Thermal Hydraulics Aspects of Top Reflood Experiment
PARAMETER-SF1 Modeling

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ABSTRACT

The PARAMETER-SF1 test conditions simulated a severe LOCA (Loss of Coolant Accident) plant sequence in which the overheated up to 2000°C core would be reflooded from the top in occasion of ECCS recovery. The test was successfully conducted at the NPO “LUTCH”, Podolsk, Russia, in April 15, 2006 and was the first of two experiments to be performed in the frame of the ISTC Project.

The object of investigation was the bundle of VVER-1000 type, which is made up of 19 fuel rod simulators (Zr1%Nb claddings with UO2 pellets inside). The residual heat of bundle was simulated by electrical power supply.

From the time t~11000s of the test it has been established that the behavior of some bundle parameters diverged from anticipated values. It was supposed that the reason of such a behavior is the heat imbalance, appeared possibly due to accidental steam bypass and steam condensation in the steam-noncondensables entry region of the facility. At the transient, the bundle was overheated up to 2000°C. At 14945s, the quench water injection was initiated, the water flow rate was ~ 41 g/s. The bundle quenching was successful.

The analysis of PARAMETER-SF1 experiment showed that the major hydrogen release was connected with reflood phase. Total amount of hydrogen produced in the test is estimated as 90g.

Thermal hydraulics in PARAMETER-SF1 experiment played very important role and its adequate modeling is important for the thermal analysis. The thermal hydraulic and SFD (Severe Fuel Damage) best estimate code SOCRAT-V1 was used for the calculation of PARAMETER-SF1 experiment. The results obtained by the code were compared with experimental data concerning different aspects of thermal hydraulics behavior including the steam condensation in the presence of non-condensable gas, the bypassing and the reflood. The temperature experimental data were found to be in a good agreement with calculated results.

INTRODUCTION

The experiment PARAMETER-SF1 was the first test from the series of two tests within the ISTC 3194 Project. The conditions of PARAMETER-SF1 test modeled the severe accident of LOCA type, during which the overheated up to 2000°C is flooded from the top in the case of the recovery of core accident cooling system (ECCS).

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 [1,2] as a close cooperative effort undertaken by the Russian and foreign organizations to study into the Russian VVER-1000 fuel assemblies behavior at the early stage of the 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 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 top and combined flooding from the top and bottom has not been sufficiently studied as well.

In this connection, the objective of PARAMETER-SF1 test considered in this paper is the experimental investigation of fuel rod VVER-1000 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 flooding from top.

At present, similar experiments are underway in the research center Forschungszentrum Karlsruhe, Germany, under the QUENCH experimental program [3-7] aimed at studying mechanical and physical and chemical behavior of overheated fuel rod cladding with quenching from bottom. These experiments, however, used zirconium dioxide imitator 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 a severe accident.

The aims of the experiment PARAMETER-SF1 were as following:
- The investigation of the thermo-mechanical behavior of VVER-1000 fuel assembly (FA) under the imitating conditions of severe accident, including the stage of low velocity cooling during the top reflood;
The investigation of the oxidation extent of structure components of VVER-1000 FA;
• The investigation of the interaction in materials of VVER-1000 FA (pellets and cladding);
• The hydrogen release in FA under the imitating conditions of severe accident, including the stage of low velocity cooling at the top reflood.

The test was successfully conducted at the NPO “LUTCH”", Podolsk, Moscow region, Russia in April 15, 2006.

From the time $t \sim 11000$s of the test it has been established that the behavior of some bundle parameters diverged from anticipated values. In particular, the increase of electrical power (from 4.6 to 6.4 kW) resulted in higher heat-up of the bundle as to which was expected: the reduction of temperature growth due to heat removal by the steam was not observed. It was supposed that the reason of such a behavior is the heat imbalance, appeared possibly due to accidental steam bypass and steam condensation in the steam-noncondensables entry region of the facility. At the transient, the bundle was overheated up to 2000°C. At 14945s, the quench water injection was initiated, the water flow rate was $\sim 41$ g/s. The bundle quenching was successful.

