermal hydraulics behaviour under flooding conditions is a very interesting issue. Another important thermal process in QUENCH-16 test is the radiative heat transfer in the *square* rod bundle relevant of PWR FA. This is why advanced model of radiative heat exchange was implemented to SOCRAT code [15] to adequately estimate the heat transfer in the fuel assembly.

The important phase of QUENCH-16 was the air ingress phase during which the air was supplied to the working section of experimental installation. It is known that zirconium oxidation in the air proceeds in a different way in comparison to oxidation in the steam.

The QUENCH-16 calculated results obtained using SOCRAT/V3 has been compared to experimental data. The calculated and experimental data are in a reasonable agreement, which is indicative of the adequacy of modelling the complicated thermo-hydraulic and chemical behaviour in the QUENCH-16 experiment.

2 QUENCH FACILITY

The QUENCH facility at KIT is designed for studies of the Light Water Reactor (LWR) fuel assemblies behaviour under conditions simulating design basis and beyond design basis accidents at the nuclear power plants (NPP).

The QUENCH-16 test bundle (Figure 1) was made up of 21 fuel rod simulators with a length of approximately 2.48 m (heated rod simulators), which were hold together by means of five spacer grids. The rods were placed in the *square* set (Figure 2). 21 fuel rod simulators were heated over a length of 1024 mm. Heating was carried out electrically using 6-mm-diameter tungsten heaters. For the heated rods, tungsten heating elements were installed in the centre of the rods and were surrounded by annular ZrO_2 pellets. The tungsten heaters were connected to electrodes made of molybdenum and copper at each end of the heater.

The test bundle was surrounded by a Zr 702 shroud, followed by a 37 mm thick ZrO_2 fibre thermal insulation axially extending from the bottom to the upper end of the heated zone. Special corner rods, inserted between bundle and shroud, additionally reduced the coolant channel area to a representative value.

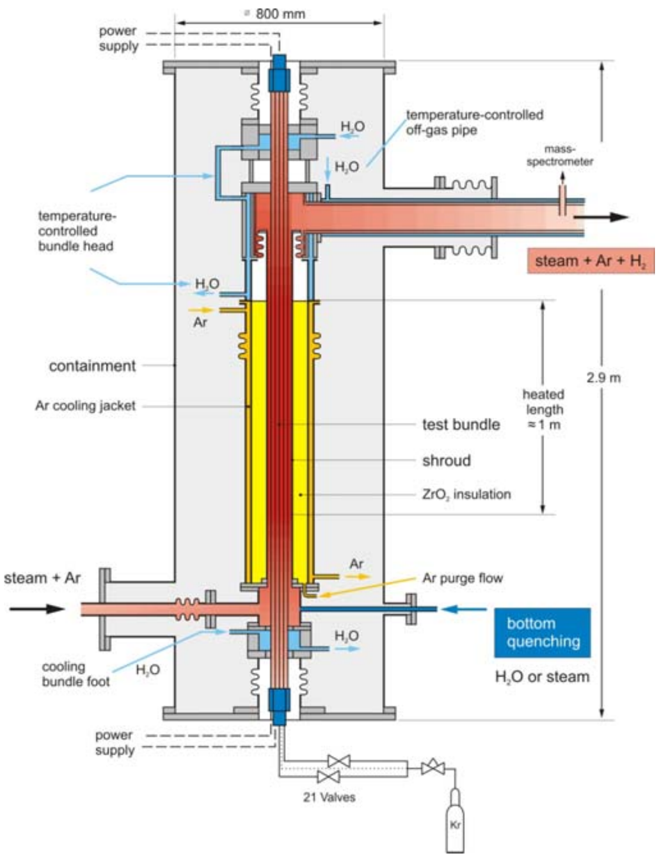

Figure: 1: Schematic representation of QUENCH test section facility

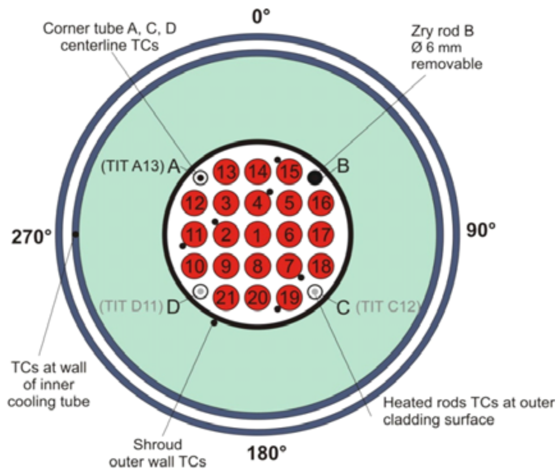

Figure 2: Cross-section of QLOCA-0 test bundle (21 heated, 4 corner rods). Consecutive numbers of rods are indicated

The rod cladding was identical to that used in LWRs: Zircaloy-4, 10.75 mm outside diameter, 0.725 mm wall thickness. The test bundle was instrumented with 1) thermocouples attached to the cladding and the shroud at 17 different elevations with an axial distance between the thermocouples of 100 mm; 2) with 21 pressure transducers connected to the internal plenum of each fuel rod simulator.

