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Research Article

Diagnostics of the critical heat flux state of a VVER reactor based on a channel steaming model*

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Abstract

The purpose of the study is to develop a model for predicting the process of a critical heat flux state with the VVER reactor core channel steaming. The model describes the dynamics of the nuclear reactor behavior in conditions of uncertainty, which are typical of abnormal situations, based on information on the process of heat exchange in the core process channels.

The use of the proposed model leads to an increase in the speed of response due to a simplified procedure to calculate the parameters of the heat exchange process in the reactor core. The quality of the reactor state assessment is improved through the prediction of the heat exchange process parameters and determination of the critical heat flux parameters in the core prior to the onset of surface boiling the potentiality of which is not predicted in modern VVER in-core monitoring systems.

A modification of the mathematical model has been proposed which offers the simplest possible way of using the advantages of neural networks in diagnostics. The model can be used to develop systems for diagnostics of in-core anomalies and systems for adaptive control of the VVER-type reactor thermal power.

Keywords

Nuclear reactor, critical heat flux, power density, thermophysical model, identification, neural networks

Introduction

The cores of large NPP reactors have complex structures with many fuel assemblies and control rods operating in stress conditions. Solving the problem of optimizing the power density within such cores and improving the cost effectiveness of the NPP operation required the development of dedicated tools and automatic devices for monitoring and control of nuclear reactors (Yemelyanov et al. 1981).

In a general case, the required scram signals and settings are based on computational and experimental studies and depend on the reactor type. Often, the parameters that define the reactor plant safety (fuel and cladding temperature, hot spot location, boiling point, etc.) cannot be measured directly. In this case, safe variation intervals of measured parameters and the core settings for each of them need to be determined based on physical and thermal engineering calculations.

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Describing the operation of a nuclear power plant (NPP) during local disturbances as a random process, the key indicators of which in the maximum thermal loading mode are the surface boiling parameters, namely the coolant pressure, the coolant temperature in the mixing chamber, and the average volumetric steam quality (Yemelyanov et al. 1981, Yemelyanov et al. 1975, Kirillov et al. 1990), makes it possible to use stochastic models as adaptive control systems.

On the one hand, there is a need for building simple and effective models of the heat exchange process in the reactor fuel channel which allows one to conclude, based on a small number of parameters, that the heat exchange is abnormal at early stages. On the other hand, this requires the development of fast-response adaptive control systems the models of which would take into account the stochastic nature of the heat exchange process.

The mechanisms of a critical heat flux in channels depend to a great extent on the two-phase mixture flow mode, the liquid subcooling and the heat flux density (Kirillov et al. 1990). Interpretation of experimental data and the form of their description by empirical dependences are somewhat difficult. The traditional way is to present experimental data in the coordinates $q_{cr}(x_{cr})$ (q_{cr} is the critical density of the heat flux, and x_{cr} is the critical steam quality), this requiring x_{cr} to be calculated from a heat balance equation which leads to additional errors. Besides, the problem of nonterminal adaptive control for the nuclear reactor thermal power based on a quadratic criterion suggests that there is a system state vector (core channel thermal power vector) which cannot be obtained based on existing thermophysical and thermodynamic models.

Many papers (Bragin et al. 1987, Sharayevskiy et al. 2001, Sharayevskiy 2000, Kovetskaya 2009, Kachur 2009, Popov and Kachur 2009, Novikov and Voskresenskiy 1977, Kramer 1960, Gerliga and Skalozub 1992, Dolinskiy et al. 2005, Popov et al. 2007, Leontyev and Olimpiyev 2007, Kirillov 2005) indicate that identifying a critical heat flux of the first kind on the VVER reactor fuel element surface is a challenging problem. The critical heat flux of the first kind is defined by the transition of nucleate boiling to film boiling which takes place at high specific heat fluxes.

Recently, there has been a heightened interest in using the capabilities of neural networks for the NPP monitoring and control (Shapovalova and Sharayevskiy 2008, Kachur and Bogma 2018). This requires developing models adaptable to the mathematical tools of neural networks (Wasserman 1992, Khaykin 2006).

Determination of the core channel steaming trend

Development of a mathematical model for the heat generation in a nuclear reactor includes

- plotting a diagram of the reactor state based on thermodynamic characteristics, experimental dependences and current variations in the key parameters;
- prediction of the parameter variations and calculation of the parameter critical values;
- predictability of an abnormal heat exchange at the surface boiling onset stage and/or in the event of a sharp variation in the heat exchange process key parameters (Kachur 2009).

