APPROXIMATE MODEL FOR EVALUATION OF THERMAL-HYDRAULIC TRANSIENTS ASSOCIATED WITH RAPID POWER INCREASE IN RESEARCH NUCLEAR REACTOR

by

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A simple model, for the estimation of changes in the nuclear fuel element cladding temperature as well as the conditions of the formation of the onset of nucleate boiling, is proposed. The results of this estimation are sufficient to assess the effect of the transient with the peak of the reactor's power on the device's safety, without the need for time-consuming thermal calculations. The basic parameters determined using the proposed model are the maximum wall temperature of the device in a hot spot, the time constant of the wall temperature change, and the course of changes in the onset of nucleate boiling ratio in time. The model may be used for investigating the thermal behavior of thin heat-generating and water-cooled elements (such as fuel elements or uranium irradiation targets) during rapid power rise. The results of the temperature estimation with the proposed model were tested considering the hot spot in the MR-6 type nuclear fuel. The SN code with coupled kinetics and thermal-hydraulics, developed in the MARIA reactor was used to validate the results. The maximum cladding temperature, the thermal time constant and the onset of nucleate boiling ratio parameter simulated by the SN code and the proposed scheme appeared to be very consistent.

Key words: reactor safety, thermal hydraulics, lumped-parameter model, heat transfer, research reactor

INTRODUCTION

Nowadays nuclear reactor accident scenarios are analyzed with validated thermal-hydraulic codes, currently using the best estimate plus uncertainty methodology [1]. However, such analyse is time-consuming and not always necessary – thus some of the transients might be preceded or even replaced with the simpler approach.

One of the transient states, analyzed in the MARIA Reactor Safety Report, is the reactivity disorder. It is a transient state induced by the reactor itself leading to a short-term power peak under the unchanged cooling conditions. This power peak affects all devices and target materials installed in the core [2].

The postulated reference event initiating the short-term, yet significant power peak in the MARIA reactor is a break of the control rod and introduction of the reactivity at the rate of +1.5 $\$s^{-1}$ (where \$ is the unit of reactivity normalized to the delayed neutron fraction). Within 0.6 seconds the reactor scram proceeds by dropping other absorption rods. Figure 1 shows the course of a single fuel element power change during a transient related to the occurrence of the power peak. In the steady-state before the power peak, the fuel element works with stable thermal power of 1.8 MW_{th} (corresponding with the total reactor power of 30 MW_{th}). After about 0.4 seconds, the fuel element power reaches the maximum value of 8.64 MW_{th} (in-

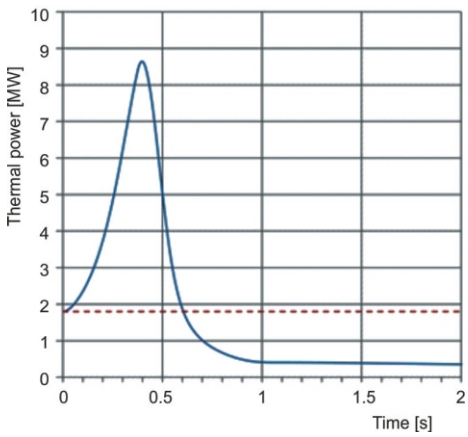

Figure 1. Power time series during transient; —— transient ----- steady-state

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crease 4.8 times) and starts to decrease. The nominal level of 1.8 MW_{th} is reached again after 0.6 seconds from the beginning of the event. The energy generated in the fuel element in the power peak in addition to the steady-state level (the area above the dashed line) is in this case about 1.7 MJ [2, 3].

During the entire transient state, a constant nominal flow rate of the coolant in the cooling circuit is assumed. In the case of experimental devices, target materials, or other core elements in which heat is generated, it may be assumed that their power changes at the same ratio as in fuel element, as described by point kinetics equations [4]. The effect of the transient state with the power peak at the maintained coolant flow is the short-term temperature increase in the device and then, due to reactor scram, its drop to a level lower than the steady-state temperature level, preceding the transient state [5].