3 QUENCH FACILITY MODELING

The nodalization scheme of the QUENCH test facility for the SOCRAT/V3 computer modelling code is presented in Figure 3. The radiative heat transfer is calculated in SOCRAT/V3 taking into account the *square* geometry of the rod bundle.

The maximum effective heat element radius $r_{\max}$ for square grid relevant to the QUENCH fuel assembly is equal to

$$r_{\max} = \frac{d}{\sqrt{\pi}} \tag{1}$$

where *d* is a pitch. This parameter is important for free volume calculations and the control of mass transfer in intact geometry and debris regions.

The nodalization scheme used for calculation of QUENCH-LOCA-0 experiment had 8 radial groups of heat structures and 18 axial meshes, most axial meshes are 0.1 m long in axial direction. The total modelling length was 1.875 m (from the lowest level -0.475 m up to highest level 1.4 m where the level 0 m corresponds to the low boundary of the heated region). The nodalization scheme includes necessarily the spacer grids and the periphery corner rods.

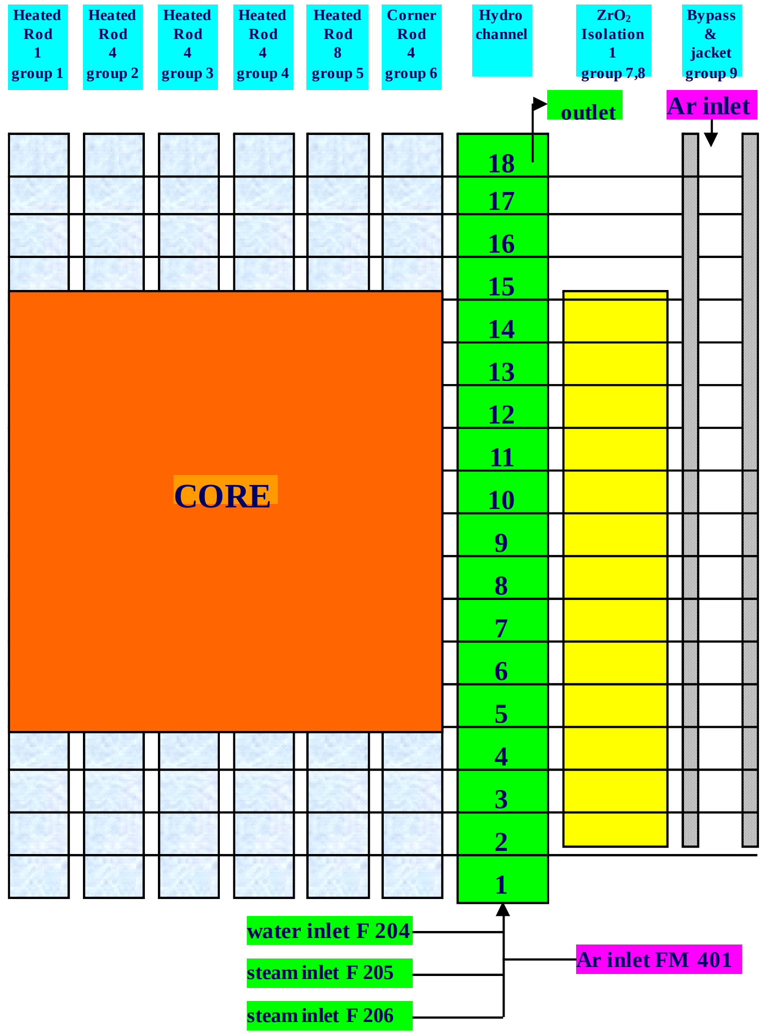

Figure 3: SOCRAT nodalization for QUENCH-LOCA-0

The thermal problem is mainly influenced by heat fluxes in a system. The thermal conductivity of the isolation is one of the most pronounced factors. In both tests the thermal conductivity data for the ZYFB-3 isolation [16] were used in the modelling.

4 BASIC PHASES OF QUENCH-16 TEST

Time sequence and main parameters of QUENCH-16 phases are presented in the Table 1.

Table 1. Phases of PARAMETER-SF4 experiment

Phase	Main parameters			
	FA temperature,°K	**Environment**	**Heating rate,°K/s**	**Time,s**
1. FA preliminary heating-up in steam-argon flow	820-1260	Steam-argon mixture (argon/steam flow rate is 3/3.4 g/s)	0÷0.33	0-3000

2. Stabilization of main parameters and FA preoxidizing	1260-1380	Steam-argon mixture (argon/steam flow rate is 3/3.4 g/s)	0÷0.05	3000-6300
3. FA cooldown	1380-930	Steam-argon mixture (argon/steam flow rate is 3/3.4 g/s)	–0.35	6300-7300
4. Air ingress	930-1870	Air-argon mixture (air/argon flow rate is 0.2/1 g/s)	0.17÷0.8	7300-11340
5. Bottom flooding of the assembly (when the assembly reached the temperature T_{max}≈1870°K)	Till complete cooling of the assembly	Water (flow rate of 50 g/s per assembly)	–35	11340-12080

