STUDY OF HEAT AND MASS TRANSFER PHENOMENA IN FUEL ASSEMBLY MODELS UNDER ACCIDENT CONDITIONS

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Abstract

The majority of the material in support of the thermal - hydraulic safety of WWER core was obtained on single - assembly models containing a relatively small number of elements - heater rods. Upgrading the requirements to the reactor safety leads to the necessity for studying phenomena in channels representing the cross - sectional core dimensions and non - uniform radial power generation. Under such conditions, the contribution of natural convection can be significant in some core zones, including the occurrence of reverse flows and interchannel instability. These phenomena can have an important influence on heat transfer processes. Such influence is especially drastic under accident conditions associated with ceasing the forced circulation over the circuit.

A number of urgent reactor safety problems at low operating parameters is related with the computer code verification and certification. One of the important trends in the reactor safety research is concerned with the rod bundle reflooding and verificational calculations of this phenomenon. To assess the water cooled reactor safety, the best fit computer codes are employed, which make it possible to simulate accident and transient operating conditions in a reactor installation. One of the most widely known computer codes is the RELAP5 / MOD3 Code. The paper presents the comparison of the results calculated using this computer code with the test data on 4 - rod bundle quenching, which were obtained at the SSCRF - IPPE.

Recently, the investigations on the steam - zirconium reaction kinetics have been performed at the SSCRF - IPPE and are being presently performed for the purpose of developing new and verifying available computer codes.

1. FLUID DYNAMICS AND HEAT TRANSFER IN MULTIASSEMBLY CORE MODELS

The tests were conducted on an integral WWER - 1000 primary circuit model including a 5 - assembly model shown in Fig.1. Each of the assemblies encumbrises 14 indirectly heated 700 mm long heater rods 9.1 mm in diameter, i.e. the scale of height is 1 : 5. The heater rods were arranged in triangular array with a pitch of 12.75 mm. All five assemblies were encased into a test vessel with internal dimensions of $38 \times 340 \text{ mm}^2$. The front and the back walls of the test vessel were manufactured of heat - resistant glass.

The partition presence or absence between the fuel assemblies makes it possible to simulate fuel assemblies with and without shrouds (WWER - 440 and WWER - 1000, respectively). In the area of the lower plenum, all fuel assemblies were in communication with each other, as it is in the case of a real reactor set - up.
Fig. 1. Schematic Diagram of Test Section and Core Model Cross-section.
a) 1-5-assembly reactor model; 2-heat exchanger; 3- cold line; 4- hot line; 5-fuel assemblies; 6-20-control valves, providing coolant supply and discharge from circuit; 21-pump; 22,23-pump cut-off valves; 24-cooling water flow rate control valves; 25-measuring tube.
b) 1-test vessel; 2-front and back walls of test vessel; 3-seal; 4-non-heated rod; 5-heated rod; 6-baffle.
One of the limiting hydrodynamic phenomena occurring in WWER-type reactors under accident conditions with a water filled core is reflooding. It is characteristic of the situation under such conditions that there is no coolant flow rate at the channel inlet, with the upward steam flow and downward water flow within the channel itself. As the power supply to the channel increases, the flow rate of steam being generated goes up and conditions are reached, when water entering the channel from above is not capable to compensate its evaporation. From this moment, the channel dryout starts, which finally leads to the fuel element overheating.

The main goal of such researches is to obtain the test data on heat fluxes and cross-sections where the heater rod overheating origins under the conditions of nonuniform cross-sectional power generation.

A few of results obtained for the symmetric power generation profile and variable power of central fuel subassembly 3 are demonstrated in Fig.2. It is evident that in the presence of nonheated zone in the center of the model, the $N_{cr}$ value is minimal (profile 1 in Fig.2). As the power of central fuel subassembly 3 (profiles 2, 3 and 4) increases, the net power increases too. Fig.2 shows also the cross-sectional location of the CHF occurrence and the distance from the upper point of heating ($z_{cr}$) for each profile. As evident, the actual CHF location substantially varies with the profile changing. This fact may be explained by the variation in the natural convection intensity in this region as well as by changing its direction from upward to downward one as the central rod bundle power increases.

It is established also that the CHF location changes at the transition from shroud-covered fuel assemblies to those without shrouds: in fuel assembly models without shrouds, the CHF occurs in the area of heater rods being on the boundary of stable upward and downward convective flows; the CHF occurring not necessarily in the hottest fuel subassembly.

