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Application of Thermal Hydraulic and Severe Accident Code SOCRAT/V3 to Bottom Water Reflood Experiment QUENCH-LOCA-0

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ABSTRACT

The thermal hydraulic and SFD (Severe Fuel Damage) best estimate computer modelling code SOCRAT/V3 was used for the calculation of QUENCH-LOCA-0 experiment.

The new QUENCH-LOCA bundle tests with different cladding materials will simulate a representative scenario of the LOCA (Loss of Coolant Accident) nuclear power plant accident sequence in which the overheated up to 1300 K reactor core would be reflooded from the bottom by ECCS (Emergency Core Cooling System).The first test QUENCH-LOCA-0 was successfully conducted at the KIT, Karlsruhe, Germany, in July 22, 2010, and was performed as the commissioning test for this series. The test bundle was made up of 21 fuel rod simulators which are placed in the square set. Heating is carried out electrically using tungsten heaters. The rod claddings are identical to that used in PWRs. The bundle was electrically heated in steam from 900 K to 1300 K with the heat-up rate of 2.2 K/s. After cooling in the saturated steam the bottom flooding with water flow rate of 90 g/s was initiated. The calculated results are in a good agreement with experimental data taking into account additional quenching due to water condensate entrainment at the steam cooling stage.SOCRAT/V3 was used for estimation of further steps in experimental procedure to reach a representative LOCA scenario in future tests.

1 INTRODUCTION

The lessons learned from severe nuclear accidents at Three Mile Island [1], US, 1979, Chernobyl, USSR, 1986, and Fukushima, Japan, 2011, showed the very high importance of accident control measures to prevent the development of design basis accident to beyond design basis accident and to mitigate the consequences of beyond design basis accident. The deep understanding of hydraulic, mechanical and chemical processes taking place under accident conditions is necessary, in particular, under LOCA nuclear power plant accident sequence conditions.

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

The experiment QUENCH-LOCA-0 was the commissioning test from the series of several tests within the new QUENCH-LOCA program. The overall objective of this bundle test series is the investigation of ballooning, burst and secondary hydrogen uptake of the cladding under representative *design basis accident* conditions. The various planned experiments will examine the effect of different cladding materials, geometric configurations and pre-hydriding.

The QUENCH-LOCA experiments will contribute to the database on PWR (pressurized water reactor) severe accident phenomena obtained in QUENCH experimental program. Those tests [2-5] aimed at studying mechanical and physical and chemical behaviour of overheated fuel rod cladding with quenching from bottom. Thanks to QUENCH tests, a good understanding of *beyond design basis* accident processes and phenomena has been achieved.

The experiment QUENCH-LOCA-0 was successfully conducted at the KIT in July 22, 2010, with the aim to study the 21-rod model fuel assembly (FA) of PWR under the simulated conditions of LOCA accident involving the stage of the high rate cooling with the bottom flooding [6]. Each rod was separately pressurized with krypton gas with initial pressures of 35, 40, 45, 50, and 55 bar for different rods. The investigation included:

- the study of thermo-hydraulic phenomena including flooding;
- the study of thermo-mechanical phenomena (ballooning and burst);
- the study of physico-chemical phenomena (hydrogen generation and secondary hydrogen uptake);
- 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.

At the transient, the bundle was overheated up to 1330 K. At the time 220 s from the beginning of the test, the bottom quench water injection was initiated, the water flow rate was ~ 90 g/s.

The best estimate computer modelling code SOCRAT/V3 was used for the calculation of QUENCH-LOCA-0 experiment. SOCRAT/V3 has been verified on many severe accident experiments, in particular, on test series PARAMETER [7]. 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-LOCA-0 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 [8] to adequately estimate the heat transfer in the fuel assembly.

The QUENCH-LOCA-0 calculated results obtained using SOCRAT/V3 are compared to experimental data. The injection of steam at slow cooldown stage resulted in rapid cooling which was caused by entrainment of condensed water (the water mass was about 3 kg). The calculated and experimental data are in a good agreement taking into account this additional quenching, which is indicative of the adequacy of modelling the complicated thermo-hydraulic behaviour in the QUENCH-LOCA-0 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-LOCA-0 test bundle (Figure 1) is made up of 21 fuel rod simulators with a length of approximately 2.48 m (heated rod simulators), which are hold together by means of five spacer grids. The rods are placed in the *square* set (Figure 2). 21 fuel rod simulators are heated over a length of 1024 mm. Heating is carried out electrically using 6-mm-diameter tungsten heaters. For the heated rods, tungsten heating elements are installed in the centre of the rods and are surrounded by annular ZrO_2 pellets. The tungsten heaters are connected to electrodes made of molybdenum and copper at each end of the heater.

