4. M. V. Tantashev and G. K. Kholopov, "The method to measure brightness indicatrices of light scattering reflecting coatings," Application for invention No. 4689397/25, 11. 05. 89 (Decision on issuing the USSR author's certificate of 28. 11. 89).

IDENTIFICATION OF CONTACT THERMAL RESISTANCES IN NUCLEAR REACTOR FUEL ELEMENTS. 2. PROCESSING OF EXPERIMENTAL RESULTS

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The results of experimental computational determination of contact thermal resistances (CTR) between the heat-evolving cores and jackets of nuclear fuel elements are presented. The dependence of the CTR magnitude on the linear heat loading of the fuel element is analyzed.

The present paper considers the problems of practical utilization of the parametric identification method suggested in [1]. Experiments were conducted with fuel elements of the water-moderated water-cooled power reactor-440 type (WMWCPR-440) in a medium reactor (MR) [2]. An experimental fuel assembly (FA) consisted of 18 elements arranged in two rows (Fig. 1); it was cooled with water under a pressure of 160 MPa. Fuel elements R1-R6 were equipped with in-fuel internal reactor (IR) thermocouples located in the central hole of the fuel element core in the axial cross section corresponding to the zone of maximum energy release. Major characteristics of the fuel elements are listed in Table 1.

Volumetric distribution of energy release in the FA was determined with the aid of axial and azimuthal arrays of neutron flux sensors [rhodium direct charge detectors (DCDs)]. The heat-transfer agent pressure and temperature were measured at the FA inlet. The heating of the heat-transfer agent over the working section of the FA was measured by a differential chromel–alumel thermocouple, the volumetric heat-transfer agent flow rate was determined with the aid of an orifice plate. An instantaneous energy release in thermometric fuel elements was controlled by a rhodium DCD equipped with an amplifier-corrector BNTK-1A. The rhodium DCD readings are described by the following system of equations

$$J(\tau) = lk\lambda_1 N_1(\tau) + \alpha l_k (\sigma_1 + \sigma_2) n\Phi(\tau),$$
$$\frac{dN_1(\tau)}{d\tau} = \lambda_1 N_1(\tau) + \lambda_2 N_2(\tau) + \sigma_1 n\Phi(\tau), \quad \frac{dN_2(\tau)}{d\tau} = \lambda_2 N_2(\tau) + \sigma_2 n\Phi(\tau), \tag{1}$$

where l, k, n are constants; $N_1(\tau)$ and $N_2(\tau)$ are the concentrations of isotopes Rh^{104} and Rh^{104m}; $\Phi(\tau)$ is the instantaneous neutron flux; α is the instantaneous signal component resulting from the nuclear Compton effect (for the DCDs used in the present experiment $\alpha = 8\%$); $\sigma_1 = 139$B and $\sigma_2 = 11$B are the neutron-capture cross sections for Rh^{104} and Rh^{104m}; $J(\tau)$ is the DCD signal (from 0 to 1-2 μA).

By applying the analog method, the BNTK-1A amplifier-corrector solves the system of equations (1) converting the DCD signal $J(\tau)$ into a normalized signal E^{im} (0-5 V or 0-10 V) associated with the power characteristics of FA and of separate fuel elements by the following relationships

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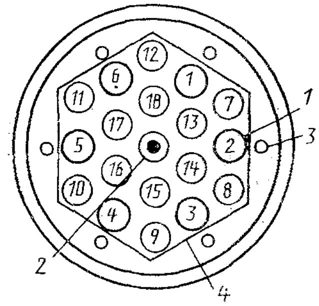

Fig. 1. Thermometric FA: 1) thermometric fuel elements; 2) axial array of DCDs; 3) azimuthal array of DCDs; 4) FA shell.