The QUENCH-16 experiment consisted of five basic phases:

- First heat-up phase, mass flow rates $A_{steam} = 3.4 g/s$ and $A_{argon} = 3.0 g/s$, the heat-up to T≈1260°K in hot region;
- Pre-oxidation phase, the cladding temperature T≈1380°K in hot region;
- Cool-down phase (preparation to air ingress) with temperature drop to ~930°K;
- Air ingress phase with air mass flow rate $A_{air} = 0.2 g/s$ at inlet, the heat-up to T≈1380°K in hot region;
- Bottom flooding phase, water mass flow rate 50 g/s.

During first heat-up transient, pre-oxidation and cool-down phases, superheated steam together with the argon as carrier gas entered the test bundle at the bottom end and leaved the test section at the top together with the hydrogen that was produced in the zirconium-steam reaction. Both argon and steam flows entered the working section during heat-up transient, pre-oxidation and cool-down phases. The zirconium oxidation begins at these phases because the initiation of oxidation chemical reaction corresponded to temperature about 1000÷1200 K.

The cooldown phase was followed by the air ingress phase to prevent possible intensive oxidation of zirconium surfaces by the air with successive sharp temperature rise because of very large heat effect of zirconium-oxygen reaction.

The quench phase was initiated by turning off the argon and air flow, filling the lower plenum with quench water at a high rate, and injecting argon at the bundle head.

Fig. 4 demonstrates the main phases of the experiment SF4. The numbering of phases corresponds to the data of table 1.

5 RESULTS OF OF QUENCH-16 EXPERIMENT MODELLING

5.1 QUENCH-16 Input and Boundary Conditions

The mass flow rates of steam, argon and air as functions of time are depicted in Figure 5.

The total electric power in QUENCH-16 is presented in Figure 6. This Figure also includes the experimental bundle power corresponding to inner and outer rods electrodes.

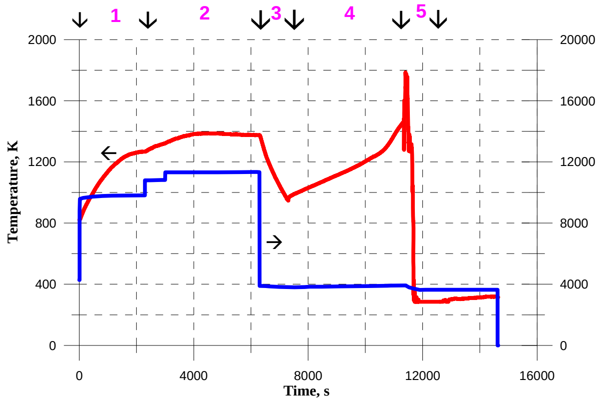

Figure 4: QUENCH-16 temperature and total power behaviour. Numbers of test phases are indicated

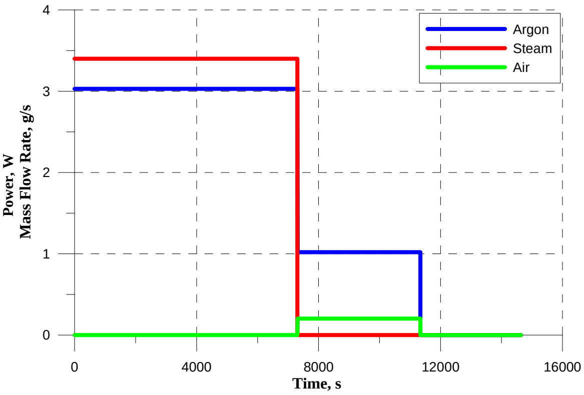

Figure 5: QUENCH-16 mass flow rates of steam, argon and air

The maximum total electric power in QUENCH-16 was about 11.5 kW as shown in Figure 6. Mass flow rate of flooding water is presented in Figure 7.