The test bundle is 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 reduce the coolant channel area to a representative value.

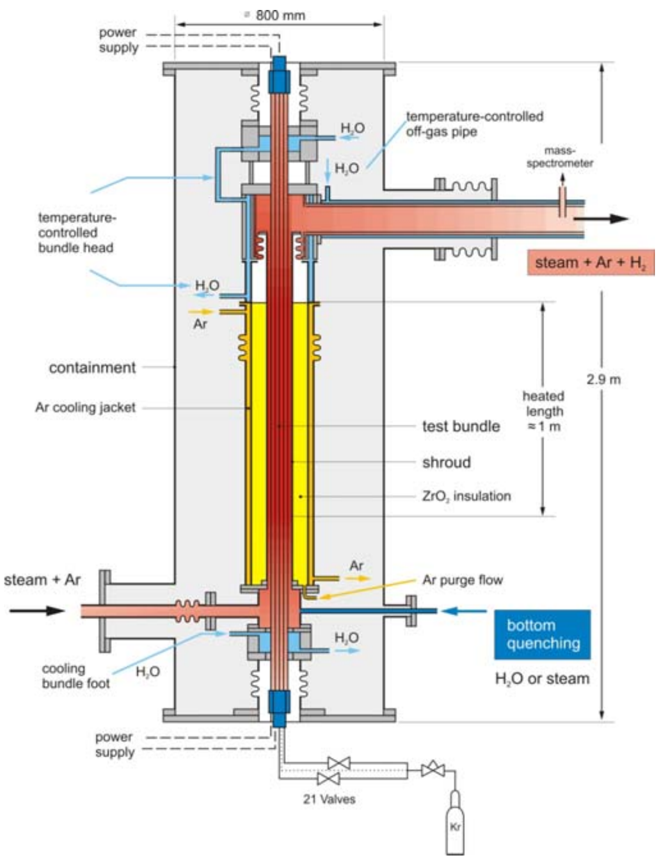

Figure: 1: Schematic representation of QUENCH test section facility

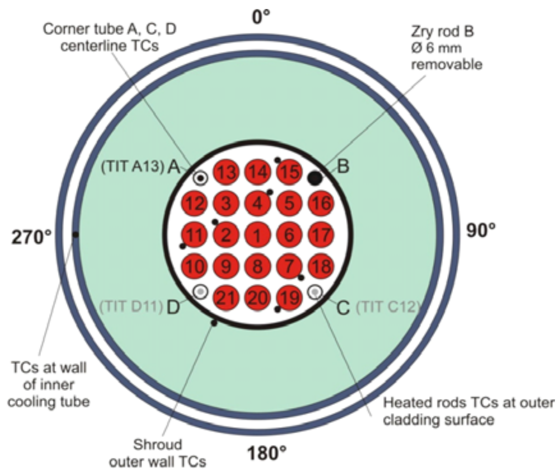

Figure 2: Cross-section of QLOCA-0 test bundle (21 heated, 4 corner rods). Consecutive numbers of rods are indicated

The rod cladding is identical to that used in LWRs: Zircaloy-4, 10.75 mm outside diameter, 0.725 mm wall thickness. The rod simulators were filled with krypton and had internal pressures between 35 and 55 bar. The test bundle is 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 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.

