ABSTRACT

The PARAMETER-SF4 test conditions simulated a severe LOCA (Loss of Coolant Accident) NPP (nuclear power plant) sequence in which the overheated up to 1700 - 2300K core would be reflooded from the bottom in occasion of ECCS (Emergency Core Cooling System) recovery. The test was successfully conducted at the NPO “LUTCH”, Podolsk, Russia, on July 21, 2009, and was the fourth of four experiments of series PARAMETER-SF.

The PARAMETER facility of NPO “LUTCH”, Podolsk, is designed for studies of the VVER fuel assemblies behavior under conditions simulating design basis, beyond design basis and severe accidents.

After the maximum cladding temperature of about 1900K was reached in the bundle during PARAMETER-SF4 test, the bottom flooding was initiated.

The thermal hydraulic and SFD (Severe Fuel Damage) best estimate computer modelling code SOCRAT/V2 was used for the calculation of PARAMETER-SF4 experiment.

The important feature of PARAMETER-SF4 test 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.

Thermal hydraulics in PARAMETER-SF4 experiment played very important role and its adequate modeling is important for the thermal analysis. The results obtained by SOCRAT/V2 were compared with experimental data concerning different aspects of air ingress phase and thermal hydraulics behavior during the reflood. The temperature experimental data were found to be in a good agreement with calculated results. It is indicative of the adequacy of modeling the complicated thermo-hydraulic behavior in the PARAMETER-SF4 test.

1 INTRODUCTION

The experiment PARAMETER-SF4 was the second test from the series of two tests within the ISTC (International Scientific and Technical Centre) 3690 Project. The conditions of the PARAMETER-SF4 test modeled the initial stage of severe accident with large break LOCA with the core drying, its heating-up to about 1900K, air ingress and bottom water flooding in the case of the recovery of ECCS. The experiment PARAMETER-SF4 reproduced for the most part the basic phases of PARAMETER-SF1, PARAMETER-SF2 and PARAMETER-SF3 test which were successfully done some time earlier [1,2]. The main difference between the tests is that in SF4 only bottom flooding was realized according to a
technical programme instead of the combined flood in SF2 and only top flooding in SF1 and SF3.

Consequently, it was interesting to compare thermal-hydraulic behavior during reflood in these three tests SF2, SF3 and SF4, and the effect of air ingress in SF4.

Out-of-pile experiments to study severe fuel damage of bundles assembled with uranium dioxide fuel pellets (CORA-W1, CORA-W2) were performed in 1993 by the research center in Forschungszentrum Karlsruhe [3,4] as a close cooperative effort undertaken by the Russian and foreign organizations to study Russian VVER-1000 fuel assembly behavior at the early stage of core damage. These experiments demonstrated the absence of major differences in trends of the severe damage progression for fuel assemblies of VVER-1000 reactors and western PWR reactors. But the mentioned experimental program did not include a study of mechanical and physical and chemical behavior of fuel rods under flooding conditions. The problem of the reactor core structural material behavior under severe accident conditions with the flooding from the top and combined flooding from the top and bottom has not been sufficiently studied as well.

In this connection, the objective of PARAMETER-SF1 [5] and PARAMETER-SF2 [6] tests investigated recently as well as the PARAMETER-SF4 test considered in this paper is the experimental investigation of VVER-1000 fuel rod assemblies (made of standard structural materials used for VVER-1000 - Zr1%Nb-alloy fuel cladding, uranium dioxide fuel pellets and guiding tubes made of Zr1%Nb alloy) behavior under simulated conditions of a severe accident including the stage of low rate top flooding (SF1), combined top and bottom flooding (SF2) and top flooding (SF3).

At present, similar experiments are underway in the research center Forschungszentrum Karlsruhe, Germany. Those tests are under the QUENCH experimental program [7-12] aimed at studying mechanical and physical and chemical behavior of overheated fuel rod cladding with quenching from bottom. However, these experiments used zirconium dioxide simulator pellets instead of fuel pellets made of uranium dioxide. This is insufficient to describe the process of high-temperature interaction of the core materials under severe accident conditions.

