LETTERS TO THE EDITOR

COMMENTS ON "OPERATING MARGIN OF SOVIET RBMK-1000 NUCLEAR POWER REACTORS"

The authors of Ref. 1 have arrived at a conclusion about the conservatism of the correlations taken to calculate the critical power of RBMK-1000 fuel channels, especially when the calculated local characteristics of fuel assemblies are involved. Reference 1 is an interesting paper, but it contains some inaccuracies and shows optimism that is, unfortunately, caused by a lack of information about the design and experimental work performed along these lines. Therefore, we feel that it might be useful to make some comments on this paper.

It is known that the operating margin to critical heat flux (CHF) for pressurized water reactors allows for the following:

1. the spread in the experimental data for the design correlation:

Correlation	Margin
W-3	1.3
B&W-2	1.3
GE-1	1.19
WRB-1	1.17

2. the anticipated operating transients, such as flow losses with four pumps running out (the margin is taken to be ~0.3)
3. the impracticality of assessing the operating condition of fuel assemblies, particularly the local parameters, by an operational monitoring system within the same error limits as those present in the derivation of the correlation (margin of ~0.3)
4. some uncertainty in the measurements of the core operating parameters (margin of ~0.2 to 0.3). These add up to a total margin of ~2 to 2.1.

It is also known that the margin to critical conditions for boiling water reactors (BWRs) is determined mostly in terms of power rather than by local parameters; that is, it involves the ratio of critical power to the maximum allowable operating power of the fuel assemblies. The operating margin to critical power for BWRs is chosen so that at least 99.9% of the fuel elements in the core should not reach a critical condition during most atypical operating transients.

For typical BWRs, the steady-state operating margin to critical power is taken to be ~1.2 to 1.3 with regard to the uncertainties in the correlations, a margin for the worst possible atypical operating transient conditions, and the uncertainties in the measurements of the core parameters.

The margin to critical power for RBMK reactors is calculated as follows:

$$\eta = \frac{N_c - 3(D_{N_c}N_c^2 + D_{N_{ch}}N_{ch}^2)^{1/2}}{N_{ch}} .$$

The channel power dispersion $D_{N_{ch}}$,

$$D_{N_{ch}} = \left(\frac{\epsilon_{MAIN}}{3}\right)^2 + \left(\frac{\epsilon_{DET}}{3}\right)^2 + \left(\frac{\epsilon_{MEAS}}{3}\right)^2 ,$$

is assessed with allowances for the maximum relative error in maintaining the reactor power by control rods ϵ_{MAIN}, the maximum relative error in determining the reactor power by the heat balance ϵ_{DET}, and the maximum relative error in measuring the channel power ϵ_{MEAS}. Channel critical power dispersion D_{N_c} is calculated in the form

$$D_{N_c} = \left(\frac{\epsilon_{N_c}}{3}\right)^2 + \frac{\left[\left(A\frac{\partial N_c}{\partial G_{ch}}\frac{\epsilon_{G_{ch}}}{3}\right)^2 + \left(B\frac{\partial N_c}{\partial P_{IN}}\frac{\epsilon_{P_{IN}}}{3}\right)^2 + \left(C\frac{\partial N_c}{\partial T_{IN}}\frac{\epsilon_{T_{IN}}}{3}\right)^2\right]}{N_c^2} ,$$

where

ϵ_N = maximum relative error in determining N_c

$\epsilon_{G_{ch}}$ = maximum relative error in measuring the channel flow

$\epsilon_{P_{IN}}$ = maximum relative error in measuring the channel inlet pressure

$\epsilon_{T_{IN}}$ = maximum relative error in measuring the coolant temperature at the channel inlet

A, B, C = upper scale values of the instruments for measuring flow, pressure, and temperature, respectively.

When speaking about the margin to critical RBMK power, it is usually meant that the coefficient η is equal to ~ 1.

If we turn to determining the margin to critical power for BWRs, i.e., to calculating the ratio of the critical fuel assembly power to its maximum allowable operating value, or equivalently, to assessing the expression

$$3(D_N N_c^2 + D_{N_{ch}} N_{ch}^2)^{1/2} ,$$

we get almost the same value as that for BWRs:

$$\frac{N_c}{N_{ch}} = 1.2 .$$

The correlation for the critical power of RBMK fuel assemblies, based on the averaged coolant parameters, was obtained from electrically heated fuel assembly models in the region of the corresponding regime parameters and was verified by in-pile tests.

The expression for N_c given in Ref. 1 has some mistakes. Its proper form is

$$N_c = \frac{4.28 \times 10^6 D_H^{0.83}(G_{ch} 10^{-3})^{0.57} + 4.07 D_H G_{ch} \vartheta_H}{664 D_H^{0.57}(G_{ch} 10^{-3})^{0.18} + 39.4L} F .$$

A few words should be said about using subchannel analysis techniques. Application of these techniques for the analysis of thermal and hydraulic parameters of fuel assemblies would be extremely useful, particularly at the design stage, although the empirical relations describing flow hydraulics in subchannels and the very correlations for calculating the critical conditions in subchannels call for thorough experimental verification. Besides, these techniques can undoubtedly be helpful in assessing the correctness of extending the model research results to real fuel assemblies, while there is usually no full similarity between them, as well as in studying the sensitivity of the critical margin to changes in the design parameters.

The local thermal-hydraulic performance of RBMK fuel assemblies was analyzed using the ЛУЧОК code. Calculations by this code for test experiments showed satisfactory agreement with those based on the COBRA code and are presented in Ref. 2.

Comparison of the power calculated by local characteristics with the experimental critical power for a 19-rod fuel assembly model shows a 10% spread in the results, depending on the relative inlet enthalpy.

The GE and other CHF correlations used for local subchannels offer more optimistic results, but it should be borne in mind that they were obtained for fuel assemblies with geometries that are different from that of RBMK assemblies. This is, perhaps, the reason for the optimistic judgments about some correlations in Ref. 1.

Therefore, the data from our analysis of local fuel assembly parameters do not allow us to be too optimistic about the margins for critical heat loads. The calculations were made using the CHF correlations best suited for describing the local subchannels of RBMK fuel assemblies.[3,4] The CHF margin in the form of the heat flux ratio in the subchannels with the highest heat flux amounted to no less than 1.5.

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RESPONSE TO "COMMENTS ON 'OPERATING MARGIN OF SOVIET RBMK-1000 NUCLEAR POWER REACTORS' "

It was with great interest that we read Ref. 1 regarding our recent paper.[2] We would like to apologize for the error in reporting the form of the RBMK critical power in our paper. We did, in fact, use the proper form of this correlation, as provided in Ref. 1, in our calculations; however, we inadvertently recorded an incorrect form during the drafting of our manuscript that continued into the final paper.

We were interested to learn that the critical power calculation for RBMK reactors proceeds by way of additional refinements to the channel critical power; that is to say, the channel critical power represents an input parameter to a higher level equation that includes other uncertainties not included in the correlation. We were unable to find any reference to this methodology in the sources that we had available to us. The calculations for the channel critical power that we did perform were in agreement with those reported by Dollezhal and Emel'yanov, who quoted critical heat flux (CHF) margins of 1 to 25%. These are also consistent with the 38% CHF margin calculated by the independent review of the RBMK design by the Central Electricity Generating