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Research Article

Outcomes of the "steady-state crisis" experiment in the MIR reactor channel*

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Abstract

To license nuclear fuel for nuclear power plants, data on the behavior of fuel elements (FE) under design-basis accidents are required. These data are obtained during tests of fuel assemblies (FA) and single fuel elements in research reactor channels followed by post-test studies in protective chambers.

A reactivity-initiated accident (RIA) with an unauthorized release of CPS rods from the reactor core leads to a pulsed channel power increase. This accident can proceed according to two scenarios: without a critical heat flux (CHF) on the fuel element jacket at the final stage and with a dry heat flux. To date, a series of experiments have been carried out according to the first scenario in the MIR reactor channel and the corresponding data on the behavior of fuel elements have been obtained. An urgent task for today is to prepare and conduct reactor experiments according to the second scenario.

The main experimental parameter that determines the behavior and final state of the studied fuel elements is their temperature. No experimental data were found on the critical heat flux for the rod bundles in the low coolant mass flow rate region (experiments in the MIR reactor channel can be conducted in the range of 200–250 kg/(m^{2}s)). The available data are in the extrapolation range.

The "steady-state crisis" experiment was conducted to obtain data on the critical heat flux value within the specified coolant mass flow rate range in the MIR reactor channel. The test object was a jacket fuel assembly composed of three shortened VVER-1000 fuel rods with a length of 1230 mm (the fuel part length = 1000 mm) installed in a triangular grid at a pitch of 12.75 mm, which is a cell of the VVER-1000 core. This assembly configuration is used for in-pile tests to study the behavior of fuel elements under emergency conditions.

The in-pile testing results are presented. The paper shows the possibility of detecting the start and development of a dry heat flux based on the readings of thermocouples located inside the FE kernel. As a result, the directly measured test parameters were used to determine the critical heat flux value.

Keywords

MIR reactor, fuel element, experimental fuel assembly (EFA), critical heat flux (CHF), RIA (reactivity-initiated accident), thermocouple (TC), temperature, coolant flow rate.

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Introduction

Data on the behavior of fuel elements (FE) during design-basis accidents are necessary for licensing VVER fuel for NPPs. To obtain these data, single fuel elements and fuel assemblies (FA) are tested in the channels of research reactors under specified and controlled conditions, after which their properties are studied in protective chambers. At the SSC RIAR, these tests are mainly conducted in the MIR reactor (Izhutov et al. 2015).

Up to the present, experiments have been carried out in the reactor channels to study the behavior of VVER-1000 fuel elements at ramp and step-like power increases, cyclic power variations (Alekseev et al. 2007), and in design-basis loss-of-coolant accidents (Alekseev et al. 2012, Goryachev et al. 2004). In addition, a series of reactor experiments was intended to determine the yield of fission products from fuel elements with artificially applied defects on their jackets (Burukin et al. 2009).

A reactivity-initiated accident (RIA) at a VVER-1000 reactor belongs to the design-basis category: it is associated with the release of the CPS rods from the reactor core, which leads to a pulsed channel power increase. It can proceed according to two scenarios: without a critical heat flux on the fuel element jacket at the final stage and with a dry heat flux. The most unfavorable is the temperature scenario with a CHF, when, according to predictive calculations, the fuel element jacket temperature rises to 700–800 °C.

Experiments in the channel of the MIR reactor were performed according to the first scenario, i.e., cooling fuel elements in crisis-free mode (Alekseev 2006, 2006a, 2008, Nechayeva et al. 2003, 2004). However, to date, there have been no reactor experiments with a pulsed power increase, in which, at the maximum channel power, a dry heat flux occurred. Performing such experiments is currently an urgent task.

The main parameter that determines the behavior and final state of the studied fuel elements is their temperature which must be predicted with the highest possible accuracy even at the experiment preparation stage. If the fuel element temperature dynamics in the experiment is known, it is possible to determine the reactor operation algorithm.

To calculate the temperature of fuel elements tested according to the scenario with a critical heat flux on the jacket, it is necessary to know the value of the critical heat flux density, Q_c.