The experiment SF4 was conducted with the aim to study the 19-rod model FA of VVER-1000 under the simulated conditions of severe accident including the stages of air ingress and high rate cooling with the bottom flooding, and namely:

- the study of the behavior of structural components of 19-rod model FA of VVER-1000 (fuel pellets and claddings, shroud, spacing grids);
- the study of the oxidizing degree of the structural components of 19-rod model FA of VVER-1000;
- the study of interaction and structural-phase changes in the materials of model FA of VVER-1000 (fuel pellets and claddings);
- the study of the hydrogen release dynamics.

The test bundle was made up of 19 fuel rod simulators with a length of approximately 3.12 m (heated rod simulators) and 2.92 m (unheated rod simulator). Heating was carried out electrically using 4-mm-diameter tantalum heating elements installed in the center of the rods and surrounded by annular UO2 pellets. The rod cladding was identical to that used in VVERs: Zr1%Nb, 9.13 mm outside diameter, 0.7 mm wall thickness.

The SF4 test was successfully conducted at the NPO “LUTCH”, Podolsk, Moscow region, Russia on July 21, 2009.

At the transient, the bundle was overheated approximately to 1900K. Then, at the time 17430 s from the beginning of the test, the bottom quench water injection was initiated, the water flow rate was ~ 50 g/s. The temperature was reached the value of about 2300K after the reflood initiation. The bundle quenching was successful.
The best estimate computer modelling code SOCRAT/V2 was used for the calculation of PARAMETER-SF4 experiment. SOCRAT code consists of two major modules: RATEG - thermal hydraulics calculation, SVECHA – severe fuel damage phenomena description.

The two-phase water-steam thermal hydraulics behavior under reflood conditions is very interesting and complicated issue. Another important thermal process in PARAMETER-SF4 test is the radiative heat transfer in the triangular rod bundle relevant of VVER FA. This is why advanced model of radiative heat exchange was implemented to SOCRAT code [13] to adequately estimate the heat transfer in the fuel assembly.

The calculated results obtained using SOCRAT/V2 are compared to experimental data. The calculated and experimental data are in a good agreement, which is indicative of the adequacy of modeling the complicated thermo-hydraulic behavior in the PARAMETER-SF4 experiment.

2 PARAMETER FACILITY

PARAMETER facility at NPO “LUTCH”, Podolsk, Russia, is designed for studies of the VVER fuel assemblies behavior under conditions simulating design basis, beyond design basis and severe accidents in nuclear power plants.

The test bundle (Figure 1) of the PARAMETER-SF4 test was similar to SF2 [6] and was made up of 19 fuel rod simulators with a length of approximately 3.12 m (heated rod simulators) and 2.92 m (unheated rod simulator). The rods were placed in the triangular set. 18 fuel rod simulators were heated over a length of 1275 mm, the one unheated fuel rod simulator was located in the center of the test bundle. Heating was carried out electrically using 4-mm-diameter tantalum heating elements installed in the centre of the rods and surrounded by prototypic annular UO2 pellets.

The rod cladding was identical to that used in VVERs: Zr1%Nb, 9.13 mm outside diameter, 0.7 mm wall thickness. The test bundle was instrumented with thermocouples attached to the cladding and the shroud at 18 different elevations with an axial distance between the thermocouples of 100 mm for most locations.

The unheated rod simulator was filled with pellets of UO₂ (bore size 1.2mm internal diameter). For the heated rods 6mm diameter tungsten heating elements were installed in the center of the rods and were surrounded by annular UO₂ pellets (bore size 4.2mm internal diameter). The tantalum heaters were connected to electrodes made of molybdenum and copper at each end of the heater.

3 RESULTS OF PARAMETER-SF4 TEST MODELING

All experiments of PARAMETER series consist of several phases: a first heat-up phase, a thermal equilibrium phase, a second heat-up phase, a pre-oxidation phase, a transient phase, and a quenching phase.