The Russian best estimate code SOCRAT-V1 was used for the calculation of PARAMETER-SF1 experiment. SOCRAT-V1 code is consisted from two modules: RATEG - thermal hydraulics, SVECHA – severe fuel damage phenomena.

The calculated results obtained using SOCRAT-V1 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-SF1 experiment.

PARAMETER FACILITY DESCRIPTION AND MODELING

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.

The test bundle (Fig. 1, Fig. 2) is 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). 18 fuel rod simulators are heated over a length of 1275 mm, the one unheated fuel rod simulator is located in the centre of the test bundle. Heating is carried out electrically using 4-mm-diameter tantalum heating elements installed in the centre of the rods and surrounded by annular UO$_2$ pellets. The rod cladding is identical to that used in VVERs: Zr1%Nb, 9.13 mm outside diameter, 0.7 mm wall thickness. The test bundle is 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 is filled with pellets of UO$_2$ (bore size 1.2mm internal diameter). In the experiment PARAMETER-SF1 the unheated rod simulator was not filled with pellets. For the heated rods 6mm diameter tungsten heating elements are installed in the centre of the rods and are surrounded by annular UO$_2$ pellets (bore size 4.2mm internal

![Fig. 1. Test rod designation and dimensions of test bundle PARAMETER](image)

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diameter). The tantalum heaters are connected to electrodes made of molybdenum and copper at each end of the heater. The general view of the facility central part with fuel rod simulator bundle is presented in Fig. 2.

![General view of the central part of PARAMETER facility for SF1 test](image)

Fig. 2. General view of the central part of PARAMETER facility for SF1 test (the bypass and condensation zones are shown)

The nodalization scheme of PARAMETER test facility for SOCRAT-V1 thermal hydraulics and severe accident phenomena numerical code is presented in Fig. 3. The nodalization scheme used in the debris module for computation of PARAMETER-SF1 experiment had 4 radial and 30 axial meshes, most axial meshes are 0.1m long.

**PARAMETER-SF1 TEST SCENARIO**

The PARAMETER-SF1 experiment consisted of 4 phases:
First heat-up (preparatory) phase (0÷11000s), the stabilization of steam and Ar mass flow rates ant cladding temperatures (mass flow rates \(A_{H_2O} \approx 3\, g/s\) and \(A_{Ar} = 2\, g/s\), \(T=500^\circ C\)), the heat-up to \(T=1200^\circ C\) in hot region;

• Pre-oxidation phase (11000÷14450s), the cladding temperature \(T=1200^\circ C\) in hot region;

• Transient phase with heat-up to \(\sim2000^\circ C\) (14450÷14845s);

• Top flooding phase (14890÷15400s), water mass flow rate \(A\approx40g/s\).

The water cooling system (water cooling jacket) was turned on at time 10250s. From this time, the external steel containment began to be cooled by the water.

The flooding phase was initiated by water injection to the channel, turning off the argon and steam flow and switching off the electric power in the bundle. The quenching was successful.

The total amount of hydrogen released during PARAMETER-SF1 experiment was 90g.

The post-test appearance of test bundle revealed severe oxidation and embrittlement (fragmentation) of the rod claddings and shroud in the upper part as well as melt formation and relocation.

**THERMAL HYDRAULICS OF PARAMETER-SF1 EXPERIMENT**

**Bypassing**

In the course of the experiment, the H\(_2\)O-Ar mixture leakage to bypass technological channels took place (Fig. 1, Fig. 2). So, the part of gas mass flow after entering the bundle channel went down to lower plenum and then moved to the bypass formed by two channels: the channel between two coaxial steel tubes and the channel placed between external boundary of the isolation and internal boundary of inner steel tube (Fig. 1).
The bypassing resulted in lowering the effective gas mass flow through the bundle. The approximate cross-section areas of these two channels are about 200 mm² (the space between steel tubes) and about 700 mm² (the space between the isolation and inner steel tube). The total cross-section area of bypass is therefore comparable to the bundle channel cross-section area of about 1550 mm² for the initial temperature conditions.