TABLE 1. Major Features of the Fuel Elements Used to Recover the CTR Values in the Gap between the Fuel and Jacket (the average size of the grains is 5 μm)

Parameters of fuel elements	R1	R2	R3	R4	R5	R6
Diametral clearance, μm	150	200	200	270	150	200
Density of pellets, g/cm^3	10,60	10,50	10,55	10,45	10,50	10,45
Gas pressure in a fuel element, MPa	0,1	0,1	0,1	0,1	0,1	0,1
Gas under the jacket	None	None	Xe	He	Xe	He
Diameter of pellets, mm	7,65	7,55	7,60	7,53	7,65	7,60
Mean load, W/cm	265	260	200	190	195	210

$$E^{im}(\tau) = C_1\Phi(r,\ z,\ \vartheta,\ \tau), \tag{2}$$

$$E^{im}(\tau) = C_2N(\tau), \tag{3}$$

$$E^{im}(\tau) = C_3K_iq_l^i(z,\ \tau), \tag{4}$$

$$E^{im}(\tau) = C_4q_v^i(r,\ z,\ \tau), \tag{5}$$

where C_1, C_2, C_3, and C_4 are constants.

Relations (2)-(5) enable one to calibrate the instantaneous normalized signal from the amplifier-corrector $E^{im}(\tau)$ (of the energy monitor) for the linear heat load or volumetric energy release in the i-th fuel element. To this end, it is sufficient, from the warming-up of the heat-transfer agent (in steady regimes), to determine the thermal power of FA and from the results of neutron-physical calculations and measurements to find the relative powers of fuel elements, axial nonuniformity of energy release, and depression of neutron flux (energy release) along the fuel element radius.

In the experiments being analyzed, we processed the regimes of bringing the reactor up to power, with the power and the energy release in the fuel elements varying from zero to the nominal level with intermediate steady stages. The readings of the probes were recorded by an automated system of acquisition and processing of data (ASAPD) with the frequency of every second. When the recorded data were handled, nonsteady segments with preceding steady segments were isolated. The steady segments were used to calibrate the energy release monitor, with the FA power being found from the formula

$$N = 7{,}48Q\frac{1}{\sqrt{\vartheta}}(I_{\text{out}} - I_{\text{in}}), \tag{6}$$

where N is the FA power, kW; Q is the heat-transfer agent flow rate, $m^3 \cdot kg$; I_{in} is the heat-transfer agent enthalpy at the FA inlet found from the heat-transfer agent temperature and pressure at the FA inlet; I_{out} is the heat-transfer agent enthalpy at the FA outlet found from the heat-transfer agent temperature at the FA inlet and from the warming-up of the heat-transfer agent.

The mean linear thermal loading of fuel elements is associated with the FA power by a simple relationship

$$q_l = N\left\{\sum_{i=1}^{n} L_i\right\}^{-1}, \quad (7)$$

where L_i is the length of the i-th fuel element column; n is the number of fuel elements in the FA.

The linear thermal loading of the i-th fuel element at the place of heat sensor installation is connected with the mean linear thermal load of FA as follows:

$$q_l^i(z_i^t) = \bar{q}_l K_i K_s \Phi(z_i^t)\{\bar{\Phi}\}^{-1}, \quad (8)$$

where z_i^t is the axial coordinate of the heat sensor installation; $\Phi(z_i^t)$ is the neutron flux found with the help of the axial array of DCDs at the section z_i^t; K_s the FA azimuthal nonuniformity factor

$$K_s = n\left\{\sum_{i=1}^{n} K_i\right\}^{-1}, \quad (9)$$

$\bar{\Phi}$ is the mean neutron flux over the FA height

$$\bar{\Phi} = \frac{1}{L}\int_0^L \Phi(z)\,dz. \quad (10)$$

From Eqs. (2)-(10) it follows that

$$q_l^i(z_i^t, \tau) = E^{im} K_q^i(E^{im}), \quad (11)$$

where $K_q^i(E^{im})$ is the calibrating function for the i-th fuel element determined from measurements over the isolated steady segments with the help of relations (3), (6), (7), (8), (9), and (10).