THE LUMPED THERMAL CAPACITY MODEL

When constructing a simplified transient model that describes rapid power increase, a number of assumptions were adopted, which will be justified later in the chapter. These simplifications are outlined below:

– Reactor devices for which thermal analyzes are carried out in the transient state are characterized by the low thermal resistance of conduction in comparison with a thermal resistance of heat absorption by the cooling medium. In this case, the model may be used in bodies with a low thermal resistance (lumped thermal capacity model, LCM) [6],
– The model performs a point estimation of thermal parameters: wall temperature, the onset of nucleate boiling ratio, ONBR – for definition see eq. (15), of the hot spot in a transient state and omits the spatial distribution of these parameters. It is assumed that based on separate calculations, the thermal parameters of the hot space are determined in the conditions of the steady-state preceding the power peak.
– The geometry of the device in a hot spot is usually a plate or pipe cooled by flowing water. The plate can consist of several layers (*e. g.* cladding, core, cladding) characterized by good thermal conductivity. An important simplification of the model is the assumption that the thermal properties of all the layers are the same, as well as the uniform density of the spatial heat source. Figure 2 shows a diagram of a plate or pipe section in a hot spot, adopted in the model.
– It is assumed that the temporal course of changes in the reactor power in a transient is known, determined based on other calculations, most often using the reactor dynamics model [7].

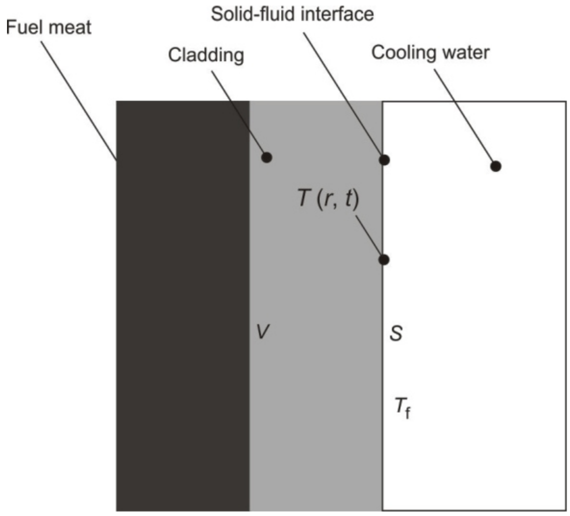

Figure 2. The system cross-section through the considered point: *V – nuclear element volume, T(r, t) – temperature as a function of radius and time, S – cladding area, T_f – cooling water temperature*

– After the power peak, the reactor is shut down. However, the reactor's power does not drop to zero. Two components remain, which in practice can be considered as permanent in a noticeably short period of time [8]. Those components are neutron power of the reactor, dependent on the introduced negative reactivity and thermal power of fission products, which can be considered as a constant value (in the short period after the reactor is turned off), equal to 7 % of thermal power in the steady-state. To assess the neutron power level immediately after switching off the reactor, the prompt-jump approximation may be used [9].
– The constant temperature of the cooling water is assumed, *i. e.*, the changes in the water temperature at the inlet and the change of the water temperature distribution in the transient state are ignored.
– Changes in the physical parameters (density, specific heat, *etc.*) are not considered when the temperature changes.

THE MATHEMATICAL FORMULATION

Let us start with introducing the concept of differential temperature, ϑ, as a difference of cooling water temperature and the temperature in the heat-generating solid plate

$$\vartheta(x,t) \quad T(x,t) \quad T_f \tag{1}$$

The temperature field in the plate defined in this way complies with Fourier's equation of heat conduction

$$\rho c \frac{\partial \vartheta}{\partial t} \quad \lambda \quad {}^{2}\vartheta \quad \ddot{q} \tag{2}$$

with the boundary condition

$$\lambda \frac{\partial \vartheta}{\partial n}\bigg|_S \quad \alpha \vartheta_S \tag{3}$$

After integrating eq. (2) by volume (per length unit) and applying Gauss' theorem with the boundary condition (3), we get eq. (4), in which $\overline{\vartheta}_V$ and $\overline{\vartheta}_S$ are the average temperatures, respectively by volume and on plate surface as defined by eqs. (5) and (6)

$$\frac{\partial \overline{\vartheta}_V}{\partial t} \quad \frac{2\alpha S}{\rho c V}\overline{\vartheta}_S \quad \frac{1}{\rho c}\dddot{q} \tag{4}$$

$$\overline{\vartheta}_V \quad \frac{1}{V} \quad \vartheta \,\mathrm{d}V \tag{5}$$

$$\overline{\vartheta}_S \quad \frac{1}{S} \quad \vartheta \,\mathrm{d}S \tag{6}$$