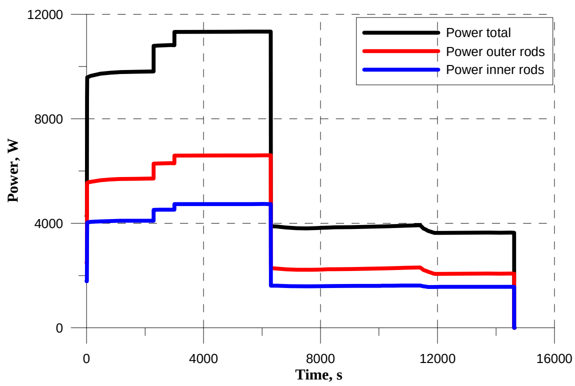

Figure 6: QUENCH-16 electric power (total, inner and outer rod rings) hystory

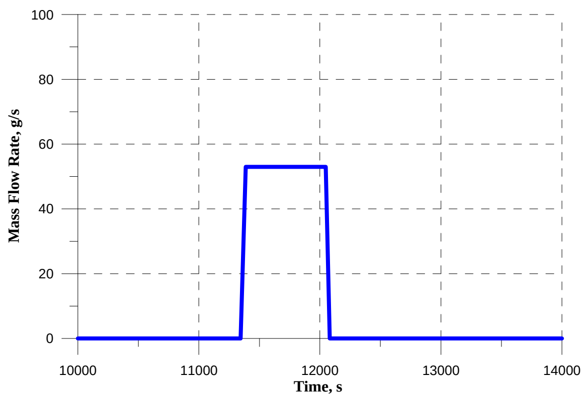

Figure 7: QUENCH-16 water mass flow rate at reflood

5.2 Modelling of Thermohydraulic Behaviour

The calculated and experimental bundle temperature at 950 mm elevation (near the upper part of heated zone) versus time for QUENCH-16 is presented in Figure 9. The maximum temperature about 1300 K was reached at this axial level. Figures 8-13 also show the temporal dependence of temperature for different axial locations: 1150, 850, 650, 550 and 50 mm.