The method suggested for calibrating the energy release monitor makes it possible to take account of the deformation in the axial and azimuthal energy release profile in FA owing to the displacement of the reactor control units with variation in power. This allows one to minimize the uncertainty in the determination of energy release in fuel elements.

The volumetric energy release in the i-th fuel element can be presented in the following form:

$$q_v^i(r, \tau) = q_r^i(r)\, q_\tau^i(\tau), \quad (12)$$

where $q_r^i(r)$ is the relative neutron flux depression along the fuel-element radius found from neutron-physical calculations (it depends on the fuel enrichment and geometry of the heat-evolving core).

The function q_τ^i characterizes the change in the volumetric energy release in time. It is given by the formula

$$q_\tau^i(\tau) = q_l^i(z_i^t, \tau)\left\{2\pi\int_{r_n}^{r_s} q_r(r)\, r\,dr\right\}^{-1}, \quad (13)$$

where r_n is the radius of the central hole of the heat-evolving core; r_s is the outer radius of the fuel element of the heat-evolving core.

The identification of CTR between the heat-evolving core and the jacket in WMWCPR-440 type fuel elements filled with helium and xenon was made by the data of nonsteady measurements in the process of the initial power attainment with fresh fuel and in the process of the third attainment of power two days later. The results were presented as a relationship between CTR and linear heat load which is an overall characteristic of the energy levels of a fuel element.

The results relating to the initial start-up with fresh fuel are presented in Fig. 2. After the power attainment with fresh fuel within the range of linear heat loads from 150 to 240 W/cm, the magnitude of CTR between the fuel and jacket decreases linearly with a rise in the linear heat load. However, as is seen from Fig. 2, with the linear heat load of 190 W/cm in fuel elements R2 and R3 and with the load of 210 W/cm in fuel element R1 one observes the occurrence of a pellet jump which is understood to represent the overall manifestation of the effects of cracking and fragmentation of fuel pellets resulting in a substantial increase in the heat-evolving core diameter and decrease in the width of the gap between the core and the fuel-element jacket.

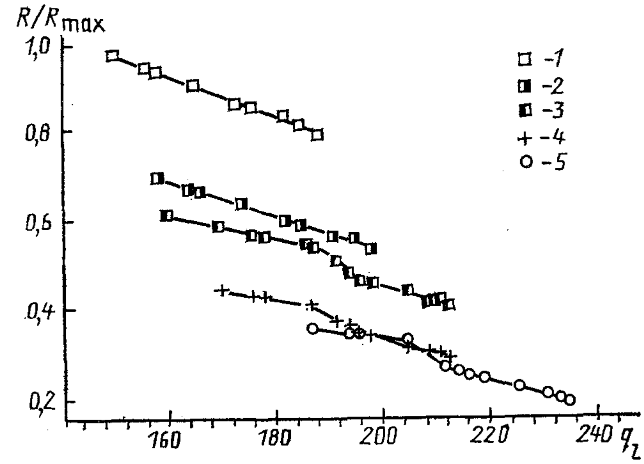

Fig. 2. The first rise in power. Dependence of CTR between the heat-evolving core and jacket on the linear heat load: 1) fuel element R3; 2) R5; 3) R2; 4) R6; 5) R1. q_l, W/cm.