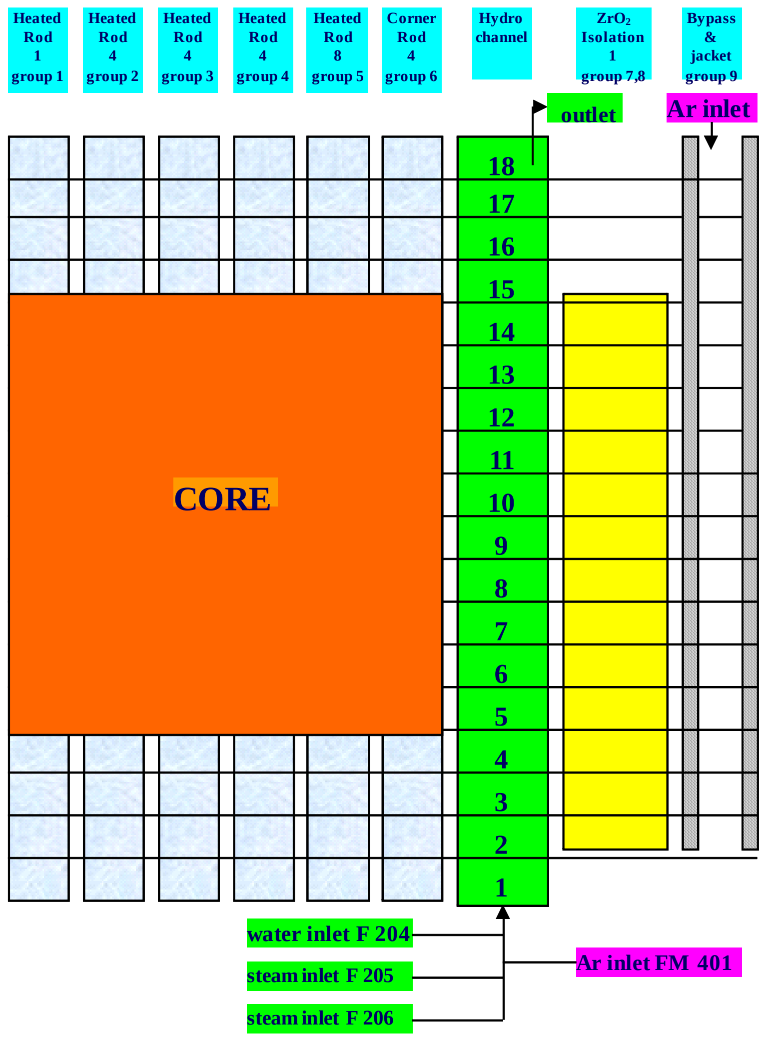

Figure 3: SOCRAT nodalization for QUENCH-LOCA-0

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 [9] were used in the modelling.

4 RESULTS OF OF QUENCH-LOCA-0 EXPERIMENT MODELLING

4.1 QUENCH-LOCA-0 Test Scenario

The QUENCH-LOCA-0 experiment was planned to consist of four basic phases:

- Preliminary heating in argon flow (mass flow rate of 6 g/s, gas temperature 570 K), the heat-up to the peak cladding temperature of 900 K;
- Heating-up in steam-argon mixture flow (mass flow rates of 2 g/s for steam and 6 g/s for argon, gas temperature 570 K), increasing peak cladding temperature up to 1300 K;
- Cooldown phase with temperature drop to about 800 K;
- Bottom flooding phase with water mass flow rate ca. 50 g/s.

Time sequence and main parameters of QUENCH-LOCA-0 phases are presented in Table 1.

During the heat-up transient phase, superheated steam together with the argon as carrier gas enter the test bundle at the bottom end and leaves the test section at the top together with the hydrogen that is produced in the zirconium-steam reaction. The total amount of hydrogen released during QUENCH-LOCA-0 experiment was about 1 g.

The slow cooldown is achieved by saturated steam with mass flow rate of 50 g/s.

Table 1: Scheduled phases of QUENCH-LOCA-0 experiment

Phase	Main parameters			
	FA peak temperature, K	**Environment**	**Heating rate, K/s**	**Time, s**
0. Preliminary FA heating in argon flow	270-900	Argon flow at temperature up to 570 K (argon flow rate is 6 g/s)	0.3	-10000–0
1. FA heating-up in the flow of steam-argon mixture (transient phase)	900-1300	Steam-argon mixture (argon/steam flow rate is 6/2 g/s)	8.0	0–60
2. FA slow cooldown (when the assembly reaches temperature T_{max}≈1300 K)	1300-800	Saturated steam-argon mixture (argon/steam flow rate is 6/50 g/s)	-5.0	60–180
3. Bottom flooding of the assembly (when the assembly reaches temperature T_{max}≈800 K)	Till complete cooling of the assembly	Water (flow rate of 50 g/s per assembly)	-50	180–700

The quench phase was initiated by turning off the argon and steam flow, and injecting argon at the bundle head.

Figure 4 demonstrates the main scheduled phases of the QUENCH-LOCA experiment. The numbering of phases corresponds to the data of Table 1.