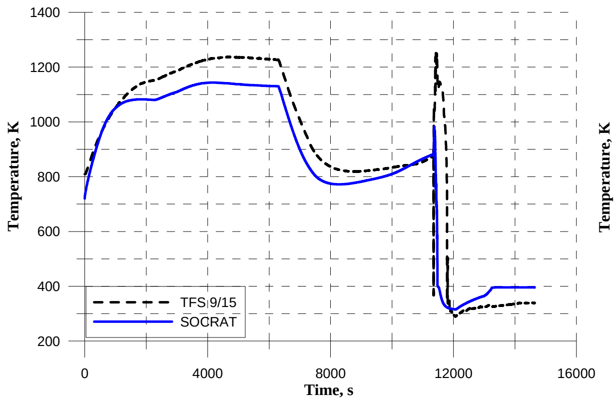

Figure 8: QUENCH-16: temperature at elevation 1150 mm

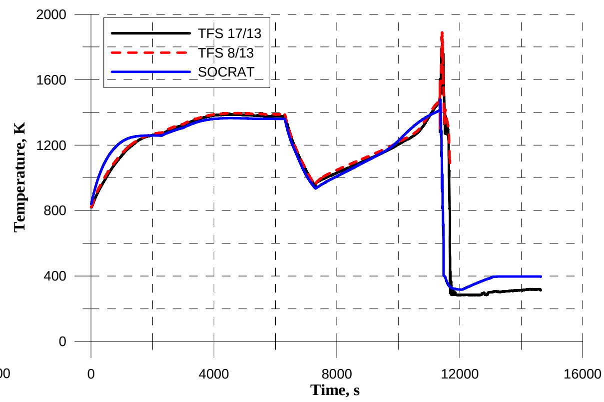

Figure 9: QUENCH-16: temperature at elevation 950 mm

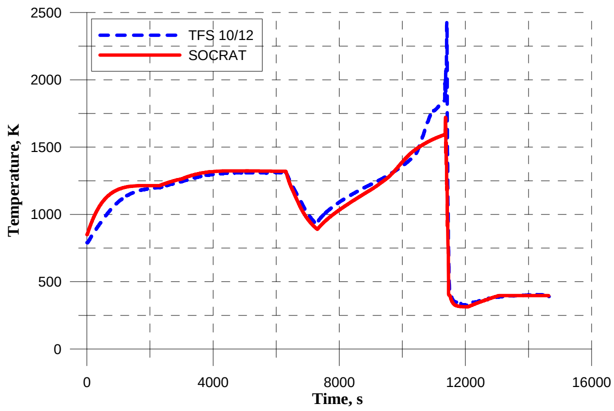

Figure 10: QUENCH-16: temperature at elevation 850 mm

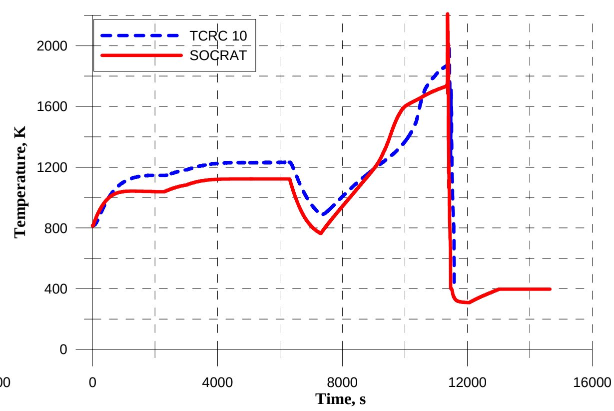

Figure 11: QUENCH-16: temperature at elevation 650 mm

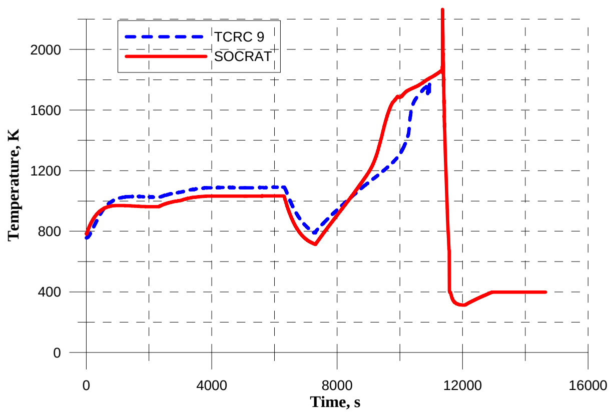

Figure 12: QUENCH-LOCA-0: temperature at elevation 550 mm

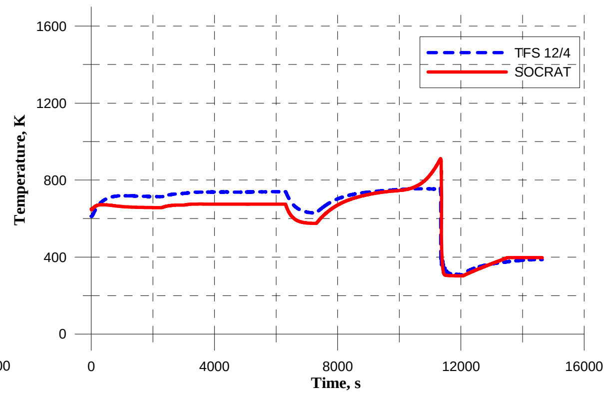

Figure 13: QUENCH-LOCA-0: temperature at elevation 50 mm

In Figure 14 the overall heat balance for the core (the heated part of the bundle) is presented. The heat transferred by steam-argon mixture dominates in comparison to the heat flux to shroud. The contribution of chemical heat is rather large in comparison to QUENCH_LOCA tests with respectively low temperatures.

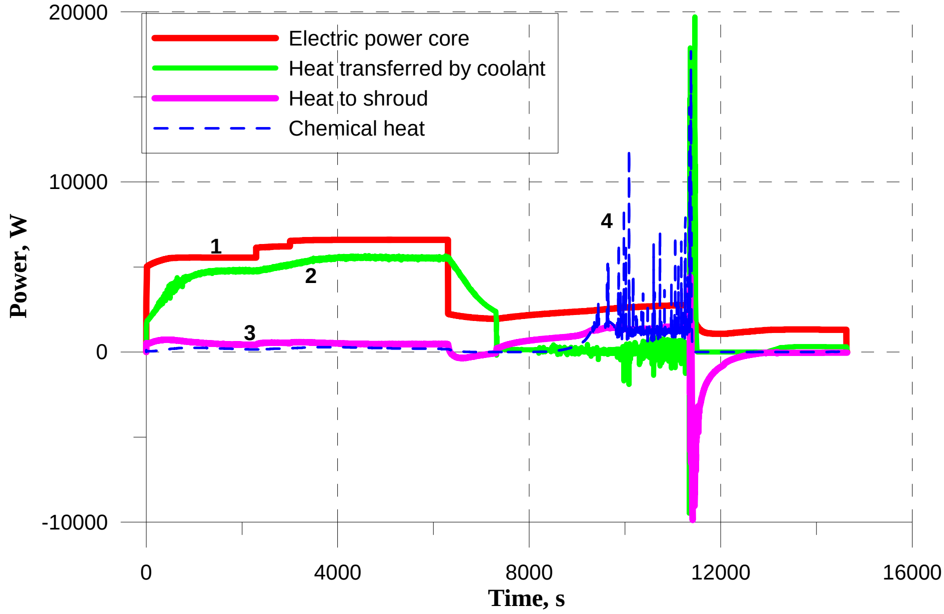

Figure 14: QUENCH-LOCA-0 calculation heat balance:
1 – total electric power,
2 – power transferred by gas,
3 – heat flux to shroud,
4 – chemical power

The basic thermal parameters of experiments QUENCH-16 are reasonably reproduced by the code. Because of respectively considerable influence of main radiative exchange parameters on thermal response, the adequacy of calculated and experimental data looks optimistic for justification of implemented radiation model [15].

5.3 Modelling of Oxidation

Calculated and experimental ZrO_2 layer thickness on the corner rod B withdrawn at time 7320 s is presented in Figure 15. One can note the reasonable consistency between predicted and measured data taking into account that SOCRAT considers both ZrO_2 and ZrO layer thicknesses, so, in calculations, the effective oxidised layer is thicker than pure ZrO_2 layer thickness.