![Fig.2. Critical Power of Core Model with Shroud-Covered Fuel Assemblies as a Function of Central Fuel Assembly Power. $z_{cr}$-distance from CHF initiation to the top of heating. Arrows in Profiles 1-4 indicate the direction of convective flows.](image-url)
The reflood test data obtained were compared with the results calculated using the correlations by Wallis, Tien, TsKTI and Bezrodny. In all cases, the TsKTI and Bezrodny correlations describe the data for multichannel systems best of all. The question is certainly about only a rough agreement of results at a power generation being close to the uniform cross-sectional one. In transition to rod bundles, the Wallis reflooding equation needs be substantially refined as it was done, for example, with the reflooding model for the RELAP5/MOD3 Computer Program. The results presented in Fig.2 as well as the other available test data on reflooding show a rather strong influence of the nonuniform cross-sectional power generation on the critical power value and the CHF occurrence point. Also it was established that these phenomena are significantly affected by fuel assembly shrouds. The fact of importance is actually the rod overheating in the area of downward flows, where maximum heat fluxes are not observed.

In a number of accidents accompanied by coolant leakage in the primary circuit, conditions under which the coolant level does not reach the top of the zone are possible. Such conditions can occur due to the absence or insufficient rate of feedwater supply, negative effect of hydraulic loop seal, etc.

In modeling situations like the abovementioned one, the following parameters were recorded along with the heater rod temperature: the power of each of 5 fuel assemblies, the mass and physical levels of coolant in each fuel assembly, the temperatures of the superheated steam above the upper edge of heater rods.

Based on the test results, the heat transfer coefficients, $a_{exp}$, were determined. The obtained results were compared with a few of correlations for laminar conditions in the uncovered zone.

![Fig.3. Heater Rod Wall Temperature in the Uncovered Zone as a Function of “Hot” Fuel Assembly Heat Flux.](image)

The height of the mass level is 400 mm; $q_{max}/q_{min}=1.15$: all fuel assemblies being heated. 1-3 - fuel assemblies without shrouds: 4-6 - with shrouds. The distance from the heating beginning: 1.4-630 mm: 2.5-570 mm: 3.6-440 mm.
In Fig.3, the comparison of some results on the heater rod temperature behavior in rod bundle systems simulating the fuel assemblies with and without shrouds is given. In both cases, the presented temperature values refer to the hottest fuel assemblies at a cross-sectional nonuniformity coefficient, $q_{\text{max}}/q$, of 1.15. To judge by the results obtained in Fig.3, it can be noted that in the fuel assembly with shroud-covered assemblies more lower temperatures are observed in the uncovered region. The partition absence might be expected at first glance to lead to the occurrence of intensive cross-sectional flows and temperature level flattening at the nonuniform cross-sectional power generation. However, an opposite situation is actually observed: the temperature values in rod bundle models with shrouds are by 50–150 °C lower under otherwise equal conditions.

The following is probably responsible for such behavior. First, in fuel assemblies splitted by partitions, the physical levels of coolant are very different at the nonuniform cross-sectional power generation, whereas in fuel assemblies without partitions, a certain flattening of the physical levels of coolant takes place. The more higher values of the physical levels in hot and partitioned fuel assemblies result in an increased amount of steam generation and augmented droplet entrainment and therefore can be the main cause of the heat transfer improvement in such fuel assemblies.

The second fact which seems to be responsible for more higher rod wall temperatures in the shroudless fuel assembly model is associated with the steam flow in the uncovered region. Not only water but also steam can leave the hotter fuel assemblies (or subchannels) due to higher pressure drops in the uncovered region. The reduction of steam flow rates over a hot fuel assembly in the uncovered region will favour its heating-up and increasing in heater rod wall temperature.

Thus, the information as obtained for heat transfer in uncovered region permits one to postulate that in the core consisting of fuel assemblies housed in shrouds, the fuel element overheating is lower. To judge by the fact that all domestic data on heat transfer in the uncovered region were produced on single fuel assemblies, these results may be expected to lead to more favourable results than they are in reality as applied to WWER-1000 cores (consisting of shroudless fuel assemblies) without due account of cross-sectional flows.

2. COMPARISON OF RELAP5/MOD3 CALCULATIONS WITH TEST DATA ON 4-ROD BUNDLE REFLOODING

The tests were conducted on a bundle of four heater rods. The flow area of the channel was the 32 by 32 mm square. The heater rods represented stainless steel 3008 mm long tubes with cross-sectional dimensions 13 x 1 mm, which were heated by direct current. Up to 10 thermocouples with 0.3 mm thermoelectrodes were installed on the internal surface of each of the four tubes by contact welding. The heater rods were centered in the channel by means of 15 mm high spacer grids. All in all 12 grids were arranged with a pitch of 245 mm: the first spacer grid being 49 mm distant from the heating start. The test facility was equipped with a IVK20- and PC-based data acquisition system making it possible to record up to 100 channels with a frequency to 4 kHz.