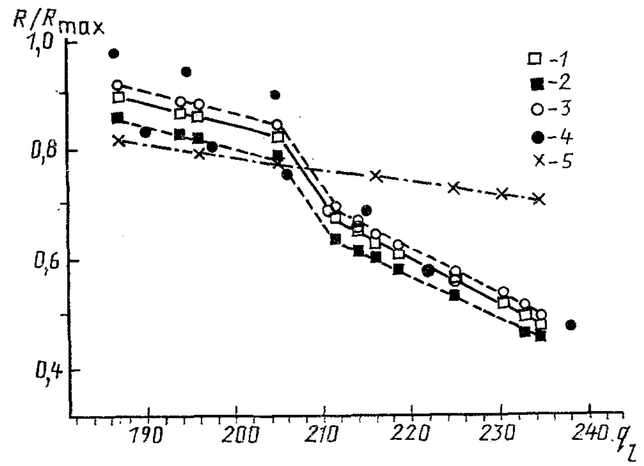

Fig. 3. The first rise in power. Dependence of CTR between the heat-evolving core and jacket of the fuel element R1 on the linear heat load: 1) the recovered value of CTR; 2) $\Delta\alpha = -0.3\alpha$; 3) $\Delta\alpha = +0.3\alpha$; 4) $\Delta q_l = \pm 0.05 q_l$; 5) calculation by the technique of [4].

A similar effect was described by Oguma [3] for a linear heat load of 60 W/cm for BWT-type fuel elements having pellets 13 mm in diameter. The results obtained also agree with the data presented in [5]. At the same time, as is seen from Fig. 3, the gap conductivity calculation by the technique of [4] based on Ross and Stauta's gap conductivity model [5], does not give the pellet jump in the given range of linear heat loads, thus leading to an error in the determination of the fuel temperature of up to 112°C and to a conservative estimation of the hazardous inception of zirconium-steam reaction in the event of disaster with the rupture of supplying pipeline. The foregoing allows a conclusion about the inadequacy of the model used for R4.

Note that in the fuel elements R3 and R4 having the same initial gap of 200 μm, the pellet jump occurs at the same linear heat load of 190 W/cm, whereas in the fuel element R1, having the initial gap of 150 μm, the pellet jump occurs at a greater linear heat load of 210 W/cm.

Such a dependence of the pellet jump threshold on the initial gap size is associated with a change in the fuel plasticity resulting from the change in the fuel temperature level. The dependence of the pellet jump threshold on the CTR magnitude between the fuel and jacket is well illustrated by the fact that in fuel elements filled up with xenon the pellet jump was not observed whatsoever in the experiments analyzed in the process of the initial bringing of the reactor to power with fresh fuel (within the range of linear heat loads up to 240 W/cm).

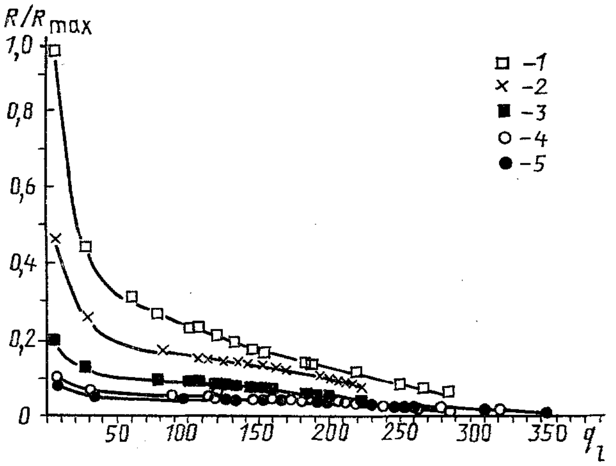

Fig. 4. The third rise in power. Dependence of CTR between the heat-evolving core and jacket on the linear heat load: 1) fuel element R3; 2) R5; 3) R4; 4) R6; 5) R1.

As seen from Fig. 2, in the process of power attainment with fresh fuel for fuel elements filled with xenon, a strong dependence of CTR between the fuel and jacket on the initial gap size is observed, whereas for fuel elements filled with helium such a dependence is nonexistent. Thus, fuel elements R2 and R6 with identical initial gaps between the fuel and jacket have substantially different CTR values at the same linear heat loads, whereas the CTR's of fuel elements R1 and R6 are nearly equal. Presumably, here some additional factors become important, for example, the difference in the composition of sorbed technological gases that exert the maximum effect on CTR in the initial period of irradiation. This problem requires additional investigation.