Using the results of analyzing models of critical phenomena during boiling in a two-phase flow, an empirical model of determining the steam quality for different boiling stages, based on Z.L. Miropolsky's data, and empirical studies of surface boiling in a channel of the IR-100 nuclear research reactor have been chosen to investigate the process of heat exchange in the reactor core (Fig. 1) (Popov and Kachur 2009).

The reactor model can be presented as an integrated model of particular FAs.

We shall assume that it is enough to know the following parameters of the current process in each channel to

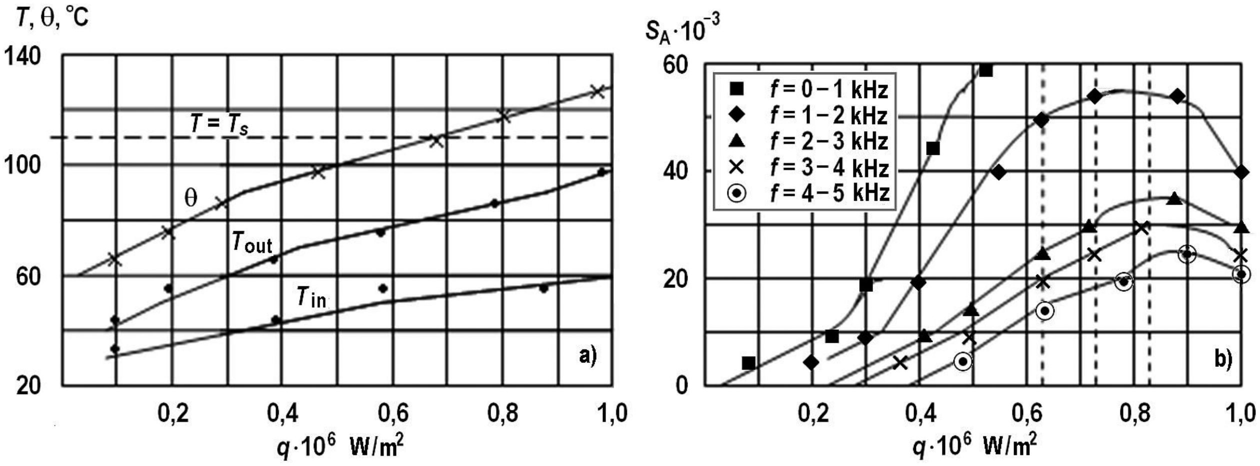

Figure 1. Experimental data describing the channel temperature mode (a) and the acoustic spectral characteristics (b)

diagnose and predict the state of the heat exchange process in the nuclear reactor core:

- channel outlet coolant temperature T;
- primary circuit pressure P;
- channel top fuel wall temperature θ;
- specific heat flux density q;
- pre-boiling temperature characteristics (Fig. 1a);
- acoustic spectral characteristics (Fig. 1b).

As the model base, we shall select the T-s-diagram (Novikov and Voskresenskiy 1977) as that reflecting, to the fullest extent possible, the thermodynamic process of energy conversion.

Stage 1 in the model development is to analyze the relationship between the system's specific entropy s and the experimental acoustic spectral characteristics. It can be seen from Fig. 1 that, in a frequency range of $f = 2 – 5$ kHz prior to the surface boiling onset ($q \leq 0.6 \cdot 10^6$ W/m^2) and subject to respective normalization, all characteristics have one slope that defines the coefficient value

$$B = \Delta q \cdot k_{\text{norm}} / \Delta A = \Delta q / \Delta T, \quad (1)$$

where ΔA is the variation of the acoustic noise fluctuation spectral density amplitude in response to the specific heat flux density variation Δq; ΔT is the channel outlet coolant temperature variation in response to the variation Δq; and k_{norm} is the normalization factor depending on the structural features of the particular reactor.

After simple transformations, the following relations can be obtained from the heat balance equation with regard for (1) and with the assumption that $\Delta G/G = \Delta\alpha/(1 - \alpha)$ (Kramer 1960)

$$\begin{aligned} \Delta q &= C \cdot G \cdot \Delta T \rightarrow \Delta q/\Delta T = C \cdot G \equiv B, \\ \Delta q &= \Delta G \cdot C \cdot T \rightarrow \Delta q/T = \Delta G \cdot C = \\ &= (\Delta G/G) \cdot G \cdot C = \Delta\alpha/(1 - \alpha) \cdot B, \end{aligned} \quad (2)$$

where q is the specific heat flux density; T is the channel outlet coolant temperature; G is the coolant mass flow rate; C is the specific heat capacity; α is the volumetric steam quality; and Δa is the steam quality variation leading to the coolant mass flow rate variation ΔG.

