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Effect of the fuel element bundle statistical characteristics on the evaluation of temperature in the sodium-cooled fast-neutron reactor core

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Abstract

Different fuel element bundle models used to calculate the coolant and fuel cladding temperatures inside fuel assemblies have been analyzed as applied to sodium-cooled fast-neutron reactors. The drawbacks of the existing models have been identified. A bundle model based on an experimental study into the actual arrangement of the fuel elements within the AF shroud has been proposed. The model's capabilities and advantages, as compared to conservative models, have been shown with regard for the need to raise the reliability of the fuel cladding working temperature estimation.

Keywords: Fast-neutron reactors; Fuel assembly; Fuel element bundle models; Cladding temperature calculation; Statistics-based model; Interchannel exchange.

Introduction

The thermal-hydraulic mode of the sodium-cooled fast-neutron reactor core operation is designed in such manner that the in-service cladding temperature of the most heated fuel elements does not exceed the preset design value. This value depends on and is defined by the strength properties of the cladding material.

Very often, the desire to improve thermodynamic characteristics of the designs developed for power units with sodium-cooled fast-neutron reactors result in a coolant temperature increase at the core outlet and, consequently, in an increase in the fuel cladding working temperature. The considered temperature modes of the unit generally take place when the permissible temperature limits of the material serviceability are achieved.

Therefore, it is very important to optimize calculation techniques for determining the coolant and fuel cladding temperatures, so that to improve the reliability of the obtained results.

Normally, a detailed heat calculation is performed for the most heated core fuel assemblies and the side blanket. Inside the fuel assemblies, most of the fuel elements in the central zone have the highest temperature.

In engineering practice, the maximum temperature on the inner surface of the fuel element cladding, t_{clad}^{max}, can be found as a superposition of the coolant temperature at the reactor inlet, t_{in}, the coolant heating temperature averaged with respect to the channels surrounding the fuel, Δt_{cool}^{n}, lengthwise of the reactor core (including the bottom end shield) in the most heated FA area, temperature gradients at the cladding boundary, Δt_{α}, on the cladding periphery, Δt_{j}, and through the cladding thickness, Δt_{clad}; and the potential temperature increase, $\delta t_{act.oh}$, due to random deviations of the key parameters from the nominal values:

$$t_{clad}^{max} = t_{in} + \Delta t_{cool}^{n} + \Delta t_{\alpha} + \Delta t_{j} + \Delta t_{clad} + \delta t_{act.oh} \quad (1)$$

The temperature differences in expression (1) are the functions of a large number of design and operating parameters of the given FA and the core as the whole. No actual values of all these parameters under reactor conditions are known.

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It is only their nominal values, x_n, and the potential maximum nominal deviations from the nominal values, $\pm\ \delta_x$, that have been specified. In the reactor thermal calculations, parameter deviations are commonly described by coefficients referred to as overheating factors. Overheating factor F_i is a random value describing the maximum relative deviation from the nominal value of the parameter x_i that defines a temperature or a temperature gradient: $F_i = \delta x_i / x_n$. Deviations from their nominal values of all random components in expression (1) are summed up, leading to the overall overheating of the fuel cladding, $\delta t_{act.oh}$.

In fast-neutron reactors, the coolant temperature plays a key role in the temperature field formation in the fuel cladding. As compared to sodium heating lengthwise of the core (230–250 °C), the temperature differences from the coolant to the inner cladding surface are small, and the total thereof does not exceed 30–40 °C. At the same time, it is exactly the calculation of the maximum nominal heating inside of the FA, $\Delta t_{cool}{}^{n}$, as well as the estimation of its maximum possible deviation $\delta(\Delta t_{cool})$ from the nominal value under the influence of random values that causes the greatest ambiguity and introduces most of the uncertainty into the total error in the fuel cladding calculations, as expressed by (1).