When solving the inverse problem of determining CTR, the initial data are taken to be a number of the nonmeasured parameters (in addition to those measured) which are determined either from the readings of intrareactor transmitters or from the data of separate experiments and calculations. These are the energy release in a fuel element, external heat-transfer coefficient, thermophysical properties (TPP) of the fuel and jacket. The uncertainty in the assignment of these parameters due to the error in their determination may influence the solution of the inverse heat conduction problem.

Using the data of nonsteady measurements made during the power attainment with fresh fuel, we analyzed the effect of uncertainties in the assignment of nonmeasured initial data for the inverse problem on the error in determining CTR. In processing experimental data, an analysis was made of the influence on the error of the inverse problem solution as exerted by systematic errors in the determination of the coefficient of heat transfer from the fuel element jacket to the heat-transfer agent $\Delta\alpha$ and of energy release (linear heat load in fuel elements at the section of installation of heat sensors) Δq_l.

The calculations have shown that the error in determining the heat-transfer coefficient α has a weak effect on the recovered value of the contact thermal resistance and that a 30% error in the determination of α leads to a 1.5% error in the determination of contact thermal resistance R. The error incurred in the determination of energy release by the above technique due to the error in determining the energy release nonuniformity over FA comprises 5%. Such an error in the determination of energy release leads to a 5.5% error in the determination of R for fuel elements filled with helium in the case of linear heat load of up to 200 W/cm and to a 2% error at higher linear heat loads owing to a jumpwise variation in CTR at a given value of q_l. The latter is due to the change in the gap between the fuel and jacket because of the pellet jump. The influence of the considered uncertainties and the appropriate corridors of solutions are demonstrated in Fig. 3 in relation to the fuel element R1.

The recovered dependences of R on q_l obtained in handling the results of nonsteady measurements in the process of the third power attainment are shown in Fig. 4. During irradiation of fuel elements, fission-product gases (K_s and K_r) are evolved under the jacket, and the jackets of fuel elements are filled with helium when being manufactured. As a result, the curves in Fig. 4 virtually span the entire range of the possible variation of R for a given type of fuel elements.

From the thermophysical properties of the fuel-element materials the most unstable property is the fuel thermal conductivity which varies under irradiation due to the change in the structure and density of the fuel. In this case, the most reliable is the thermal conductivity of the fuel in the initial period of irradiation, analyzed in this work, when thermoradiative structural changes in the fuel have insufficient time to accumulate. According to the data of nonsteady measurements for the third attainment of power, an analysis was performed to determine the effect of uncertainty in the assignment of the fuel thermal

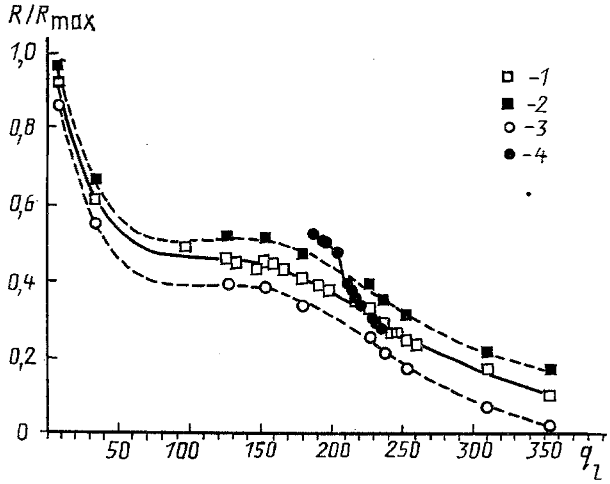

Fig. 5. Third rise in power. Dependence of CTR between the heat-evolving core and jacket of the fuel element R1 on linear heat load: 1) recovered value of CTR; 2) $\Delta\lambda = +0.1\lambda$; 3) $\Delta\lambda = -0.1\lambda$; 4) recovered value of CTR during power attainment with fresh fuel (the first rise in power).