The PARAMETER-SF4 experiment consisted of five basic phases (Figure 2):
- First heat-up (preparatory) phase, the stabilization of steam and Ar mass flow rates at cladding temperatures (mass flow rates \( A_{\text{H₂O}} \approx 3 \text{ g/s} \) and \( A_{\text{Ar}} \approx 2 \text{ g/s} \), \( T \approx 500^\circ\text{C} \)), the heat-up to \( T \approx 1000^\circ\text{C} \) in hot region;
- Pre-oxidation phase, the cladding temperature \( T \approx 1150^\circ\text{C} \) in hot region;
- Cooldown phase with temperature drop to \( \approx 700^\circ\text{C} \);
- Air ingress phase with air mass flow rate at inlet \( A_{\text{Ar}} \approx 0.5 \text{ g/s} \);
- Bottom flooding phase, water mass flow rate 50g/s.
During the preliminary heating and heat-up transient phases, superheated steam together with the argon as carrier gas enters 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 quench phase was initiated by turning off the argon and steam flow, filling the lower plenum with quench water at a high rate, and injecting argon at the bundle head.

The water cooling system (water cooling jacket) was turned on at the time about 3500s from the beginning of the test. As a result, the external steel containment began to be cooled by the water.

The total amount of hydrogen released during PARAMETER-SF4 experiment was 105g.

Figure 2 demonstrates the main phases of the experiment SF4. The electric power in PARAMETER-SF4 is presented in Figure 3. It includes the calculated bundle power corresponding to electric power generation in the bundle (Figure 1) and in the heated core region corresponding to axial levels 0-1275mm.

The nodalization scheme of PARAMETER test facility for SOCRAT/V2 had 4 radial and 30 axial meshes, most axial meshes are 0.1m long.

The radiative heat transfer is calculated in SOCRAT/V2 taking into account the triangular geometry of the rod bundle (Figure 1).

In flowing of steam-gas mixture through the working channel the velocity profile is settled not at once but some distance after entering the channel from the plenum. Typical
dimension on which the thermohydraulic parameters are determined is called the length of transition section (LTS).

The heat losses in upper part of FA are considered thoroughly in calculations due to detailed modeling of this section and taking into account not only radial but also axial radiative heat fluxes in a system. The adequate heat transfer modelling of this part is very important due to large temperature gradients.

Using for evaluation the relationship \( I_{\text{transition,hydr}} \approx 0.065D_{\text{hydr}} \cdot \text{Re} \) from [14] for LTS (on length of transition section the velocity in the central section differs from the settled value by not more than 1%), valid for round tubes, we’ll obtain:

\[ I_{\text{transition,hydr}} \approx 59D_{\text{hydr}} \approx 0.52 \text{ m.} \]

In the light of the above-mentioned during calculations with the use of SOCRAT code the hydraulic diameter in the input file was taken not constant but depending on the elevation.

In Figure 4 the bundle temperature at 1000 mm elevation (near the upper part of heated zone) versus time for PARAMETER-SF4 is presented. The maximum temperature about 1900K was reached at this axial level. Figures 5-8 show the temporal dependence of temperature for different axial locations: 900, 700, 500 and 300 mm.

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 ZYFB-3 isolation [15] were used in the modelling.

In Figure 9 the overall heat balance for the core (the heated part of the bundle) are presented. The heat flux to shroud due to radiation dominates in comparison to convective heat flux to shroud, which is indicative of importance of adequate radiative transfer calculation for the test.
Figure 4: PARAMETER-SF4: temperature at elevation 1000 mm

Figure 5: PARAMETER-SF4: temperature at elevation 900 mm

Figure 6: PARAMETER-SF4: temperature at elevation 700 mm

Figure 7: PARAMETER-SF4: temperature at elevation 500 mm

Figure 8: PARAMETER-SF4: temperature at elevation 300 mm

Figure 9: PARAMETER-SF4 calculation heat balance:
1 – total electric power,
2 – power transferred by gas,
3 – heat flux to shroud,
4 – chemical power

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Let us summarize the main results of investigation. The basic thermal parameters of experiments PARAMETER-SF4 are adequately reproduced by the code. Integral hydrogen production is in a satisfactory agreement with the experimental value. Because of strong influence of main radiative exchange parameters on thermal response, the adequacy of calculated and experimental data looks optimistic for justification of implemented radiation model [13].