The corresponding scheduled total electric power in QUENCH-LOCA-0 is presented in Figure 5. This Figure also includes the calculated bundle power corresponding to electric power generation in the bundle and in the heated core region corresponding to axial levels 0–1000 mm.

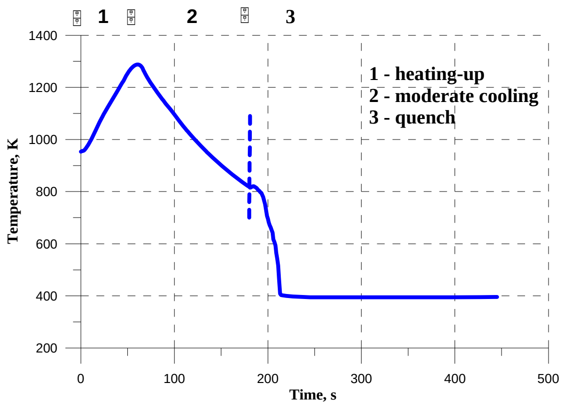

Figure 4: QUENCH-LOCA-0 scheduled characteristic temperature behaviour. Numbers of test phases are indicated

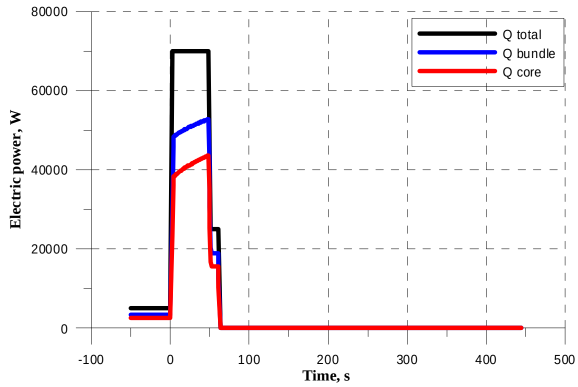

Figure 5: QUENCH-LOCA-0 scheduled characteristic power (total, bundle and core) history

Due to technical limitations, the real maximum total electric power in QUENCH-LOCA-0 was about 43 kW as shown in Figure 6. Mass flow rates of argon, steam and flooding water are presented in Figure 7.

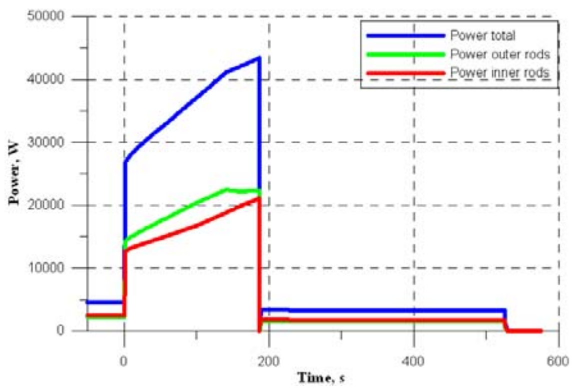

Figure 6: QUENCH-LOCA-0 electric power (total, inner and outer rings) hystory

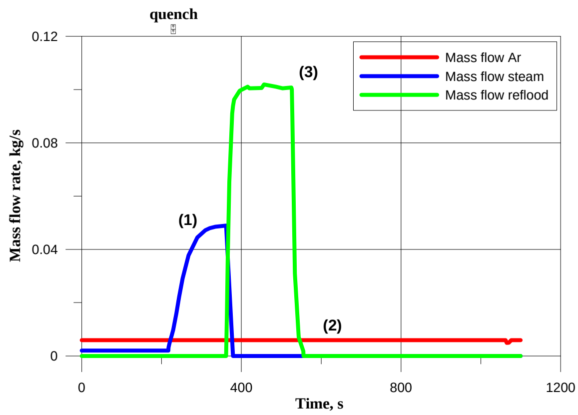

Figure 7: QUENCH-LOCA-0 mass flow rates of steam (1), argon (2) and water (3)

4.2 Modelling of Thermohydraulic Behaviour

To take into account additional cooling by entrainment of condensed water during phase 2 the mass flow rate of saturated water at this stage was defined in calculations as equal to saturated steam rate at this stage (Figure 7, line 1). This value is in a good agreement with the experimental estimation of water mass.