The specific entropy variation Δs is determined with regard for (2) as follows:

$$\Delta s = \Delta q/T = \Delta\alpha/(1 - \alpha) \cdot B. \quad (3)$$

It follows from relation (3) that the maximum specific entropy variation is achieved provided that $\Delta\alpha/(1 - \alpha) = 1$ and has a value of $\Delta s_{\max} = B$.

Stage II is to determine the surface boiling onset temperature T_{bo}. By analyzing Fig. 1a and taking into account the functional dependence of the boiling temperature on pressure, $T_s = f(P)$, we get

$$T_{\text{bo}} = T \text{ with } \theta = T_s. \quad (4)$$

Stage III in the model development is to plot the work line in the T-s-diagram. The work line is defined by the point of the boiling onset (point A, Fig. 2) and that of the transition to the supersaturated steam state (point D, Fig. 2). The point A is defined by the intersection of the T-s-diagram phase equilibrium curve with the line $T = T_{\text{bo}}$, and the point D is defined by the intersection of the T-s-diagram phase equilibrium curve with the line $s = s_{\text{bo}} + B$. With no external impacts, the variation of the heat exchange process parameters matches the straight line motion.

The work line equation has the form

$$T(s) = (s - s_{\text{bo}})(T_{\text{be}} - T_{\text{bo}})/(s_{\text{be}} - s_{\text{bo}}) + T_{\text{bo}}. \quad (5)$$

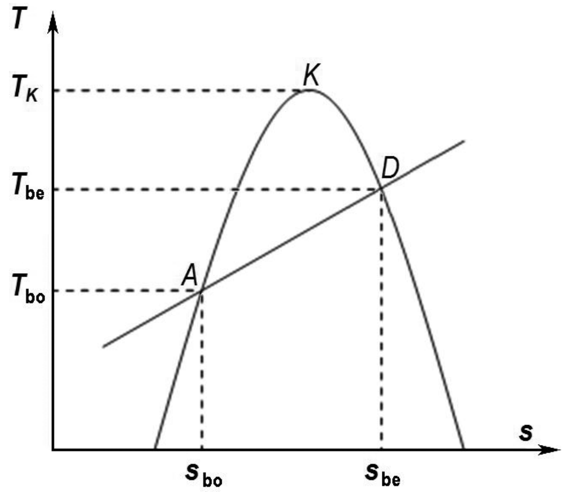

Figure 2. Work line plotting in the T-s-diagram

Stage IV is to predict the steam quality a during surface boiling ($T_{\text{bo}} < T < T_s$) in the event of the coolant temperature variation by ΔT. Using line equation (5) for $T = T_{\text{bo}} + \Delta T$, we determine $\Delta s = s - s_{\text{bo}}$. After simple transformations (4) and (5), assuming that $\Delta\alpha = \alpha - \alpha_{\text{bo}}$, the value a is calculated by the formula

$$\alpha = (\alpha_{\text{bo}} + \Delta s/B) / (1 + \Delta s/B). \quad (6)$$

The value α_{bo} is determined in accordance with the formula

$$\alpha_{\text{bo}} = 1.17 q^{0.35} / P^{0.15} (\rho w)^{0.15}, \quad (7)$$

where ρw is the mass velocity (Kirillov et al. 1990).

Stage V is to predict the steam quality in the event of the coolant temperature variation by ΔT. Using the relations for the surface boiling region ($x_0 < x < 0$, $T_{\text{bo}} < T < T_s$) in accordance with (Yemelyanov et al. 1975),

$$\alpha = \alpha_{\text{bo}} (1 - x/x_{\text{bo}})^{1.35}, \quad (8)$$

$$x_{\text{bo}} = -0.573 q^{0.7} (P/(\rho w))^{0.3}, \quad (9)$$

following simple transformations, we get

$$x = x_{\text{bo}} (1 - \alpha^{0.74}/\alpha_{\text{bo}}). \quad (10)$$

Stage VI is to predict the specific heat flux density q_{pred} during surface boiling ($T_{bo} < T < T_s$) in the event of the coolant temperature variation by ΔT. Based on the current information on the value q and T on the work line in the T-s-diagram, we determine $\Delta q = \Delta s\ (T + \Delta T)$ and calculate the predicted value

$$q_{pred} = q + \Delta s(T + \Delta T). \quad (11)$$

The presented model development procedure suggests monotonous variation of parameters.