The test procedure was as follows. Using the pressurizer, a pressure about 0.25 MPa was maintained in the test section outlet with a prescribed inlet water flow rate and temperature. After the circulation which proceeded during a few minutes and heating the equipment, the pumps were turned off and the test channel was cut from the circuit and drained. Then, the channel was jointed to the circuit; using gas from cylinder, the preset pressure was restored and electric power was
1. Operating Parameters Versus Time at Reflooding.

   Test run 10. 1-power (N/25) kW; 2-pressure (P/4) bar; 3-outlet temperature (T/250)°C; 4-flow rate (G/200) kg/h; 5-inlet temperature (T/150)°C; 6-pressure drop on channel (ΔP/0.5) bar.

2. Test Section Surface Temperature Versus Time.

   - test (4.35.31.17-distance from the heating beginning is 1568, 2138, 2488 and 2838 mm, respectively);
   - calculation (1.2.3.4-distance from the heating beginning is 1075, 1505, 2150 and 2580, respectively).
Fig. 6. Schematic of Test Apparatus for Study of Hydrogen Release Kinetics during Steam-Zirconium Reaction on a Heater Rod.

1-heater rod; 2-transparent test vessel; 3-pillar; 4-flange; 5-thermocouples; 6-cooler; 7-hydrogen collector; 8-differential pressure gauge; 9-collecting tank; 10-water feed tank.

supplied to heater rods. After one of the thermocouples had reached the preset temperature (about 970 K), the data acquisition system was automatically put on and the coolant circulation was resumed. From this moment, the reflooding of the channel started. The quenching front went up and the channel inlet pressure rose due to the steam generation a little. With going over the preset level of pressure, steam was partially released into the atmosphere.

Using a specially developed program, the test data were averaged by means of base splines; and the heat flux, the heat transfer surface temperature and heat transfer coefficients in the unwetted region were determined according to the special programs.

The calculations were performed using the RELAP5/MOD3/V5M5 Code [2]. The nodalization scheme of the test section consists of 7 components including the inlet chamber, pump, fuel
assembly, a simple fuel assembly junction to the outlet chamber, the outlet chamber, setting part and fuel assembly shroud.

The following parameters were initially preset in calculations: thermal power, pressure $P_0$, and temperature $T_o$ (in the fuel assembly), coolant flow rate at the inlet (being equal to zero), coolant flow rate at the outlet (being equal to zero), pressure, $P_o = 2.7$ bar, and steam quality, $X_o = 1$ (in outlet chamber).

If only in one of the sections the wall temperature, $T$, reached 973 K, the inlet flow rate varied stepwise from zero to the preset value (0.05 m/c). At this moment, the bundle was considered uncovered and from this moment, its reflooding started. The variation of main parameters in the course of the test is shown in Fig.4.

The comparison of calculated results with experimental data on heater rod surface temperature variation is demonstrated in Fig.5. As evident from the figure, the calculation substantially underestimates the time of the fuel assembly quenching in first 11 sections: at the same time, complete cooling does not occur in outlet sections. An attempt was made to improve the agreement of the results by taking account of the effect of the insulation material restricting the channel flow area. This made it possible to eliminate the quenching front blockage and reduce the peak wall temperatures to experimental values. However, the agreement in quenching time was actually not improved.

In the present authors opinion, the main causes of the discrepancy are as follows. To evaluate heat transfer in the uncovered zone, the code version under consideration employees the modified Brombey equation for a horizontal pipe at natural convection, where the tube diameter is substituted by the length of a quick-rising wave. Besides, to average the heat transfer coefficient within the wide range of two-phase flow qualities, a weight factor varying from 1 to zero is used. However, based on the available experimental results [3], the quality varying from zero to 1 actually does not exert any effect on the heat transfer coefficient. It is worth noting also that the data obtained by the present authors are in fair agreement with the results calculated by the Sacurai and Dhir formulae having the same form as the Bromley equation. Thus, the expression used in the Code for evaluating the heat transfer under the conditions of saturated film boiling (mod 8) and subcooled boiling (mod 7) is incorrect both from the viewpoint of its application for absolutely different conditions and from the viewpoint of quality accounting.

3. EXPERIMENTAL AND CALCULATIONAL STUDIES OF THERMAL-HYDRAULIC AND PHYSICAL-CHEMICAL PHENOMENA COUPLING IN UNCOVERED WWER CORE

The WWER core contains about 20 t of zirconium and zirconium melt capable of producing about 900 kg of hydrogen in the assumption of full oxidation. In principle, other core materials (like steel, Inconel, etc.) can be the additional source for hydrogen. Nevertheless, zirconium is the basic source of hydrogen in accident situations; the rate of its liberation being dependent on many conjugated phenomena.