The calculated and experimental bundle temperature at 950 mm elevation (near the upper part of heated zone) versus time for QUENCH-LOCA-0 is presented in Figure 8. The maximum temperature about 1300 K was reached at this axial level. Figures 9-13 show the temporal dependence of temperature for different axial locations: 1150, 750, 550, 450 and 300 mm.

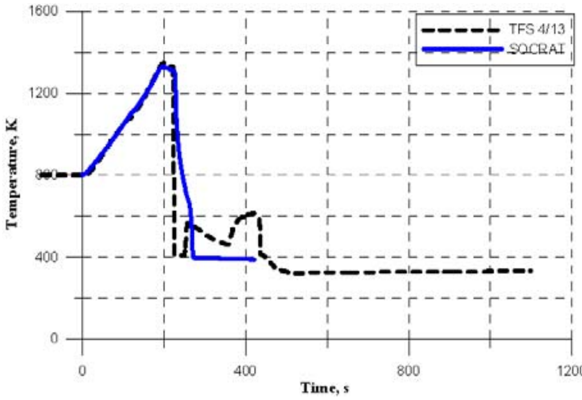

Figure 8: QUENCH-LOCA-0: temperature at elevation 950 mm

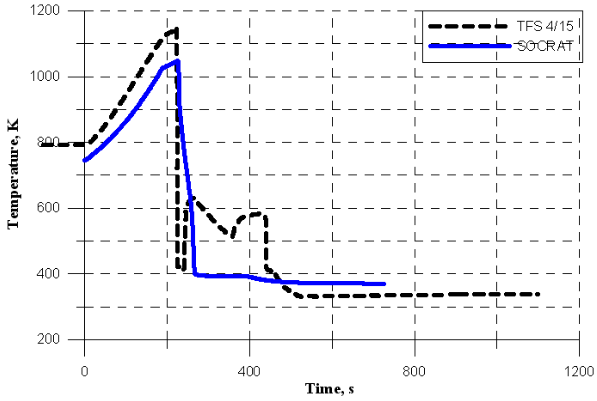

Figure 9: QUENCH-LOCA-0: temperature at elevation 1150 mm

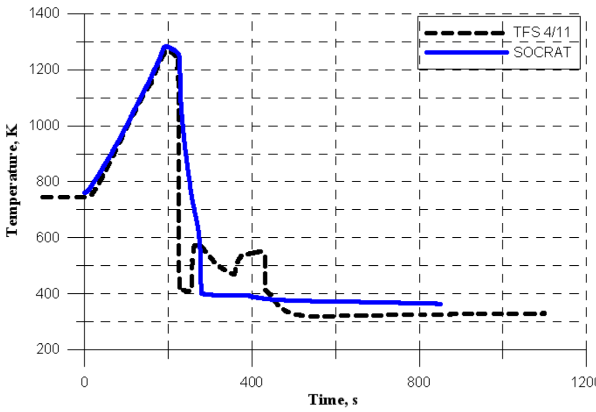

Figure 10: QUENCH-LOCA-0: temperature at elevation 750 mm

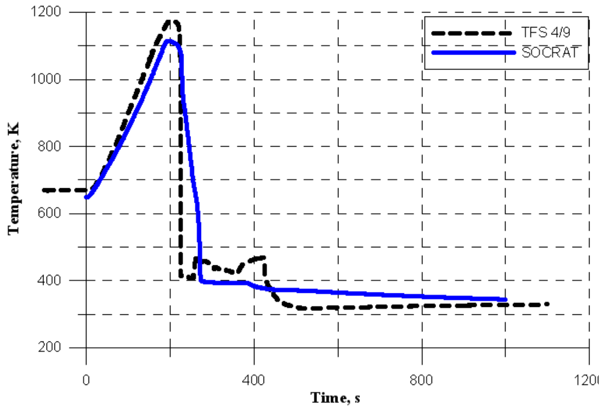

Figure 11: QUENCH-LOCA-0: temperature at elevation 550 mm

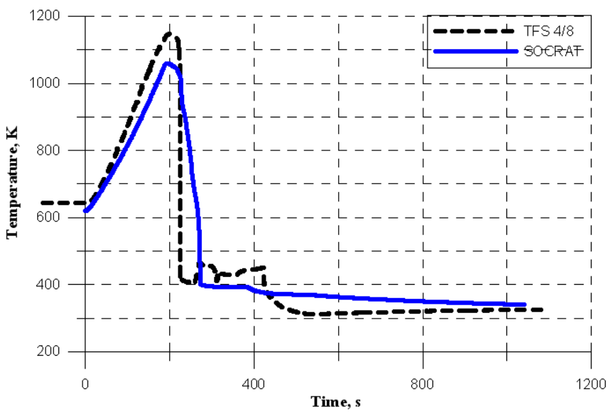

Figure 12: QUENCH-LOCA-0: temperature at elevation 450 mm

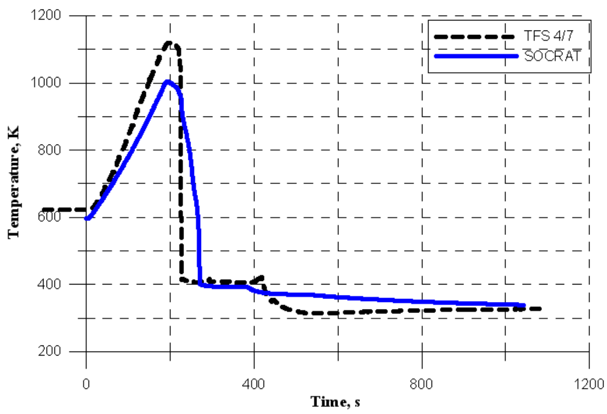

Figure 13: QUENCH-LOCA-0: temperature at elevation 350 mm