The criterion of using the model of heat exchange in bodies with a low thermal resistance of conduction (LCM) is the low value of the Biot criterion [10, 11], defined as the ratio of thermal resistance of conduction to the thermal resistance of convective heat transfer. In the case of a double-sided cooled plate or pipe, the Biot number is given by

$$\mathrm{Bi} \quad \frac{\alpha d}{2\lambda} \tag{7}$$

The MR-6 nuclear fuel elements used in the MARIA reactor, consist of six concentric tubes, each having a thickness of $d = 0.002$ m as presented in fig. 3. Tube thickness is similar to the corresponding dimension in the other research reactors [12, 13]. Heat transfer coefficient has an order of magnitude $\alpha = 10000$ $\mathrm{Wm^{-2}K^{-1}}$ and thermal conductivity $\lambda = 100$ $\mathrm{Wm^{-1}K^{-1}}$ [14, 15]. Therefore, the Biot number is roughly 0.1 and fulfils the condition Bi << 1.

In the LCM model, the assumption that $\overline{\vartheta}_V$ and $\overline{\vartheta}_S$ are equal exists, therefore from now on, both will be marked as ϑ [17]. Additionally, let us introduce the thermal time constant of the plate – τ_0, as described as

$$\tau_0 \quad \frac{\rho c d}{2\alpha} \tag{8}$$

With that constant eq. (4) can be written as

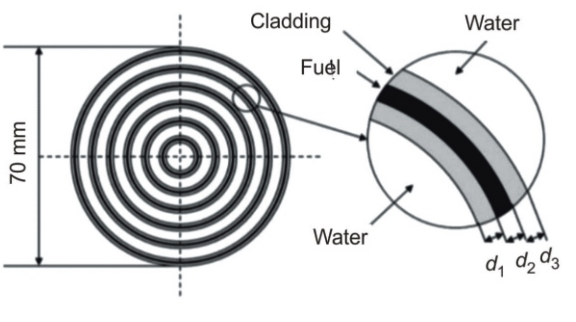

Figure 3. The MR-6 type fuel cross section [16]: $d_1 = 0.6$ mm, $d_2 = 0.8$ mm, $d_3 = 0.6$ mm

$$\frac{\mathrm{d}\vartheta}{\mathrm{d}t} \quad \frac{1}{\tau_0}\vartheta \quad \frac{d}{\alpha \tau_0}\dddot{q} \tag{9}$$

In the steady-state, solution of the eq. (9) is given by

$$\vartheta_0 \quad \frac{d}{a} q_0 \quad \frac{\dddot{q}_0}{\alpha} \tag{10}$$

Both the temperature difference between the cladding and cooling water in the hot spot – ϑ and heat flux per area $\ddot{q}_0$ must be taken from the steady-state calculation. If heat transfer coefficient, α, is unknown, it can be calculated from eq. (10).

TEMPERATURE AND ONBR CHANGES DURING TRANSIENTS

Knowing the course of changes in the reactor power in a transient state, counted relative to the steady-state power, let us introduce the function of relative power changes η(t), given by eq. (11) with the initial condition $\eta_{(0)} = 1$

$$\eta(t) \quad \frac{P(t)}{P(0)} \tag{11}$$

Equation (10) can be now written as eq. (12) and its solution is given in eq. (13).

$$\frac{\mathrm{d}\vartheta}{\mathrm{d}t} \quad \frac{1}{\tau_0}\vartheta \quad \frac{1}{\tau_0}\vartheta_0 \eta(t) \tag{12}$$

$$\vartheta(t) \quad \vartheta_0 \mathrm{e}^{\ t/\tau_0} \ 1 \quad \frac{1}{\tau_0} \ {}_0^t \mathrm{e}^{\tau/\tau_0}\eta(t)\mathrm{d}\tau \tag{13}$$

As can be seen from eq. (13), the only parameter that characterizes the dynamics of fuel element temperature changes is the time constant, τ_0, that combines thermal parameters of the plate and heat transfer coefficient.