4 AIR INGRESS PHASE FEATURES AND THERMAL HYDRAULICS DURING FLOODING IN PARAMETER-SF4 EXPERIMENT

The air ingress phase (Figure 2, phase 4) was very important phase of SF4 experiment. The thing is that the oxidation of zirconium claddings in the air behaves in different way in comparison to oxidation in the vapor. First, the heat effect of the chemical reaction of oxidation in the air is approximately two times larger than in the vapor. 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 4-8) 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 vapor (PARAMETER-SF1, PARAMETER-SF2 [1] and PARAMETER-SF3 [2]) where the highest temperatures were reached definitely at highest elevation 1250 mm from the bottom of heated region.

The model of oxidation in the air in the code SOCRAT/V2 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 10 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 oxygen starvation was observed in this experiment PARAMETER-SF4.

![Figure 10: PARAMETER-SF4 calculated O₂ mass flow rate at the inlet and the outlet of test section. The oxygen starvation region is indicated](image-url)
Complicated picture of quench during reflood was observed in the experiment as shown in Figure 11. One can note that the temperature at the level 800 and 900 mm (close to top of the core) experienced the non-monotone behaviour: after the rapid decrease initiated by top reflood, the temperature again had begun to rise and finally, some time after the bottom reflood initiation, it reached the saturation point.

Let us consider the features of reflood during the bottom flooding (the experiment SF4 considering in this paper) and the top flooding (the experiment SF3 [2]). The experimental cladding temperatures at different elevations in SF3 is shown in Figure 13. The top mass flow rate in SF3 was 40 g/s very close to bottom mass flow rate 50 g/s per assembly in SF4 experiment.

Comparing Figure 11 and Figure 13 one can see two main differences in flooding behavior during top and bottom reflood. First, the time needed to complete quench of the fuel assembly is more in the case of bottom flooding (approximately 420 seconds against 270 seconds in the case of top flooding). Second, the movement of quench front in the case of bottom flooding is generally more uniform than in the case of top flooding when the non-monotonic quench front displacement occurred (Figure 15). Possible reason for it is the influence of CCFL (counter-current flooding limitation) phenomenon.

The calculated temperature dynamics obtained by SOCRAT code is reproduced in the Figure 12. Figure 14 shows bottom quench front dynamics in the PARAMETER-SF4. The analysis of this figure shows that not all aspects of reflood are adequately described by the code. However, the main parameters of reflood as the temperature decrease rate and the characteristic time of quench, are in a reasonable agreement with the experiment.

The reason for underestimation of quench time in SOCRAT calculations both for top front in SF3 and for bottom front in SF4 may be neglect of the spacer grids influence on thermal hydraulics during the reflood.

Figure 11: PARAMETER-SF4 experimental cladding temperature behavior during bottom reflood

Figure 12: PARAMETER-SF4 calculated cladding temperature behavior during reflood
5 CONCLUSIONS

The Russian computer modelling code SOCRAT/V2 was used for calculations of the integral experiment PARAMETER-SF4, the VVER fuel simulator assembly experiment with air ingress phase and bottom reflood done at high temperature 1900K.

The air ingress phase was important PARAMETER-SF4 test feature, influencing on the test behavior.

The adequate modeling of the thermal hydraulic and chemical phenomena which occurred during the test, including the convective and radiative heat transfer in the bundle, the oxidation during the air ingress phase and the two-phase coolant behavior during the bottom reflood, has allowed to calculate the correct thermal behavior in the test.

The neglect of the spacer grids influence on thermal hydraulics during the reflood may be the reason for underestimation of calculated quench time.
On the whole, the calculated and experimental thermal-hydraulic data are in a good agreement, which shows the adequacy of modeling the complicated thermo-hydraulic behavior including the air ingress phase and the bottom reflood in the PARAMETER-SF4 test.

**NOMENCLATURE**

- $A$: mass flow, kg/s; constant in Arrhenius equation
- ECCS: emergency core cooling system
- FA: fuel assembly
- LOCA: loss-of-coolant accident
- NPP: nuclear power plant
- PWR: pressurized water reactor
- $T$: temperature, K
- $t$: time, s
- VVER: Russian type of pressurized water reactor

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