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, which is the opposite case to severe accident QUENCH tests with higher temperatures obtained than in QUENCH-LOCA-0. The contribution of chemical heat is now rather small in comparison to severe accident QUENCH tests.

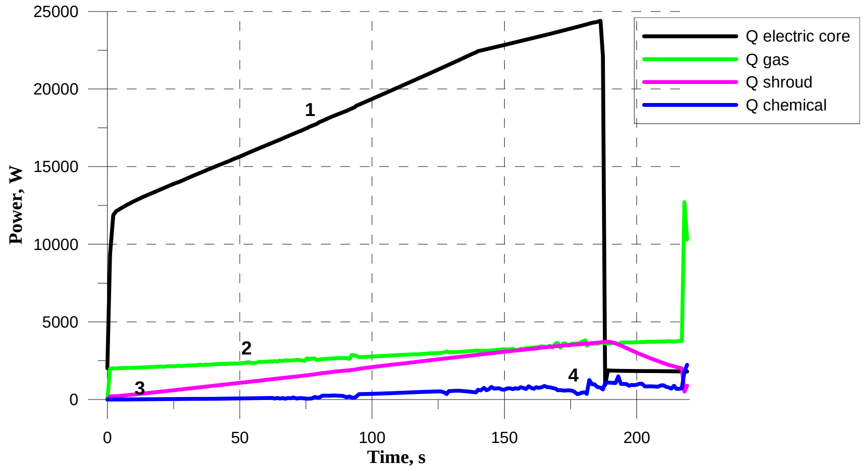

Figure 14: QUENCH-LOCA-0 calculation heat balance:
1 – total electric power,
2 – power transferred by gas,
3 – heat flux to shroud,
4 – chemical power

The basic thermal parameters of experiments PARAMETER-SF4 are adequately 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 [8].

4.3 Modelling of Hydrogen Generation

The experimental value for integral hydrogen production in QUENCH-LOCA-0 is about 1 g.

Calculated hydrogen rate is presented in Figure 15. Calculated hydrogen integral production for QUENCH-LOCA-0 is shown in Figure 16. Final calculated value for H_2 production is estimated as 0.7 g, so this value is in a reasonable consistency with experimental results.

