SOME PROBLEMS OF EMERGENCY COOLING OF THE THERMAL RESEARCH REACTOR

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Inzhenerno-Fizicheskii Zhurnal, Vol. 10, No. 1, pp. 46–50, 1966

UDC 621.039.566.8

The cooling regime of a reactor in emergency shutdown of the circulating pumps is studied on a type MN-7 electronic model and on a model with electrical heating of a fuel element assembly.

In the normal working of the thermal research reactor the coolant circulates downwards, so that in emergency shutdown of the circulating pumps the active zone will be cooled by a convective flow of water only when, on account of the residual heating in the active zone, the direction of motion of coolant changes and settles to a rising convective flow. In this case, the instantaneous value of the directional velocity of the water will at some moment be zero.

It is natural, therefore, to consider the temperature regime of the fuel element assembly from the point of view of a critical heat exchange situation. For this case, it is necessary to obtain an upper estimate of the temperature of the wall of the fuel element assembly. Such an approach to the solution of the problem essentially permits the simplification of the method of finding these estimates. First of all, the processes in the active zone may be considered in an elementary approximation, i.e., without taking account of the spatial dependence of the basic variables, writing the basic equation for a volume element as the equation for the total derivative and choosing the initial conditions for the most highly rated volume element in the active zone.

Then, for a region with the cross section shown in Fig. 1 and length equal to the channel height, the total system of differential equations describing the transient process brought about by emergency shutdown of the circulating pumps may be written in the form

$$m_0 \frac{d^2x}{d\tau^2} + \operatorname{sign}\left[\frac{dx}{d\tau} - W(\tau)\right] C_x S_r \frac{\gamma_w}{2}\left[\frac{dx}{d\tau} - W\tau\right]^2 - (m_r - m_w) g + F_w = 0, \quad (1)$$

$$\delta K = \delta K\left[x(\tau, \tau_0)\right], \quad (2)$$

$$\frac{dQ_n}{d\tau} = \frac{\delta K(\tau) - \beta}{l} Q_n + \lambda Q_c + \dot{S}_n, \quad (3)$$

$$\frac{dQ_c}{d\tau} = \frac{\beta}{l} Q_n - \lambda Q_c, \quad (4)$$

$$\frac{dt_c}{d\tau} = \frac{1}{m_e C_e}\left[Q_n(\tau) + 0.07 Q_n(0) - G_w C_w (t_2 - t_1)\right] - \frac{m_{wc} C_w}{2 m_e C_e} \frac{dt_2}{d\tau}, \quad (5)$$

$$\frac{dt_2}{d\tau} = -2W \frac{t_2 - t_1}{L} + \frac{\alpha S}{V \gamma_w C_w}(2t_c - t_2 - t_1), \quad (6)$$

$$\frac{dW}{d\tau} = -\eta W^2 + \varepsilon (t_2 - t_1). \quad (7)$$

As the pumps are lost, so the quantity of water available for heat removal changes. The change in water velocity is described by (7). At the moment of tripping, the emergency protection system injects negative reactivity as determined in (1) by movement of the emergency rods, in (7) by movement of water in the channel and also the dependence of rod efficiency on its movement (2). The power drops as in (3) and (4).

The heat balance and heat emission equations, (5) and (6), give the relationship between the mean wall temperature, the mean coolant temperature and the generated and extracted powers. The power drop equations consider one group of delayed neutrons.

Numerical values of the coefficients in (1)–(7) are chosen on the basis of rational considerations, always taking into account the need to obtain an upper estimate of the temperature.

As a first step, (1)–(7) were solved on a nonlinear electronic model type Mn-7, working in the integrating regime. The following assumptions were introduced:

1) The β- and γ-ray power was taken as constant and equal to 7% of the nominal reactor power.

2) The efficiency of the emergency rods was chosen minimal from the point of view of safety, and equal to 2%.

3) the effect of the temperature coefficient of reactivity was ignored.

The basic conclusion from the results obtained at this stage may be expressed as follows to decrease the thermal stress acting on the fuel element assembly owing to the delay in shutting down the reactor after the circulating pumps stop, it is necessary to reduce as far as possible the delay in the emergency protection, τ_0.

By comparing the curves of Fig. 2 is is seen that the initial rise in temperature is substantially reduced by decreasing τ_0 from 0.8 to 0.1 sec. The existing technical possibilities enable the emergency protection system to be made sufficiently effective and quick-acting. This conclusion permits the second stage of the study to be considered.