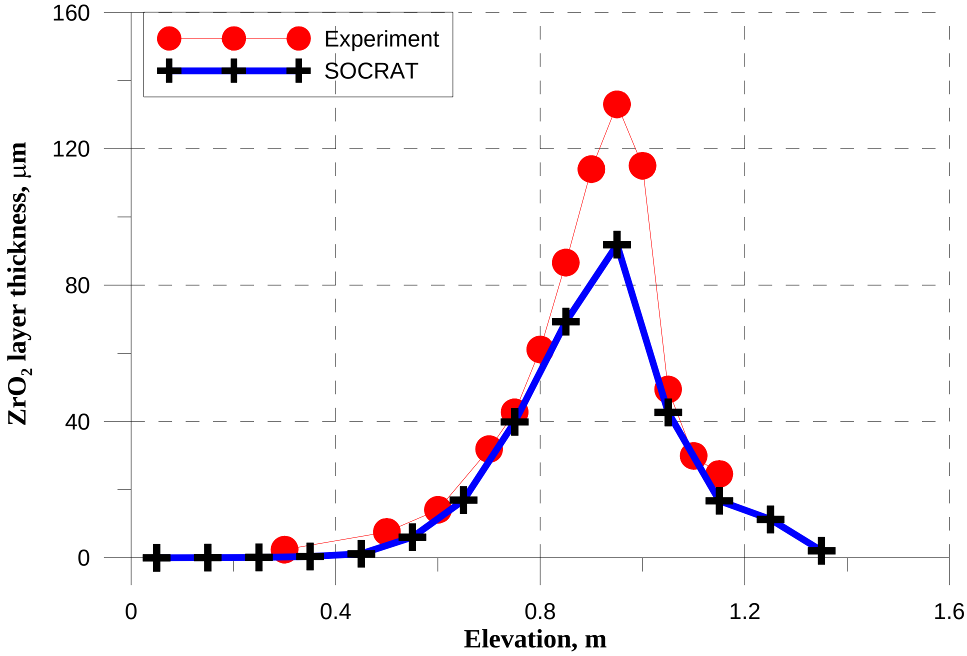

Figure 15: QUENCH-16 corner rod B ZrO_2 layer thickness at time 7320 s

6 AIR INGRESS PHASE FEATURES IN QUENCH-16 EXPERIMENT

The air ingress phase (Figure 4, phase 4) was very important phase of QUENCH-16 experiment. The thing is that the oxidation of zirconium claddings in the air behaves in different way in comparison to oxidation in the vapour. First, the heat effect of the chemical reaction of oxidation in the air is approximately two times larger than in the vapour. Second, the kinetics of oxidation in the air is non-parabolic (approximately linear, that is more strong) in contrast to parabolic kinetics of zirconium oxidation in the vapor.

This is why the temperature behavior at different elevations (Figures 16) is that there is the tendency to reach highest temperatures at medium or even at bottom elevations (300-500 mm from the bottom of heated region). This fact is in contrast to oxidation in the vapour (without air) tests where the highest temperatures were reached definitely at highest elevation 950 mm from the bottom of heated region.

The model of oxidation in the air in the code SOCRAT/V3 uses the same oxygen diffusivity coefficients as for oxidation in the vapour but takes into account different heat effect of this reaction. This consideration corresponds in both cases to parabolic correlation of oxidation.

Figure 14 shows the oxygen (which is a constituent part of the air) mass flow rate at the inlet and at the outlet part of the fuel assembly. One can see from this figure that the oxygen consumption grows as long as the cladding temperature becomes higher. Finally, the situation arises when all the oxygen entering the fuel assembly is consumed for oxidation of zirconium claddings. This state is called as total oxygen starvation. The total oxygen starvation was observed in three experiments QUENCH-16, QUENCH-10 [4] and PARAMETER-SF4 [14].

SOCRAT underestimates the time of total oxygen starvation in QUENCH-16. This may be due to overestimation of temperatures at medium levels (see Figures 10-12).