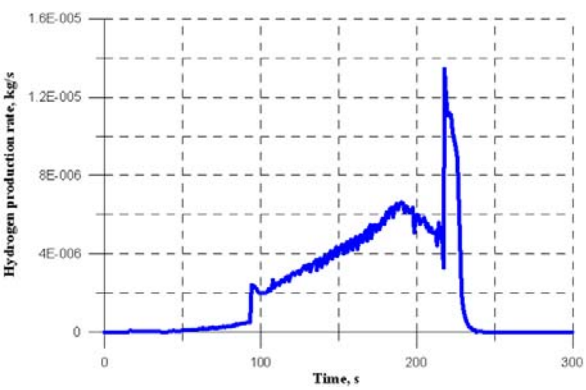

Figure 15: QUENCH-LOCA-0: calculated hydrogen generation rate

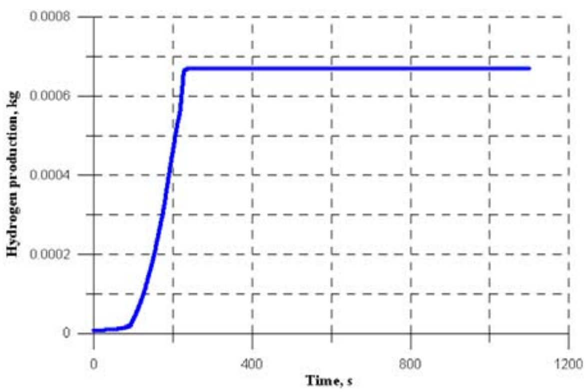

Figure 16: QUENCH-LOCA-0: calculated integral hydrogen generation

4.4 Modelling of Thermo-Mechanical Behaviour

The important purpose of QUENCH-LOCA-0 test was the investigation of thermo-mechanical behavior of the claddings. Burst time measured were between 111 s (rod #1) and 174 s (rod #10) after initiation of the transient phase. Burst temperatures were between 1064 and 1141 K.

Figure 17 shows experimental and calculated burst time for every rod used in the bundle. One can see that for the inner row rods the code predicts burst times slightly more in comparison to experimental ones. This circumstance shows that SOCRAT thermo-mechanical model with the code parameter CROX=1 (default value) uses such a burst criteria which overestimate burst time. The parameter CROX=1 denotes that the thermo-mechanical models of PWR-type Zry rod claddings are activated in SOCRAT while CROX=2 is used for the description of Russian VVER-type rod claddings behavior.

Besides that, Figure 17 clearly shows that the calculated outer rods burst times are slightly less in comparison to experimental values. In our opinion, it doesn't indicate unadequate working of thermo-mechanical model. It rather shows inaccuracies of one-dimensional thermal model. The thing is that the measured temperatures of outer rods were considerably lower (by 100 K and even slightly more) than the temperatures of inner rods due to noticeable radial heal flow to the shroud.

The current version of SOCRAT cannot consider these two-dimensional effects in principle. The difference between the temperatures of inner and outer rows is considerably lower in the calculations (about several tens of degrees). It results to earlier burst times in calculations for the rods of outer rows.

The reciprocal dependency of the cladding burst temperature on the loading pressure is well known (e.g. [10]) and was implementing in the modeling. Figure 18 presents corresponding thermo-mechanical results for the QUENCH-LOCA-0: the dependency of the temperature at burst time at elevation 950 mm (hottest axial level) on initial rod pressures (remember that the rods with different initial pressures from 35 to 55 bar were used in the experiment). We should take into account the inaccuracies in thermal hydraulic modelling, which describes very roughly the radial temperature distribution and gives the burst temperature very approximately. However, the results of comparison between calculated and experimental values look optimistic.

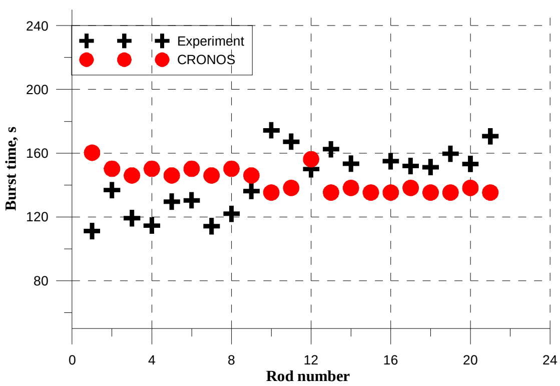

Figure 17: QUENCH-LOCA-0: burst time dependence on the rod number

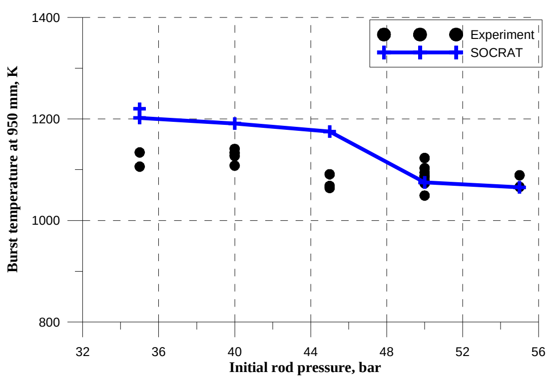

Figure 18: QUENCH-LOCA-0: the temperature at burst in dependence of initial rod pressure

The calculated and experimental pressure in rods with different initial pressures is shown in Figure 19. The definition "ring=1, p=50 bar" denotes that the rod is placed in radial ring 1 and has the initial pressure of 50 bar. The code predictions have a tendency to overestimate the pressure in some extent. On the whole, however, the code models thermo-mechanical behaviour rather well.