Determination of the critical heat flux parameters using a mathematical model

The changes in the work line position for the point P, provided there are random external impacts capable to lead to the critical heat flux, are as follows:

- an abrupt reduction of the primary circuit pressure;
- an abrupt temperature increase (ΔT);
- an abrupt pressure (ΔP) and power (ΔQ) increase.

The physical meaning of the work line in the T-s-diagram can be defined as follows. For gases, the heat supply process can be nearly isothermic if consisting of alternating isobaric heat supply processes with a subsequent adiabatic expansion in a small interval of pressures (Fig. 3a) (Popov and Kachur 2009). The larger the number of such steps and the smaller the expansion in each of the steps are, the closer the process curve representing a saw-toothed curve is to an isotherm. For a reactor core, in the event of uncontrolled heat supply (with untimely steam and gas removal) in emergencies, the process may be presented as in Fig. 3b.

In accordance with the work line AD (Fig. 4), the point K that defines the critical temperature shifts to the point K' with the coordinates $s'_K = s_{bo} + \Delta s_{max}/2$ ($\Delta s_{max} = B$) and T'_K. To find the critical point C that defines the critical heat flux density q_{cr}, we shall identify two classes of the reactor core states in the event surface boiling starts:

- there is no critical heat flux;
- there is a critical heat flux.

We shall take entropy as the system state parameter.

Assuming that the distribution of the indicator s within the classes is described by normal law with the mathematical expectation s_{bo} for class 1 and s'_K for class 2 and with an equal dispersion of σ^2, Fig. 5 presents the functions of the entropy distribution density for these classes. The point s_{cr}, where the distribution density functions intersect, corresponds to the entropy of the system in a condition for which the probabilities of the critical heat flux taking place and being absent are equal, that is, defines q_{cr}.

The coordinates of the point $C(s_{cr}, T_{cr})$ on the work line (see Fig. 4), where

$$s_{cr} = (s_{bo} + s'_K)/2 = s_{bo} + B/4, \quad (12)$$

$\Delta s = B/4$, $q = q_{bo}$, $\Delta T = T_{cr} - T_{bo}$, make it possible to calculate a_{cr}, x_{cr}, and q_{cr} using formulas (6), (10), and (11).

The proposed model makes it possible to simplify to a great extent the calculation of such parameter as steam quality by substituting the iterative algorithm for its calculation by a sequence of several formulas. And the initial work line is plotted with a sufficient time to the boiling onset and until the need arises for monitoring the surface boiling process parameters.

Express diagnostics of the channel steaming state

It was assumed in the process of the mathematical model development that surface boiling was already taking place

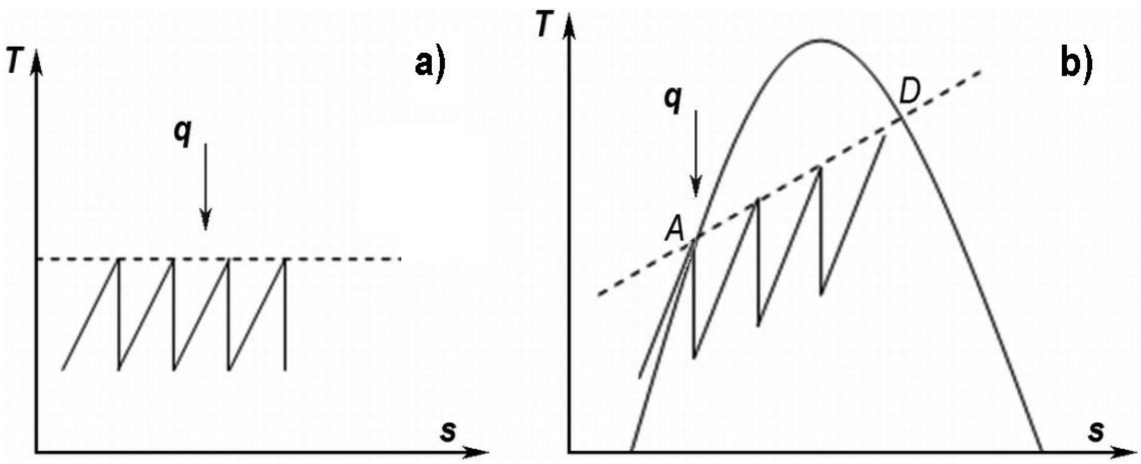

Figure 3. Thermodynamic process of heat supply for a heat engine or for a nuclear reactor in design conditions (a) and for a nuclear reactor in emergencies (b)