As it was shown in the modeling with SOCRAT-V1, the bypassing together with the condensation effect resulted in lowering the vapor mass flux through the bundle from the initial 3.4 g/s to the value of about 1 g/s and even less. This changed dramatically heat balance in the bundle.

**Condensation in Lower Plenum**

The second important factor, which interfered the projected test parameters, is the steam condensation, which took place in the lower part of the bundle (Fig. 2, Fig. 4). The condensation process was especially intensified after switching on the water cooling system of the containment at time 10250s. The condensation phenomenon in PARAMETER-SF1 test is complicated due to presence of non-condensable gas Ar.

Fig. 4. Schematic representation of condensation phenomenon (the wall is the inner steel tube wall)

The heat flux through the gas-liquid interface in the condensation process (Fig. 4) is the sum of convective and condensation (due to latent condensation heat release) heat fluxes: 

\[ q = q_{\text{conv}} + q_{\text{cond}}. \]

The first constituent in the heat flux dominates in the case of large concentration of non-condensable gas.

The constituents of total heat flux are determined by this way:

\[ q_{\text{conv}} = h_{\text{conv}}(T_b - T_i) , \quad q_{\text{cond}} = h_{\text{cond}}(T_{b,\text{sat}} - T_i) , \quad (1) \]

where \( T_{b,\text{sat}} \) is the saturation temperature corresponding the bulk parameters of the mixture; \( T_b \) the bulk temperature; \( T_i \) the temperature at the gas-liquid interface, \( h_{\text{conv}} \) and \( h_{\text{cond}} \) the convection and the condensation heat transfer coefficients.

Note, that the driving temperature difference for two constituents is different: \((T_b - T_i)\) for convective flux and \((T_{b,\text{sat}} - T_i)\) for condensation flux, which is in accordance with the paper [8].

Take into consideration the reduction factor of condensation heat transfer coefficient in the presence of non-condensable gas. In the paper [9] the following value for this factor was obtained:

\[ k = \frac{h_{\text{noncond}}}{h_{\text{cond}}} = 0.0151215 (X_s)^{0.6000237} , \quad (2) \]

where \( X_s \) is mass fraction of non-condensables.
Let us estimate convective and condensation heat fluxes in the condensation region of the bundle during the PARAMETER-SF1 test.

The convection is governed by Dittus-Boelter dependency:

\[ Nu = \frac{h_{conv} D_{nom}}{\kappa_g} = C_1 \text{Re}^{C_2} \text{Pr}^{C_3}, C_1 = 0.023, C_2 = 0.8, C_3 = 0.4. \]  (3)

where \( Nu \), \( \text{Re} \), \( \text{Pr} \) are Nusselt, Reynolds and Prandtl numbers, \( \kappa_g \) the thermal conductivity, \( D_{nom} \) the hydraulic diameter.

Using the analogy between heat- and mass transfer \([8]\), the total heat transfer coefficient can be written as

\[ h = h_{conv} \left[ 1 + \frac{\mu_g}{\kappa_g} \frac{1}{Sc} \left( \frac{Sc}{Pr} \right) \frac{X_{s,w} - X_{v}}{1 - X_{v}} \left( \frac{r_v + c_p(T_w - T_f)}{(T_b - T_w)} \right) \right]. \]  (4)

where \( Sc \) is the Schmidt number; \( \mu_g \) the gas mixture dynamic viscosity; \( r \) the latent evaporation heat, \( c_p \) the vapor specific heat; \( T_f \) the water film temperature; \( X_{s,w} \) and \( X_{v} \) the steam mass fractions in the bulk and at the condensation interface, respectively.

The condensation mass flux, \( kg/(m^2 s) \) is equal to:

\[ j \approx \frac{h(T_w - T_f)}{r_v + c_p(T_w - T_f)}. \]  (5)

Under the conditions in lower plenum \( p = 3 \times 10^5 \text{ Pa}, T = 750\text{K} \), the density of the mixture \( \rho_g \) is equal approximately to the value of 1.1kg/m\(^3\).