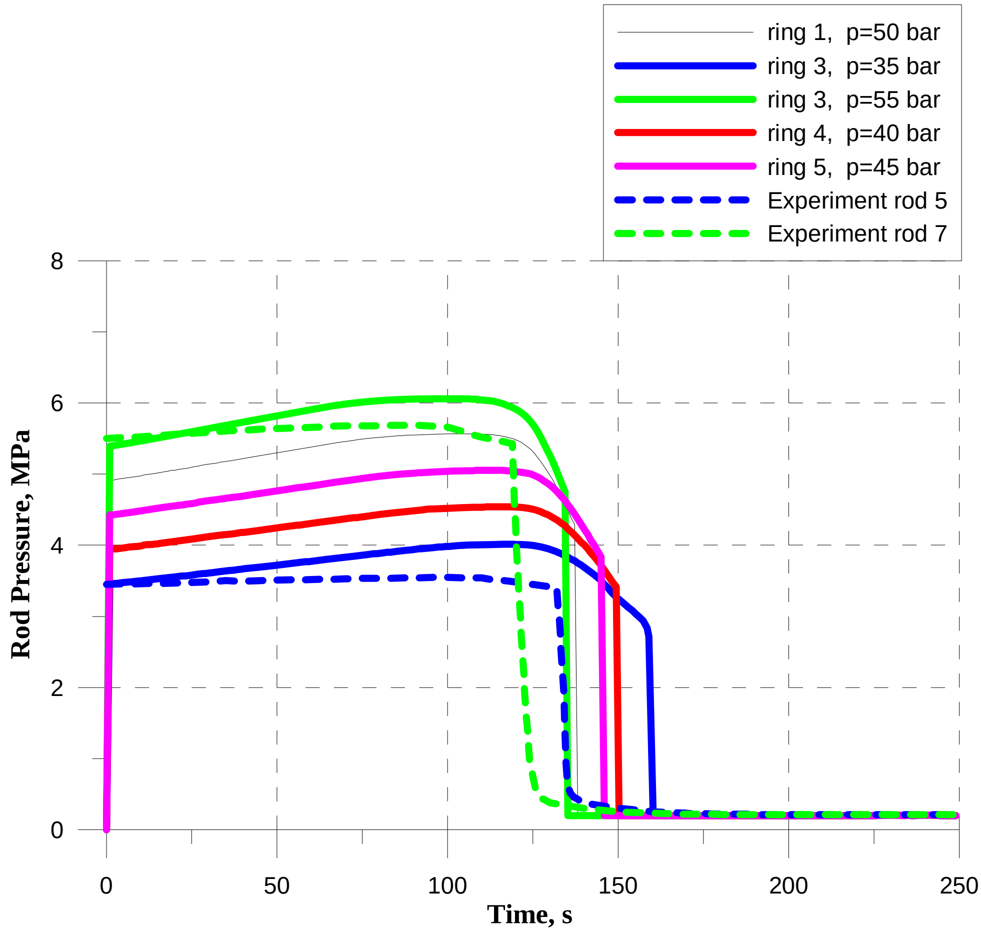

Figure 19: QUENCH-LOCA-0: the calculated pressure in rods placed in corresponding radial rings with different initial pressures

5 CONCLUSIONS

Posttest numerical modeling of QUENCH-LOCA-0 test was performed using the SOCRAT/V3 code. Results of both thermal hydraulics and thermal mechanics modeling are presented.

The calculated results are in a good agreement with experimental data taking into account additional quenching due to water condensate entrainment at the cooldown stage.

Further steps in experimental procedure are highlighted to reach a representative LOCA scenario in future tests.

To reach high heat-up rate (8 K/s) it is desirable to use upper limit of QUENCH assembly power of 70 kW.

To get moderate temperature cool-down rate it is desirable to use high mass flow rate of steam of 50 g/s (573 K – temperature of steam)

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