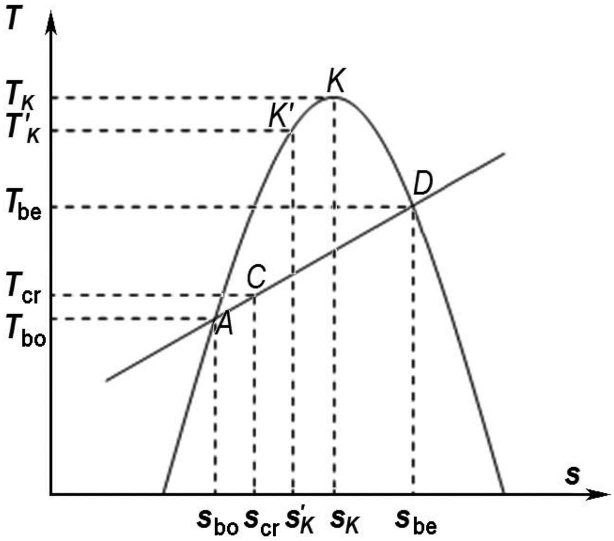

Figure 4. Plotting of the critical point C on the T-s-diagram work line

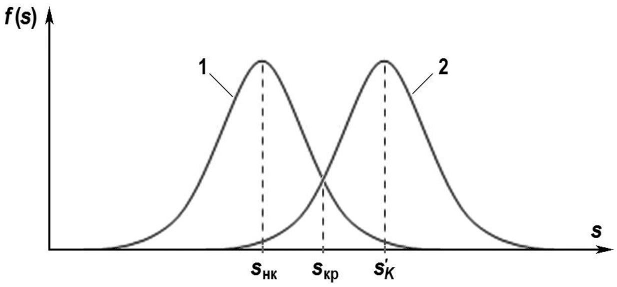

Figure 5. Functions of the entropy distribution density for two classes of situations: 1 – with no critical heat flux; 2 – with a critical heat flux

since the fuel wall temperature had reached the coolant boiling point. Therefore, the boiling onset temperature corresponds to the channel outlet coolant temperature when surface boiling occurs. In (Kachur 2009), the boiling temperature was determined through the steam quality calculation and the intersection point of the theoretical and experimental dependences was found.

The coolant flow is initially convective. The boiling onset temperature requires to be predicted and the work line equation built based on this prediction (5). The transition from the convective phase to surface boiling is possible in the event an accelerated power variation process is taking place.

The work line's maximum slope angle b will be defined by the second-order differences of the entropy with $\Delta s = B$:

$$\Delta T/\Delta^2 s = \Delta T^2/\Delta^2 q = \mathrm{tg}\ \beta, \tag{13}$$

where $\Delta^2 q$ is the second-order difference of q.

We shall assume that the T-s-diagram is described by the function F_{Ts}. Since the distance between the point of the work line intersection with the phase equilibrium curve of the T-s-diagram corresponds to B, then the boiling onset point can be determined from the relations

$$F_{Ts}(s_{bo} + B) - F_{Ts}(s_{bo}) = B \cdot \mathrm{tg}\ \beta, \tag{14}$$
$$T_{bo} = F_{Ts}(s_{bo}). \tag{15}$$

In the event F_{Ts} is given in a tabulated form, the boiling point coordinates are searched for by simple enumeration until condition (14) is fulfilled with the preset error. The rest of the parameters are determined in accordance with the proposed mathematical model.

A neural network with one perceptron is proposed to be used for the rapid identification of the channel state. The value s_{cr} calculated using formula (12) divides the work line into two portions (two subsets of points). By training the perceptron such that the values of the work line points before s_{cr} will correspond to the zero class, and those after s_{cr} will correspond to class 1, it is possible to classify the given vector of the values (s, T) as one of the two classes. Based on the classification results, a message is displayed on if the current channel state complies with the requirements.

Conclusion

The proposed mathematical model makes it possible to improve the operating safety of such a complex system as nuclear reactor by defining the boiling process as a principal manifestation of its operation. The model offers an opportunity to identify and predict in a timely manner an emergency caused by worsened heat removal from fuel thanks to using direct measurements of the heat exchange parameters, minimizing indirect calculations and employing empirical formulas. The model extends the class of the problems addressed, that is, makes it possible to proceed from the problem of identifying the nuclear reactor parameters and state to the problem of the critical heat flux prediction.

A possibility has been considered for rapid diagnostics of the channel state using neural network technologies.

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