According to the results of preliminary calculations, it is possible to conduct a RIA experiment with a CHF on a fuel-element jacket in the MIR reactor channel at a coolant mass flow rate (ρω) from 200 to 250 kg/(m^{2}s). This is explained by the small length of the fuel column of refabricated fuel elements (200 mm); therefore, it is impossible to achieve the required coolant heating at high flow rates. In this computational domain, there are no domestic data on Q_c.

In (Bobkov et al. 2011), a method for calculating Q_c using a table for bundles is described: this region was not marked as experimentally justified and the data were obtained by extrapolation. To date, several papers have been published, the parameters of which do not correspond to the above range of changes in the coolant mass flow rate. The data are presented in Table 1.

Table 1. Parameters for studying critical heat flux density.

Source	Pressure, MPa	ρω, kg/(m^{2}s))	Organization, note
(Logvinov et al. 2004)	7.45–16.7	700–4000	OKB Gidropress
(Krylov et al. 1995)	0.2–3	700–4000	OKB Gidropress
(Shchekoldin et al. 1998)	7.9–17	999–3684	OKB Gidropress
(Sergeev 1998)	0.1–9 12–16.8	50–2400 1000–3680	SSC IPPE, only comparison with calculations
(Lozhkin et al. 1998)	15.7	2000–3600	SSC IPPE
(Bobkov and Smogalyov 2001)	2–20	500–4000	SSC IPPE
(Bolshakov et al. 2009, 2009a)	6.9–16.4	819–4400	RRC Kurchatov Institute, Critical facility

Foreign data on the pipe are combined in a skeleton table (Groeneveld et al. 2007) (LUT-2006); for bundles, it is proposed to use Q_c from this table, multiplied by the corresponding coefficients. The region ρω < 500 kg/(m^{2}s) at a pressure of 16 MPa is in the “extrapolation” zone.

When determining Q_c for a three-element assembly in the specified range of coolant mass flow rates (this assembly is typical for experiments), a “steady-state crisis” experiment was prepared and carried out in the MIR reactor channel (Dreganov et al. 2014, 2015). The critical heat flux density is determined by direct measurements of the experimental parameters, i.e., coolant flow rate, coolant temperature and temperature in the fuel pellet center. The moment when a critical heat flux occurred was recorded by a ramp temperature increase in the fuel pellet, in the zone of the alleged crisis. For earlier detecting the occurrence of a critical heat flux in real time, the rates (derivatives) of temperature changes in the fuel pellet center were determined and recorded.

Experimental device design

Figure 1 shows the device for the “steady-state crisis” experiment, in which only part of the coolant total flow rate is passed through the channel to cool the fuel elements (the total flow rate is divided into two parallel flows). The device was designed to take into account the fact that, in the existing process loop of the loop-type installation, the minimum coolant flow rate through the channel should not be less than 2.5 t/h (reliably measured by control instrumentation), which significantly exceeds the calculated mass flow rate value; therefore, only part of the total flow is directed to cool the fuel elements.

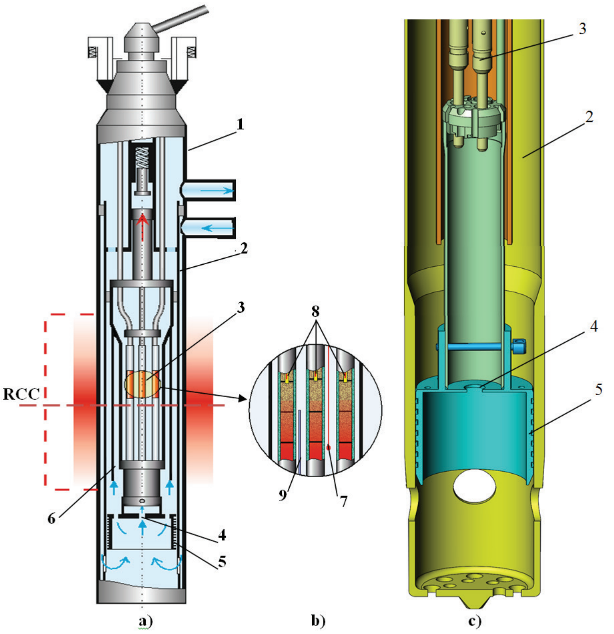

Figure 1. Experimental device: a) overall view; b) cross section in the reactor core center (RCC); c) coolant flow separation node; 1. channel shroud; 2. coolant flow separator; 3. fuel elements; 4. hole; 5. labyrinth seal; 6. cavity with a stagnant aqueous medium; 7. thermocouple (TC) in the coolant; 8. TC (T_1, T_2, T_3 respectively) in the fuel kernel center; 9. direct charge detector (DCD).