In fact, the nominal and the mixed average heating of sodium in an FA are easily calculated using the specified FA power values and the flow through it. However, the distribution of the local flow rates and, respectively, the heating temperatures in the channels across the fuel bundle fully depends on how these fuel elements are arranged within the FA shroud. This is the main focus of this study.

Fuel element bundle calculation models

Generally, fuel assemblies of fast-neutron reactors are designed in the form of a hexagonal shroud with a bottom nozzle and a top nozzle welded to its ends. The fuel elements are secured in the bottom portion of the shroud and arranged in a triangular lattice. Most commonly, fuel elements are spaced by a round wire spirally wound on the shroud.

Each FA has a minimum positive manufacturing clearance (the so-called theoretical fitting gap), $2\delta_{th}$, between the tightly compacted fuel element bundle and the opposing shroud walls. Its smallest possible value can be estimated based on the total of the positive tolerances for the fuel claddings within the compacted bundle and of the negative tolerance for the inner dimension of the shroud. The said gap is required to enable the FA assembly.

To ensure a more uniform temperature distribution around the periphery of the fuel cladding adjacent to the shroud, displacers of a round wire may be introduced into the wall-adjacent channels. It is for the same purpose of spacing the peripheral fuel elements that an elliptic half-width strap is used, which reduces the dimensions of the wall-adjacent channels.

A thermal-hydraulic calculation of an FA requires the flow areas of the channels surrounding the fuel element to be known. The presence of the fitting gap introduces an uncertainty into the calculated inner geometry of the bundle.

Table 1
Estimated nominal coolant heating values in the FAs of the BN-800 reactor found using different fuel bundle models.

	Bundle model			
	1 Spaced	2 With distributed gaps	3 Tightly compacted	4 Statistical
t_{in} (°C)			354	
$\Delta t_{FA}{}^{av}$ = °C			232	
Δt_{cool} (°C)	246	252	283	266
$\Delta t_{cool}/\Delta t_{FA}{}^{av}$	1.06	1.086	1.22	1.146

In fast reactor design, a variety of hypotheses are used concerning the arrangement of the fuel elements within the FA in the limits of the allowable fuel element displacements. To a greater or smaller extent, these reflect the probable reality, and change depending on the design maturity and the optimization level of the reactor heat-engineering parameters:

- *a spaced bundle model*: the wall-adjacent fuel elements contact the shroud through the spacing strap, while the others are uniformly distributed across the section in the regular triangle lattice nodes, thus forming the channels to be calculated with the nominal area Ω_{cal} [1,2];
- *a model with distributed gaps* in which the fitting gap (starting from the shroud walls) is uniformly distributed across the bundle. The fuel elements are in the regular triangular lattice nodes; the nominal area of the channels to be calculated, Ω_{cal}, is a bit smaller than the previous one [3,4];
- *a tightly compacted bundle model* with the axial arrangement of fuel elements lengthwise of the fuel bundle shroud tube. The fuel elements are so tightly compacted that the spacing wire contacts the neighboring fuel element cladding. The area of the channels to be calculated, $\Omega_{cal}{}^{min}$, is the smallest possible [5,6];
- *a model with axially variable gaps* which assumes that there is a fuel bundle with distributed gaps in the lower part of the fuel elements before the core. Lengthwise of the core, the mean pitch decreases gradually due to the fuel thermal deflection until a tightly compacted bundle is formed in the core's upper plane. And the bundle retains the regular triangular shape of its pin lattice, which ensures that the channels are of the same size in any FA cross-section.

Since the channel size variation along the core is very small (0.6% per 10 cm), and the channels are identical in the FA cross-sections, there is no axial interchannel heat and mass exchange in the bundle. The bundle becomes practically equivalent to the tightly compacted bundle and is not discussed herein.

Assuming that the channels are isolated (which makes the calculation conservative), and the average coolant heating in the FAs ($\Delta t_{FA}{}^{av}$) is constant, Table 1 (columns 1 through 3) presents estimated nominal coolant heating values, $\Delta t_{cool}{}^{n}$, in