The Prandtl and Schmidt numbers are of the order of unity. Under the pressure and temperature conditions in lower plenum \( p = 3 \times 10^5 \text{ Pa}, T = 750\text{K} \), the parameters of the mixture are equal approximately to the following values:

\( \mu_g \approx 2.5 \times 10^{-5} \text{ Pa} \cdot \text{s}, \kappa_g \approx 3.6 \times 10^{-2} \text{ W/(mK)} \), \( r_v \approx 2.2 \times 10^6 \text{ J/kg} \), \( D_{nom} \approx 8.3 \times 10^{-3} \text{ m} \). At temperature difference \( T_w - T_f \) about 50K the value of condensation heat transfer from Eq. 3,4 is about 400 W/(m\(^2\)K), which corresponds to mass flux of pure condensed steam on unit surface area \( j \approx 2 \times 10^{-3} \text{ kg/(m^2 s)} \). The condensation area is about \( S_{cond} = \pi D_{cond} L_{cond} \approx 7.5 \times 10^{-2} \text{ m}^2 \), \( D_{cond} = 6 \times 10^{-2} \text{ m} \), \( L_{cond} \approx 4 \times 10^{-1} \text{ m} \). Finally, using the parameters and correlations given above and a factor (2), the estimation for vapor mass condensed per unit time is therefore \( J \approx 1+2g/s \).

**PARAMETER-SF1 CALCULATION RESULTS**

In Fig. 5a the total electric power history is presented. Fig. 5b, 6 show the temporal dependence of temperature for different axial locations: 400, 800 and 1250 mm (near the upper part of heated zone).

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. The thermal conductivity data for ZYFB-3 isolation (www.zircarzirconia.com) were used in the modelling.

Calculated and experimental hydrogen production rate is presented in Fig. 7a. There were some difficulties in experimental determination of this parameter. Experimental value for \( H_2 \) production is estimated as 90g with most hydrogen amount released during the quench. This value is only in a limited consistency with computational results. The calculated amount of hydrogen produced before the reflood is in a reasonable agreement with the experiment, but SOCRAT-V1 underestimated the hydrogen released during the reflood. The total calculated amount of \( H_2 \) was 37g. The possible reason for such underestimation is that oxidation in debris and molten pool regions has some peculiarities in comparison to oxidation in solid. The oxidation of metallic melts is not described by application of usual kinetic correlations. Melt oxidation together with melt relocation, dispersion and fragmentation is obviously one of the most important quench phenomena.

Fig. 7b presents the temporal behaviour of water volume condensed in lower part of the bundle. It is seen from the figure that the condensation rate during the test after turning on the water jacket is approximately \( 1+1.5 \text{ g/s} \) which is in a good agreement with the estimations made in previous paragraph as well as with the experimental data.
CONCLUSIONS

The bypassing and steam condensation were important PARAMETER-SF1 test features, influencing the test behavior.

The best estimate code SOCRAT-V1 was used for modeling of PARAMETER-SF1 test. The adequate modeling of the thermal hydraulic phenomena occurred during the test including the bypassing and steam condensation in lower plenum, allowed to calculate the correct thermal behavior in the test.

The calculated results obtained using SOCRAT-V1 are compared to experimental data. The calculated and experimental thermal-hydraulic data are in a good agreement, which is indicative of the adequacy of modeling the complicated thermo-hydraulic behavior in the PARAMETER-SF1 test.

Fig. 5a,b. PARAMETER-SF1. Total electric power history (left) and cladding temperature at elevation 400 mm:
1 – experiment (T224), 2 – SOCRAT-V1

Fig. 6a,b. PARAMETER-SF1: cladding temperature at elevation 800 mm (left) and 1250mm (right)
1 – experiment (T258 – 800mm, T2212.5-1250mm), 2 – SOCRAT-V1
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