The device contains the following main components:

- a three-element assembly with shortened VVER-1000 fuel elements;
- a thermal barrier to reduce heat loss from the volume with the fuel elements;
- an extension shaft for installing the fuel assembly in the reactor core;
- a sealed terminal assembly for "sensor – secondary device" communication cables;
- a node (head) for sealing the experimental device in the channel.

The fuel assembly contains three VVER-1000 fuel elements with an active part length of 1000 mm and a total length of 1230 mm (without TC), two spacer grids (SG), one support grid (lower SG), and a TC to control the coolant temperature and the temperature in the fuel pellet center (650 mm away from the fuel column bottom). The first SG is installed in the cross-section of the upper fuel column boundary and the second one is located at a distance of 300 mm away from the support grid to eliminate disturbances in the upward coolant flow, in the expected CHF region. The spacer grids form a framework by means of three fastening tubes. These tubes at the same time are flow displacers intended to align the coolant flow rate field along the assembly cross section. The fuel assembly is placed in a cylindrical jacket (ID = 32 mm). The pipe is a structural component of the thermal barrier intended to reduce heat loss from the volume with the fuel elements. The thermal barrier is made in the form of two coaxial pipes, between which there is a gap with the aqueous medium with minimal leakage. The outer pipe, which is part of the thermal barrier, divides the total coolant flow into two parallel flows, one of which (about 0.4–0.5 t/h) cools the fuel elements of the experimental assembly. Both coolant flows are run in one above the fuel elements. A throttle device is installed at the bottom of the experimental device (Fig. 1c). This device is a thin plate with a central hole, the diameter of which (5 mm) was determined by calculations.

To minimize the coolant leakage through the gap between the flow separator and the throttle body, a labyrinth seal is provided. Most of the coolant flow goes through holes in the peripheral part of the plate into the cavity, which forms a thermal barrier.

The framework with the fuel elements is mounted on the central extension shaft, which is rigidly connected to the head of the experimental device by a weld seam. To output the "sensor – secondary device" communication lines, a special sealing unit is provided in the device design.

Equipping the EFA with parameter control sensors

The following measuring sensors are installed in the EFA:

- W-Re5/W-Re20 thermocouples: in each FE pellet center;
- Cr/Al thermocouples: in the coolant at the assembly inlet (1 pc. in the bundle center);
- Cr/Al thermocouples: in the coolant at the assembly outlet (2 pcs., their working junctions are spaced along the channel section); and
- Cr/Al thermocouples: in the coolant at a distance of 600 mm away from the support grid (the coordinate of the EFA cross section) where the CHF occurrence on the fuel element jacket is predicted (1 pc.).

The TC working junctions in the center of the fuel kernel and in the coolant are located in one section.

To measure the temperature in the fuel pellet center, we used W-Re5 /W-Re 20 (tungsten-rhenium (95% W + 5% Re)/tungsten-rhenium (80% W + 20% Re)) thermocouples – the 12X18H10T (12Cr18Ni10Ti) steel jacket (D = 2.0 mm) and beryllium oxide thermoelectrode insulation; the TC working junction (D = 0.2 mm) and the jacket are separate (not connected to each other).

To measure the coolant temperature, we used cable Cr/Al (chromel/alumel) thermocouples with a 12X18H10T (12Cr18Ni10Ti) steel sheath (D = 1.5 mm), magnesium oxide insulation, and a joint working junction of thermoelectrodes (D = 0.27 mm).

Before the experiments, all the thermocouples were certified according to GOST R 8.5852001 and the second tolerance class. During the certification, thermoelectric heterogeneity was taken into account as well as the connection of the temperature sensor with the secondary device (signal transducer). The temperature measurement error was 1.5%.

The pressure in the cooling circuit was measured by a Sapphire-22EMA-DI sensor (model 2170): the error is 1.5%.

The error in measuring the total coolant flow through the channel was 1.5% (a restrictive flow orifice device complete with a pressure transducer-converter – Sapphire-22EMA-DD (model 2170)).