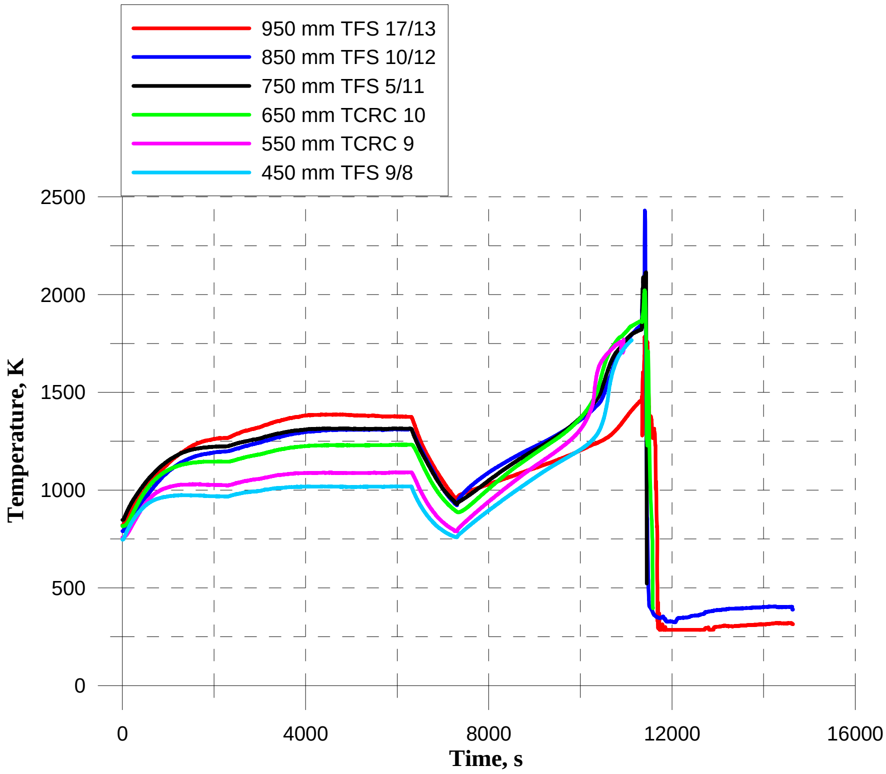

Figure 16: QUENCH-16 experimental temperature dynamics at different axial levels

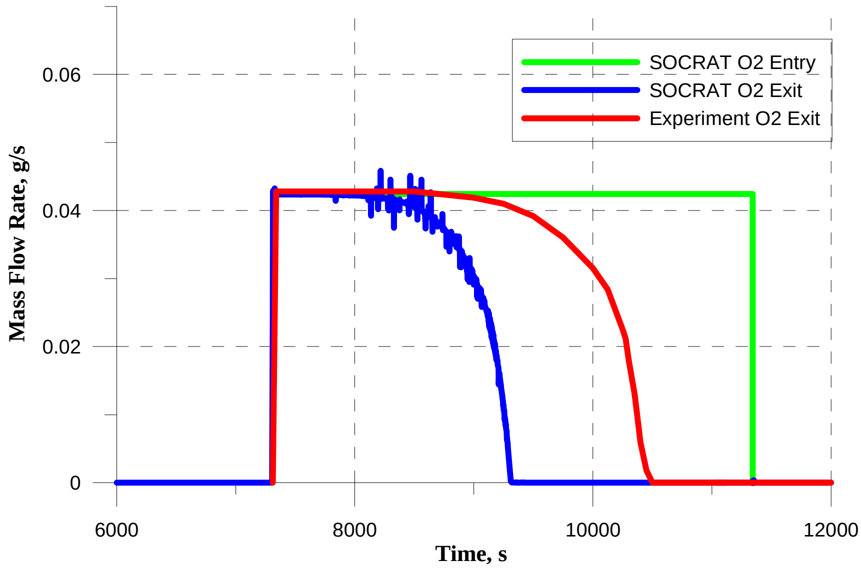

Figure 17: QUENCH-16: oxygen mass flow rate

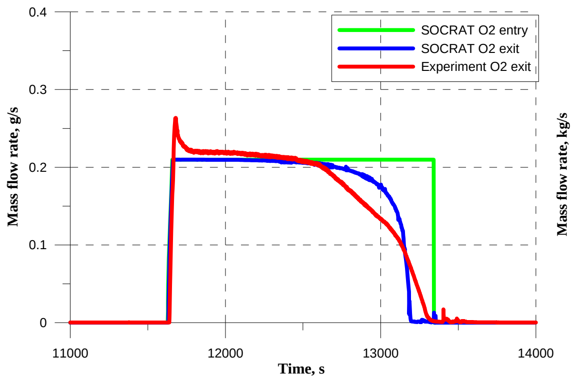

Figure 18: QUENCH-10: oxygen mass flow rate

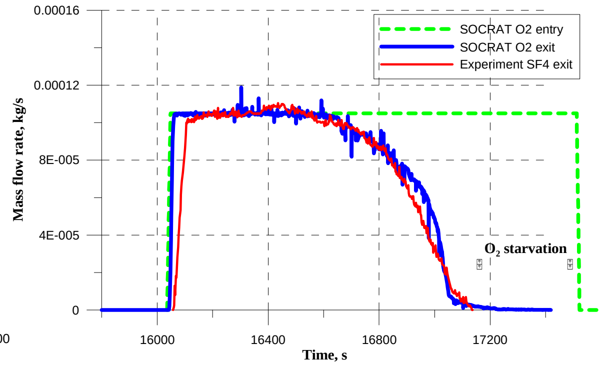

Figure 19: PARAMETER-SF4: oxygen mass flow rate

7 CONCLUSIONS

Posttest numerical modelling of QUENCH-16 test was performed using SOCRAT/V3 code. Results of thermal hydraulics and air ingress modelling are presented.

The air ingress phase was important QUENCH-16 test feature, drastically influencing on the test behaviour.

SOCRAT underestimates the time of oxygen starvation which may be connected with overestimation of oxidation by air at fuel assembly medium levels in calculations.

On the whole, the calculated and experimental thermal-hydraulic and chemical data are in a reasonable agreement, which shows the adequacy of modeling the complicated thermo-hydraulic behavior including the air ingress phase and the bottom reflood in the QUENCH-16 test.