The destruction of the oxide film on the zirconium surface is the factor which intensifies both the hydrogen release and the destruction of the shell itself. The free circulation in the emergency core also has an important bearing on the time of the hydrogen generation and its amount. A great influence on the hydrogen release rate is affected by the steam flow rate to the zirconium
surface. Thus the ingress of an additional amount of water into the zone, for instance, due to the personnel action may result in an augmentation of hydrogen release and thus, from this viewpoint, in worsening the situation in the core.

The studies of the steam - zirconium reaction kinetics were primarily restricted by integral characteristics and aimed at the development of new computer codes and verification of the available ones so far. However, in view of the sophistication and versatility of phenomena occurring in the accident core, the existing codes not always predict them adequately.

The differences are actually due to both the conditions of experimental simulation and discrepancy in the assessment of the contribution of separate components of phenomena occurring under the accident conditions in the core.

Thus, test data are urgently needed to refine the thermal - hydraulic models describing the hydrogen generation processes in the WWER core under the accident conditions.

The main goal of the present paper is to provide test data and develop computer codes giving account of the steam - zirconium reaction proceeding peculiarities and the effect of the two - phase mixture thermal - hydraulics on it during the WWER core uncovery.

The tests are expected to be conducted both using a single rod ( a heater rod with internal heating ) and a 3 and 7 rod bundle of zirconium tubes under the rod bundle uncovery conditions at the atmospheric pressure. The initial conditions are assumed to be as follows. The rod bundle is completely inserted into water ( at a temperature of 20 °C ). The electric power is supplied to heaters installed inside rods. In the process of boiling, the water level goes down, the heater rods become uncovered and start to heat - up. At a certain temperature, the steam - zirconium reaction initiates. In the course of the experiment are recorded: the rod bundle temperature behavior, the rod bundle outlet steam temperature, the steam flow rate, the average void fraction, the local quality ( the phase component measuring transmitter ), the flow rate of steam generated, etc. If necessary, a water feed system is provided for the rod bundle.

These tests are planned to be extended by subsequent metallographic examination of heater rods and inspection of disturbances in the heater rod bundle ( or separate heater rod ) geometry. The schematic diagram of the test facility is presented in Fig.6. The basic components of the test facility are:

• a rod ( heater rod ) or rod bundle ( 3 ) with a 9.1 mm - 0.d., 1 m long zirconium shroud.

The internal space of rods where a tungsten - niobium heater is positioned is filled with argon at a pressure of 0.7 * 1.9 MPa. The rods are arranged in bundle in correspondence with the geometric parameters of WWER core. The temperature range of the external rod surface is 500 – 1400 °C;

• the rod bundle vessel ( 1 ) is a pipe manufactured of quartz glass ( for the visualization of the rod bundle uncovery phenomenon ). If necessary, a stainless steel pipe may be used;

• a cooler ( 6 );

• a calibrated vessel ( 5 ) for hydrogen collection;

• a tank ( 7 ) for collection of water forced out from the rod bundle;

• a tank ( 9 ) providing the test section with water feed.

If need be, the test section can be joined to the facility which makes it possible to deliver the superheated steam to the rod bundle and provide the hot rod bundle water reflooding, etc.

The main distinction of this paper from similar ones known from different literature sources is the wide range of phenomena under consideration, from the kinetics of the steam - zirconium reaction and the rod bundle thermal - hydraulics effect on it to the rod bundle configuration change being due to the rod deformation.
1. The test data on the multiassembly core models evidence that the non-uniform asymmetrical power generation has a strong influence on the critical heat flux at reflooding. In a model consisting of fuel assemblies without shrouds, the CHF is primarily observed to occur on the boundary of stable upward and downward convective flows; whereas in a model consisting of shroud-covered fuel assemblies, the CHF may occur in low power generation regions where downward flows are predominant.

2. It is established that at the nonuniform cross-sectional power generation, the core model consisting of fuel assemblies with shrouds provides more lower overheating of hot heater rods in the uncovered region as compared to that consisting of fuel assemblies without shrouds. This fact can be explained by a higher physical level of coolant in the first case in hottest fuel assemblies. As at reflooding, higher temperatures are observed in components indicating a lower level of power generation. This fact should be born in mind in planning studies on the fuel-coolant interaction.

3. The heater rod bundle quenching phenomenon has been investigated and the calculations of the quenching phenomenon have been performed using the RELAP5/MOD3/V5M5 Code. A rather large disagreement between the test data and calculational results in rod bundle quenching time results from the fact that the thermal-hydraulic part of this code version does not fit the phenomenon under consideration adequately.

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