All the TCs were connected to a high-speed system for collecting and recording parameters with a frequency of up to 100 Hz. The temperature was recorded in digital files.

In addition to the readings of all the TCs, the assembly power was recorded in real time.

Experimental parameters and algorithm

To complete the assembly, shortened VVER-1000 type fuel elements with a reduced uranium-235 concentration (2.4%) were used. In this assembly, the results of the neutron-physical calculation carried out according to the MCU-PTR program (Alekseev et al. 2010) show that the EFA cross-sectional power peaking factor is equal to 1.01–1.02, which creates the conditions for a CHF occurrence on all the three elements almost simultaneously.

The axial the reactor core power distribution and the coordinate of the TC installed in the center of the fuel element kernel are shown in Fig. 2.

The TC working junction is installed in the cross section, where a CHF occurrence is predicted.

The initial cooling circuit parameters:

- pressure at the channel outlet – 16.1 MPa;
- coolant flow rate through the channel – 2.6 t/h (0.722 kg/s);
- coolant temperature of at the assembly inlet – 280 °C;

The calculation according to the MUZA code (Alekseev 2013) of the coolant upward flow rate distribution over the two parallel flows (through the fuel assembly and the external bypass flow) gave the following results:

- flow rate through the fuel assembly – 0.135 kg/s (mass rate = 200.4 kg/m2s);
- flow rate through bypass – 0.587 kg/s.

The calculation took into account the hydraulic resistance of the flow contraction/expansion and friction resistance.

The limiting values of the flow rate through the EFA for the "steady-state crisis" experiment were estimated with simultaneous deviations of several parameters leading to the flow rate minimum and maximum values (flow rate through the channel by ± 2%, diameters in the labyrinth seal – manufacturing tolerances (60 – -134 μm), hole diameters (0 – +30 μm)). The errors in the formulas for calculating the hydraulic resistance are ± 2%. The minimum (0.122 kg/s) and maximum (0.137 kg/s) flow rates through the EFA were determined within possible variations in the device geometry.

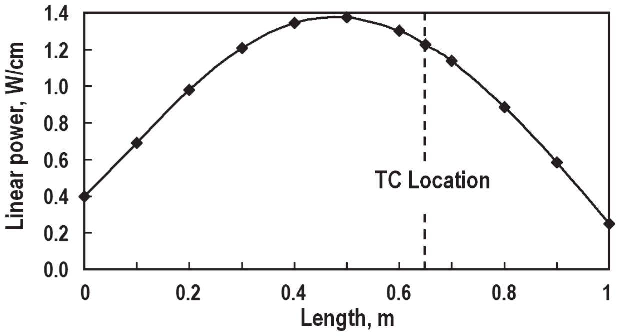

Figure 2. Axial active part linear power distribution.

The assembly power was calculated by the heat balance method, taking into account the heat loss power in the downcomer region and the radiation power density in the shroud pipe. To determine the assembly power, the errors of heating (0.2 °C), water heat capacity (0.5%), and radiation power capacity (± 770 W) were used.

The coolant flow rate through the assembly is calculated by several iterations, since there is a dependence of the flow rate on the assembly power (the coolant temperature).

In order to obtain a critical heat flux on the fuel elements in the assembly, the power capacity was increased in steps; the time between two steps was determined by the time when a steady-state thermal condition was established on the fuel assembly.

Outcomes

The measured values of the coolant temperature and the temperature in the center of the fuel pellet located at a distance of 650 mm away from the lower butt end at the final stage of heating the fuel assembly are shown in Figs 3, 4.

From the primary results of measuring the experimental parameters (Fig. 4), it can be seen that approximately at the 180th second of the final stage of heating the fuel assembly, a ramp temperature increase occurred in the kernel of fuel elements with TC T_1 and fuel elements with TC T_3, which is associated with the critical heat flux.

A critical heat flux did not occur on the fuel element with TC T2, which is explained by the uneven coolant flow rate distribution over the assembly cross section and is confirmed by the calculation results.

At the time when the critical heat flux occurred, the fuel assembly power was 70.5 kW, and the average linear power (LP) of fuel elements was 235.1 W/cm. The LP value for each fuel element was determined by the assembly cross-sectional power peaking factor.