At FA temperature less than 2000K the water reflood may be a good control measure to mitigate the consequences of beyond design basis accident at NPP with PWR.

NOMENCLATURE

ECCS	emergency core cooling system
FA	fuel assembly
LOCA	loss-of-coolant accident
NPP	nuclear power plant
PWR	pressurized water reactor
VVER	Russian type of pressurized water reactor

ACKNOWLEDGMENTS

The work has been performed in the frame of the cooperation agreement between IBRAE and KIT in the field of nuclear energy research.

REFERENCES

[1] P. Hofmann, C. Homann, W. Leiling *et al*., “Experimental and Calculational Results of the Experiments QUENCH-02 and QUENCH-03”, Forschungszentrum Karlsruhe, FZKA 6295, July 2000.

[2] W. Hering, Ch. Homann, J.-S. Lamy, “Comparison Report on the Blind Phase of the OECD International Standard Problem No. 45 Exercise (QUENCH-06),” Forschungszentrum Karlsruhe, FZKA 6677, 2002.

[3] C. Homann, W. Hering, G. Schanz, “Analysis and Comparison of Experimental Data of Bundle Tests QUENCH-07 to QUENCH-09 about B4C Control Rod Brehavior”, Forschungszentrum Karlsruhe, FZKA 7101, July 2006.

[4] G. Schanz, M. Heck, Z. Hozer *et al*., “Results of the QUENCH-10 Experiment on Air Ingress”, Forschungszentrum Karlsruhe”, FZKA 7087, SAM-LACOMERA-D09, May 2006.

[5] A. Stefanova, T. Drath, J. Duspiva *et al*., “SARNET Benchmark on QUENCH-11. Final Report”, Forschungszentrum Karlsruhe GmbH, Karlsruhe, Germany, FZKA 7368, SARNET CORIUM P008, 2008.

[6] J. Stuckert, M. Heck, A. Goryachev *et al*., “Results of the QUENCH-12 Experiment with a Russian VVER-type Bundle”, Forschungszentrum Karlsruhe, FZKA 7307, 2008.

[7] M. Steinbrück, M. Große, L. Sepold, J. Stuckert, “Synopsis and outcome of the QUENCH experimental program”, *Nuclear Engineering and Design*, **240**, 2010, pp. 1714–1727.

[8] J. Stuckert, J. Birchley, M. Große, T. Haste, L. Sepold, M. Steinbrück, “Experimental and post-test calculation results of the integral reflood test QUENCH-12 with a VVER-type bundle”, *Annals of Nuclear Energy*, **36**, 2009, pp. 183–192.

[9] J. Stuckert, J. Birchley, M. Grosse, B. Jaeckel, M. Steinbrück, “Experimental and calculation results of the integral reflood test QUENCH-14 with M5 cladding tubes”, *Annals of Nuclear Energy*, **37**, 2010, pp. 1036–1047.

[10] J. Stuckert, J. Birchley, M. Große, B. Jaeckel, M. Steinbrück, “Experimental and calculation results of the integral reflood test QUENCH-15 with ZIRLO cladding tubes in comparison with results of previous QUENCH tests”, *Nuclear Engineering and Design*, **241**, 2011, pp. 3224– 3233.

[11] Ju. Stuckert, A. Vasiliev, " Application of Thermal Hydraulic and Severe Accident Code SOCRAT/V3 to Bottom Water Reflood Experiment QUENCH-LOCA-0", Proc. Int. Conf. Nuclear Energy for New Europe 2011, Bovec, Slovenia, September 12-15, Nuclear Society of Slovenia, 2011, NENE-2011-911.

[12] A.D. Vasiliev, “Modeling of Thermal Hydraulics Aspects of Combined Top and Bottom Water Reflood Experiment PARAMETER-SF2 Using SOCRAT 2.1 Code”, Proc. 2008 ASME International Mechanical Engineering Congress and Exposition (IMECE2008), Boston, US, October 31- November 6, 2008, IMECE2008-69043.

[13] A.D. Vasiliev, "Application of Thermal Hydraulic Code SOCRAT/V2 to Top Water Reflood Experiment PARAMETER-SF3", Proc. 2009 ASME International Mechanical Engineering Congress and Exposition (IMECE2009), Lake Buena Vista, Florida, US, November 13-19, 2009, IMECE2009-10960.

[14] A.E. Kisselev, V.F. Strizhov, A.D. Vasiliev, Application of thermal hydraulic and severe accident code SOCRAT/V2 to bottom water reflood experiment PARAMETER-SF4. *Nucl. Eng. Des.* (2012), doi:10.1016/j.nucengdes.2012.01.016.

[15] G.V. Kobelev, V.F. Strizhov, A.D. Vasiliev, "Advanced Model of Radiative Heat Transfer in a Rod Geometry", Proc. 2007 ASME International Mechanical Engineering Congress and Exposition (IMECE2007), Seattle, US, IMECE2007-43375, November 11-15, 2007.

[16] www.zircarzirconia.com.