conductivity $\Delta\lambda$ on the error in finding the CTR magnitude. As shown in Fig. 5, the analysis on the example of the fuel element R1 has revealed that the uncertainty in the assignment of the thermal conductivity of fuel leads to an equidistant displacement of the "CTR vs linear heat load" curve downward when the fuel conductivity is underestimated and upward when it is overestimated. This leads to the effect of the narrowing of the range of solutions in the region of low heat loads where the CTR magnitude gradient is maximal for the linear heat load and to a higher error in the determination of CTR in the region of high linear loads, since the CTR value decreases with an increase in the linear heat load. The error in determining the CTR value with allowance for the 10% uncertainty in the assignment of the thermal conductivity of fuel comprised from 8 to 20% depending on the linear heat load magnitude for fuel elements filled with helium and from 4 to 10% for fuel elements filled with xenon. In this case, the 10% errors for the thermal conductivity of fuel were prescribed at the lower and upper limits of the uncertainty range [6]. The analysis performed allows one to estimate the error in determining the CTR value as ±10%. It should be noted that accounting for the uncertainties entails no change in the character of the dependency of R on linear heat load. Moreover, the effect of the equidistant displacement of the curves in Fig. 5 makes it possible to construct a simple and reliable method for determining the magnitude of thermoradiative sintering of fuel by comparing the relations $R = R(q_l)$ obtained for the initial attainment of power and those for the studied range of irradiation. However, for the method to be realized, a purposive change in the loading of fuel element is required, since the process of thermoradiative sintering occupies a relatively small period of time.

The fact that CTR between the fuel and jacket as one of the criteria of nuclear reactor safety, is determined from the results of nonsteady measurements conducted directly in the course of reactor operation without exerting a special effect on the mode of its operation is a significant advantage of the proposed method of investigation. It allows one to determine CTR between the fuel and fuel-element jacket under actual conditions. This is especially important, since the fuel element is a complex technical system whose properties depend on the history of irradiation. The method also allows one to investigate the level of mechanical interaction of the fuel with the jacket depending on the rate of increase in the reactor power. This level determines the reactor safety on regimes that involve the increase in the nuclear reactor power.

NOTATION

R is the contact thermal resistance; τ, time; $N(\tau)$, thermal power of FA; $q_l^i(z, \tau)$, linear thermal power of i-th fuel element; $q_v^i(r, z, t)$, volumetric energy release in i-th fuel element; K_i, relative energy release in i-th fuel element; $\Phi(r, z, v, \tau)$, density of neutron flux in FA; Q, heat-transfer agent flow rate; z_l^t, axial coordinate of heat sensor installation; v, specific volume of heat-transfer agent.

LITERATURE CITED

1. E. A. Artyukhin, A. V. Nenarokomov, A. P. Tryanin, et al., Inzh.-Fiz. Zh., **60**, No. 3, 478-487 (1991).
2. V. V. Goncharov, E. N. Babulevich, P. I. Shavrov, et al., Experience on Operation of an MR Reactor and Testing of Fuel Elements and Materials, Preprint No. 91, Institute of Atomic Power, Moscow (1965).
3. M. Oguma, Nucl. Eng. Des., **76**, 35-45 (1983).
4. P. N. Strijov, F. Pazdera, and V. V. Yakovlev, "The improved version of the PIN cod and its verification," Preston, England, 19-22 September 1988, IAEA-TC 657/34.
5. A. M. Ross and R. L. Staute, Heat Transfer Coefficient Between VO_2 and Zircalloy 2. CRFD-1075/AEC (1952).
6. V. I. Kolyadin, E. P. Il'in, A. G. Kharlamov and V. V. Yakovlev, Thermal Conductivity of Uranium Dioxide, Preprint No. 2227, Institute of Atomic Power, Moscow (1972).