Equation (13) has to be integrated numerically; after adding a constant water temperature, a time course of temperature changes is obtained as

$$T(t) \quad \vartheta(t) \quad T_f \tag{14}$$

To perform the safety analysis, two additional parameters are needed: ONBR and the onset of nucleate boiling temperature (TONB). They are given by eqs. (15) and (16), respectively [18, 19]

$$\mathrm{ONBR} \quad \frac{T_{\mathrm{ONB}} \quad T_{\mathrm{fin}}}{T \quad T_{\mathrm{fin}}} \tag{15}$$

$$T_{\mathrm{ONB}} \quad T_{\mathrm{sat}} \quad 0.182\frac{\ddot{q}^{\,0.35}}{p^{0.23}} \tag{16}$$

In the transient state, both the TONB (due to the heat flux change) and the wall temperature of the plate are changed. It is assumed, however, that the other parameters are constant, *i. e*., pressure, the water temperature at the inlet to the gap, and saturation temperature.

In summary, to determine the maximum plate temperature and the minimum ONBR parameter in a transient state with a power peak, the following data should be used:

– Relative power change $\eta(t)$ in a transient state (based on reactor dynamics calculations).
– Thermal parameters of the experimental device or fuel element in steady-state: water temperature at the inlet to the cooling channel, water temperature and plate temperature in the hot spot, heat flux and water pressure in the hot spot, and the water saturation temperature corresponding to this pressure.
– Plate or tube parameters (effective): density, specific heat, thermal conductivity coefficient, and plate thickness.

COMPARISON WITH NUMERICAL RESULTS

The results of the estimation according to the proposed model were tested for the hot spot in the MR-6 fuel, where full calculations of the transient state were carried out using the SN code [5]. The SN code is coupled kinetics and thermal-hydraulic code used for the MARIA reactor analyses. Its accuracy has been validated by RELAP5 Mod 3.3 [20, 21].

The MR-6 type nuclear fuel element contains six concentric tubes with 19.7 % enriched uranium in the form of UO_2. The fuel layer is surrounded by aluminum cladding. In the MARIA reactor core, each fuel element is located inside pressurized Field-tube channels [22].

In the steady-state, the hot spot is located on the inner wall of the 6th (outer) MR fuel pipe, 195 mm beneath the core mid-height. The fuel parameters adopted for the SN code were:

– power of the fuel element: 1.8 MW,
– water temperature at the inlet to the gap: T_{fin} = 45 °C,
– water flow through the fuel channel: 25 m^{-3}h^{-1}.

The parameters of the hot spot, calculated by the SN code, are:

– water temperature: T_f = 77.9 °C,
– wall temperature: T = 136.5 °C,
– heat flux: $\ddot{q}$ = 1.798 MWm^{-2},
– heat transfer coefficient: = 3.07 104 Wm^{-2}K^{-1},
– water pressure: p = 15.6 bar, where 1 bar = 10^5 Pa,
– saturation temperature: T_{sat} = 199.7 °C.

For calculation of the thermal time constant, the volumetric heat capacity ρc = 2.4 10^6 Jm^{-3}K^{-1} was assumed. It is an average value between 2.37 10^6 Jm^{-3}K^{-1} for the fuel meat and 2.47 10^6 Jm^{-3}K^{-1} for the cladding.

With the plate thickness d = 2 mm, the time constant, calculated using eq. (8), is: τ_0 = 0.078 seconds.