Since the drop in power at the moment of tripping of the emergency protection is determined by the efficiency and speed of action of the latter, and afterwards is described as a slowly-decreasing function,

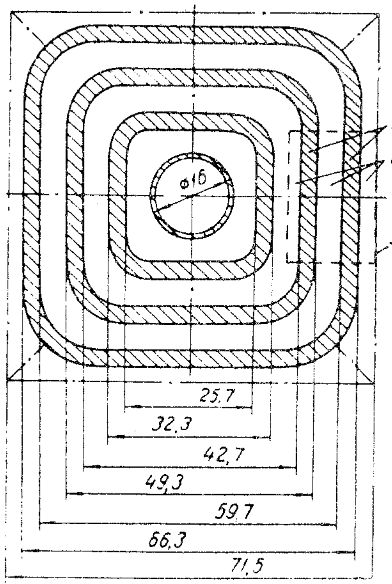

Fig. 1. Cross section of fuel element assembly: 1) fuel; 2) water; 3) cross section of modeled element of fuel element wall.

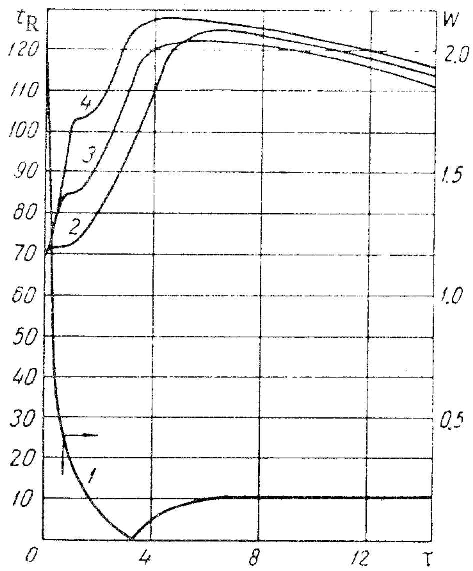

Fig. 2. Change in temperature of fuel element assembly for various delays of the emergency protection system: 1) coolant velocity $W(\tau)$; 2, 3, 4) wall temperature (°C) for τ_0 = 0.1, 0.3 and 0.8 sec, respectively.

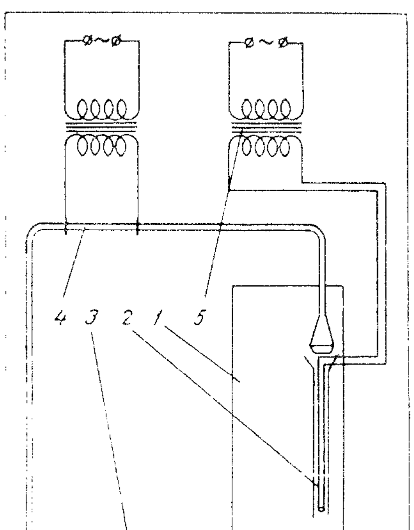

Fig. 3. Principal scheme of the stand: 1) tank; 2) working section; 3) circulation system; 4) electrical heating of water in the loop; 5) power transformer.

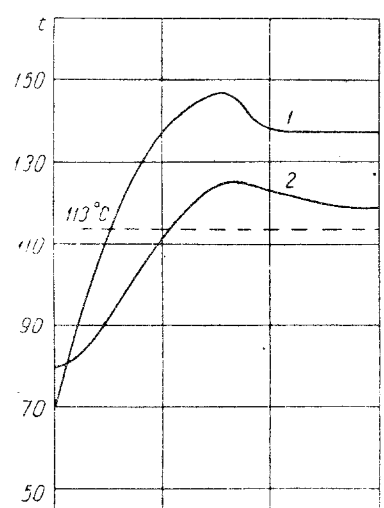

Fig. 4. Change of fuel element assembly wall temperature for q = 349 joules/m^2 sec: 1) electrical model; 2) thermal stand; t in ° C, τ in sec.

for sufficiently rapid emergency protection it may be assumed that the decrease in power with time of tripping of the emergency protection and the reduction in flow through the active zone are such that there is essentially no change in the temperature of the fuel element walls after tripping of the emergency protection (Fig. 2). In connection with the fact that the subsequent change in power is slow (dependent on the delayed neutron sources) the power may be taken as constant. Hence, assuming sufficiently rapid protection, the problem may be studied for the specified condition modeled on reverse coolant circulation with constant heat generation and with the condition $t_c|_{\tau=0} = t_{c\,nom}$.