The maximum temperature jump in the CHF took place in the fuel rod with TC T_1, which disabled it.

The critical heat flux value at a height of 650 mm (from the lower butt end of the fuel column) of the TC working

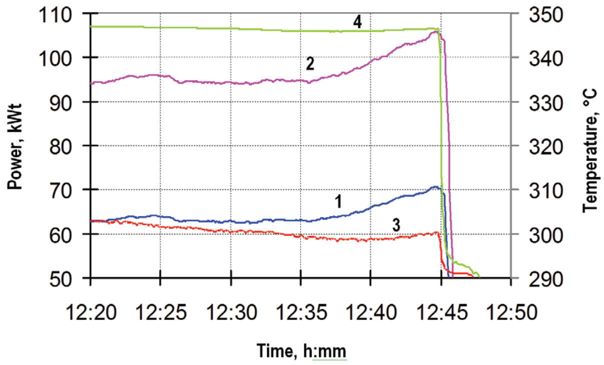

Figure 3. The dynamics of the EFA/experimental channel power and the coolant temperature: 1. EFA power; 2. experimental channel power; coolant temperature: 3. at the EFA inlet; 4. in the EFA inside; 5. at the EFA outlet.

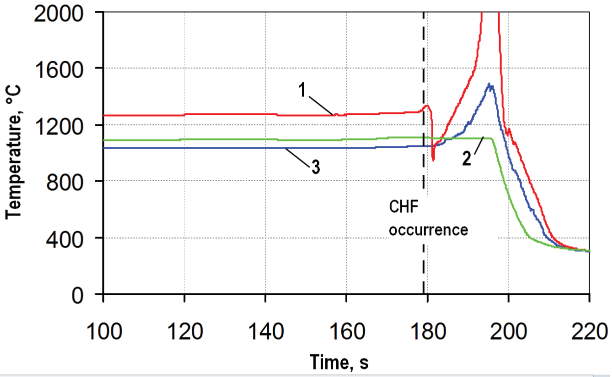

Figure 4. The temperature in the center of the fuel element kernel at the final stage of heating (the 0 s mark corresponds to a time of 12 h 41 min 32 s in the graph of Fig. 3): 1. TC T_1 readings; 2. TC T_2 readings; 3. TC T_3 readings.

junction location in the center of the fuel element kernel, which corresponds to an assembly power of 70.5 kW, is 1080 kW/m^2. The coolant mass flow rate was about 200 kg/(m^{2}s), the pressure in the cooling circuit at the channel outlet (15.6 MPa) decreased compared to the original value.

The method (Bobkov 1999) was used to calculate Q_c for the parameters implemented in the "steady-state crisis" experiment. To obtain a satisfactory correlation between the calculated data and the experimental data, a multiplying factor of 1.29 was determined.

The Q_c calculations were performed taking into account deviations in errors and tolerances (the coefficient of 1.29):

- with a minimum flow rate of 0.122 kg/s and a fuel assembly power of 63.4 kW, Qc is 955 kW/m2;
- with a maximum flow rate of 0.137 kg/s and a fuel assembly power of 75.1 kW, Qc is 955 kW/m2.

Taking into account the error in the FE-length linear power distribution = 2%, the Q_c error is -7.0 – + 3.0%.

The data on the critical heat flux density for a bundle of three fuel elements obtained in the "steady-state crisis" experiment at a coolant mass rate of about 200 kg/(m2s) were used to calculate the parameters of a three-element assembly in the region of low coolant mass rates in preparation of reactor experiments.

Conclusion

A "steady-state crisis" experiment was conducted in the MIR reactor channel, in which the critical heat flux was recorded at a three-element assembly of shortened VVER-1000 fuel elements (the fuel column length = 1000 mm) with parameters close to the calculated values.

Using the results of direct measurement, the critical heat flux was determined for specific experimental conditions. Based on the obtained experimental data for Q_c calculations under similar conditions, it is recommended to use the published method with the introduction of an upward correction.

The experimental data are used to calculate the temperature conditions for testing fuel assemblies in the MIR reactor, particularly, in the experiment with a reactivity-inititated accident (RIA), where, according to the technical requirements, it is necessary to obtain the critical heat flux on the fuel element jacket.

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