Using these parameters and the time course of the relative power of the fuel element in the transient

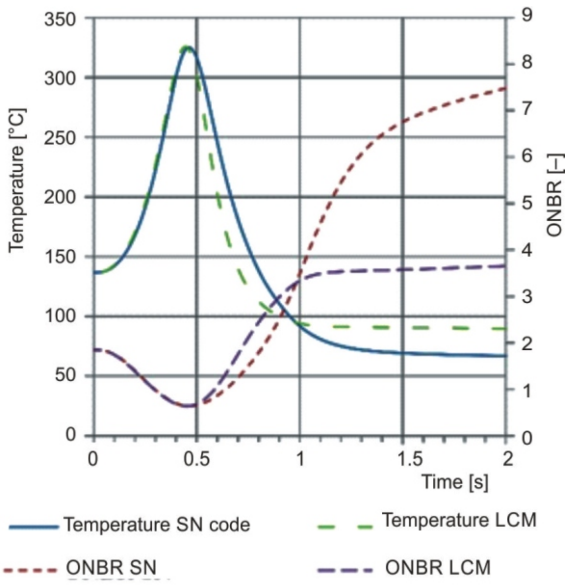

Figure 4. Cladding temperature and ONBR at the hot spot during the calculated transient

state, the temperature changes of the cladding and the ONBR parameter in the hot spot can be calculated using the LCM equations. Figure 4 presents a comparison of temperature changes of the cladding and ONBR parameter value in a transient state, calculated employing the SN code and by using the approximate LCM model.

CONCLUSIONS

An approximate model based on the lumped capacity method allows a very good reproduction of the maximum cladding temperature after transient with a power peak. A similar conclusion applies to the minimum value of the ONBR parameter in the same scenario.

A simplistic assumption about the constant temperature of the cooling water causes discrepancy of the LCM with an accurate calculation model, but only for times after the occurrence of temperature peaks. This range of transients is, however, not essential for the safety analysis of this particular transient.

The use of a simplified LCM is justified in the case of the safety analysis of experimental devices in the form of a relatively thin plate, or pipe cooled with water and the fulfilment of the condition of a small Biot number (Bi $\ll$ 1).

NOMENCLATURE

c – specific heat [Jkg^{-1}K^{-1}]
d – fuel plate or pipe thickness [m]
t – time [s]
T – temperature [°C]

T_{fin} – water temperature at inlet [°C]
T_{ONB} – the onset of nucleate boiling temperature [°C].
T_{sat} – water saturation temperature [°C]
P – reactor power [MW]
p – pressure [bar], 1 bar = 10^5 Pa
$\ddot{q}$ – heat flux [Wm^{-2}]
$\dddot{q}$ – heat generation volumetric density [Wm^{-3}]

Greek letters

α – heat transfer coefficient [$Wm^{-2}K^{-1}$]
ϑ – non-dimensional temperature [–]
ϑs – temperature difference between the cladding and cooling water [°C]
λ – thermal conductivity [$Wm^{-1}K^{-1}$]
ρ – density [kgm^{-3}]

AUTHORS' CONTRIBUTIONS

All authors have contributed equally.

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АПРОКСИМАТИВАН МОДЕЛ ЗА ПРОЦЕНУ ТЕРМОХИДРАУЛИЧКИХ ПРЕЛАЗНИХ ПОЈАВА ПОВЕЗАНИХ СА БРЗИМ ПОВЕЋАЊЕМ СНАГЕ У ИСТРАЖИВАЧКОМ НУКЛЕАРНОМ РЕАКТОРУ

Предложен је једноставан модел за процену промена температуре кошуљице нуклеарног горивог елемента као и услова формирања почетка нуклеарног кључања. Резултати ове процене довољни су за анализу утицаја прелазног стања са пиком снаге реактора на сигурност уређаја, без потребе за дуготрајним топлотним прорачунима. Основни параметри утврђени применом предложеног модела су: максимална температура зида уређаја у жаришту, временска константа промене температуре зида, и ток промена односа почетка нуклеарног кључања по времену. Модел се може користитит за испитивање топлотног понашања водом хлађених танких елемената који стварају топлоту (као што су горивни елементи или мете озрачиване уранијумом) током брзог пораста снаге. Резултати процене температуре предложеним моделом тестирани су помоћу жаришта нуклеарног горива типа MR-6. За потврђивање резултата коришћен је SN код са спрегнутом кинетиком и термохидрауликом, развијен за реактор MARIA. Показало се да су максимална температура кошуљице, термичка временска константа и параметар односа почетка нуклеарног кључања симулирани SN кодом и предложена шема врло конзистентни.

Кључне речи: сигурност реактора, термохидраулика, модел са груписаним параметрима, пренос топлоте, истраживачки реактор