In this case, we first obtain an upper estimate of the temperature of the fuel element assembly wall. This new assumption permits a description of the problem in (5)–(7) taking account of the dependence of the heat transfer coefficient for convective flow of water ($\alpha = 39\,p^{0.5}\cdot\Delta t^{2.33}$) [7] and the choice of the initial conditions as calculated nominal temperatures.

In our case, the pressure in the active zone is $1.67 \cdot 10^5$ N/m^2, $t_s = 113°$ C.

The problem was solved by modeling on the MN-7 and on a model element of the fuel element assembly (Fig. 1, 3) in a special thermal hydraulic stand (Fig. 3).

The maximum power which may be dissipated with transformers is 50 kW. Assuming that the maximum heat flux for the operating reactor at nominal power is no more than 700 joules/m^2 · sec, two parallel plates of stainless steel were substituted for the fuel element in the stand (Fig. 1) to achieve such a heat flux. All the dimensions of the plates, except their thickness, corresponded to the dimensions of the element of the fuel element assembly. The thickness of the plates was chosen to be 1 mm for a 3.3 mm thick plate in the fuel element assembly. The working plates were contained in a demountable case, such that the hydraulic properties of the model coincided with those of the particular part of the fuel element assembly.

The tank, in which the working part was placed, was modeled on the thermal research reactor tank. Its dimensions guaranteed constant water temperature at the inlet to the working section for a time exceeding the duration of the studied transient.

The temperatures of the plates in the middle at a high point were measured by low-gauge chromel-alumel abraded thermocouples connected to a loop oscillograph OT-24 with a special filter with time constant $T_f = 0.3$ sec. This value of T_f is sufficiently low for the effect of the filter on the studied transient to be neglected. In addition to the temperature of the plates, the following parameters were measured: water temperatures at inlet and outlet of the working section, power generated in the working section, and water flow.

The studies were made for heat fluxes of 167, 279, 349, and 524 joules/m^2 sec. After the experiment, the plates were examined and no indications of boiling crisis were found.

Comparison of the results obtained on the stand and on the MN-7 model showed that the values of temperature from the latter were higher than those from the stand, even in the case when the value of the initial conditions in the stand was higher than in the model. Figure 4 shows curves of the change of temperature of the working plates for the stand and for the model at a heat flux of 349 joules/m^2 · sec.

To compare these curves, it should be remembered that the error of the model solution of (5)–(7) is 10%, and the error of the measured temperature in the stand is about 2% of the maximum value of the function.

Analysis of the results shows that with heat fluxes of 167 joules/m^2 · sec, and above, bubbling of the cooling water boils up if the circulating pumps fail. With increase in heat flux, the intensity of boiling increases, but up to a heat flux of 524 joules/m^2 · sec nucleate boiling persists. Hence, an emergency, given sufficiently rapid and effective protection, does not present a hazard from the point of view of a heat transfer crisis in the reactor fuel element assembly, although reversal of coolant circulation occurs. Thus, it is not necessary to create special emergency cooling systems, and the reactor tank can be used for cooling in unforseen shutdowns of the circulating pumps.

NOTATION

χ—emergency control rod movement; τ—time; m_R—rod mass; m_0—sum of reduced mass of movable part of drive and mass of control rod; $W(\tau)$—velocity of water in fuel element assembly; C_χ—drag coefficient of rod; S_R—rod cross section; γ_w—water density; m_{wr}—mass of water in rod volume; F_B—rod driving force; η—coefficient of resistance; δK—reactivity; Q_n—power; β—delayed neutron fraction; l—mean life of neutrons; λ—mean decay constant of delayed neutron source; Q_c—power due to fissions produced by delayed neutrons; S_N—power per unit time due to independent neutron source; t_s—saturation temperature; t_c—wall temperature; m_e—mass of element of fuel element assembly; C_e—specific heat of material of fuel element assembly; $0.07Q_n(0)$—power due to residual γ-radiation; G_w—water flow rate; C_w—specific heat of water; t_1—water temperature at channel inlet; t_2—water temperature at channel outlet; m_{wc}—mass of water in a channel; h—channel height; α—heat transfer coefficient; S—perimeter of working channel; V—area of cross section of channel containing water; ε—temperature coefficient of convective force; τ_0—delay of emergency protection system; q—heat flux.

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20 October 1964

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