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.

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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, $t_{\text{clad,max}}$, can be found as a superposition of the coolant temperature at the reactor inlet, $t_{\text{in}}$, the coolant heating temperature averaged with respect to the channels surrounding the fuel, $\Delta t_{\text{cool,n}}$, lengthwise of the reactor core (including the bottom end shield) in the most heated FA area, temperature gradients at the cladding boundary, $\Delta t_{\text{f,d}}$, on the cladding periphery, $\Delta t_{\text{f,p}}$, and through the cladding thickness, $\Delta t_{\text{clad}}$, and the potential temperature increase, $\delta t_{\text{act,oh}}$, due to random deviations of the key parameters from the nominal values:

$$t_{\text{clad,max}} = t_{\text{in}} + \Delta t_{\text{cool,n}} + \Delta t_{\text{f,d}} + \Delta t_{\text{f,p}} + \Delta t_{\text{clad}} + \delta t_{\text{act,oh}}$$

(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.
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, $\delta I_{fact,ob}$. 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.

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 $\Omega_{cal}$ [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, $\Omega_{cal}$, 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 ($\Delta t_{FA}^{*}$) is constant, Table 1 (columns 1 through 3) presents estimated nominal coolant heating values, $\Delta t_{cool}^n$, in

<table>
<thead>
<tr>
<th>Bundle model</th>
<th>Spaced</th>
<th>With distributed gaps</th>
<th>Tightly compacted</th>
<th>Statistical</th>
</tr>
</thead>
<tbody>
<tr>
<td>$t_{in}$ (°C)</td>
<td>354</td>
<td>232</td>
<td>283</td>
<td>266</td>
</tr>
<tr>
<td>$\Delta t_{FA}^{(*)}$</td>
<td>246</td>
<td>252</td>
<td>1.06</td>
<td>1.086</td>
</tr>
<tr>
<td>$\Delta t_{cool}^n$</td>
<td>1.06</td>
<td>1.086</td>
<td>1.22</td>
<td>1.146</td>
</tr>
</tbody>
</table>
triangular FA channels of the BN-800 reactor for the above bundle models.

As can be seen from the table, the heating difference and, therefore the nominal cladding temperature difference is 37 °C for the two extreme cases (a spaced bundle and a tightly compacted bundle). This value is very great if the design-basis fuel element operation mode is close to the permissible temperature level. What is meant here are the fuel elements in the FA central zone surrounded by triangular channels in which, in general, the greatest possible coolant heating is observed. It is also assumed that there is no radial power density drop in the FA.

Apart from the nominal temperature calculation, problems arise with the above calculation of the potential cladding overheating under the influence of random factors, primarily, thermal-hydraulic ones. The problems are caused by the uncertainty in the limits of the potential disruption of the regular lattice with the fuel element bundles, leading to a change in the design coolant flow modes, especially in the wall-adjacent FA channels.

All of the above models are inherently static because they do not involve random scatter of the channel dimensions against a certain average value. And the natural desire to somehow use the initial fitting gap (especially, in the tightly compressed model) may cause one to wish to move arbitrarily some of the fuel elements in the limits of possible displacements, leading so to a variation in the design dimensions of the channels (as one of the overheating factors). These shifts are highly subjective and are in no way justified.

In the existing calculation techniques [6–8], an overheating induced by random factors is proportional to the temperature or the temperature difference it relates to. Therefore, in accordance with expression (1), all overheatings are proportional to the differences taking place at the core outlet, also including the maximum nominal coolant heating. Under otherwise equal conditions, the greater is the heating, the greater is the overheating.

With regard for the above, it should be noted that the static fuel bundle models do not allow taking fully into account actual effects from the interchannel heat and mass transfer (HMT) on the coolant temperature leveling lengthwise of the core due to similar channel dimensions and thermal-hydraulic characteristics (by hypothesis). In a regular lattice, this process will manifest itself effectively only on the FA periphery, where there are channels of different types, leveling off so the temperature gradient in two or three fuel element rows, but will not extend its influence further towards the center [9].

Statistics-based fuel bundle model

The existing contradictions are eliminated to a great extent by the IPPE-developed fuel bundle model in which the flow areas of the channels are calculated based on results of an experimental study into the fuel element arrangement within the shrouds of non-irradiated fuel assemblies [9, 10].

To build a statistics-based fuel bundle model (the statistical model hereinafter), dummy FAs of the BN-350 and BN-600 fast-neutron reactors were used. The dimensions of the fuel element cladding and the hexagonal shroud were measured for each FA. The fitting gaps in the FAs were different, up to the maximum value $2\delta_{th} = 2.9$ mm.

The FA shroud was filled with epoxide compound to avoid fuel element displacements, and then it was cut across into many sections. In each section, the mutual locations of all fuel elements in the bundle were measured, and the flow area values, $\Omega_i$, were calculated for each channel.

Studies have shown that the channel size formation in the course of the FA fabrication is a random process. Those fuel elements having a certain freedom of radial movement under the influence of some factors bend and move in the limits of the existing gap, failing to form so an ideal pin lattice. These displacements have led to the channel flow area dimensions distributing axially and radially inside the assembly in accordance with some statistical laws.

The processing of experimental data revealed a correlation between the theoretical fitting gap $\delta_{th}$ and the actual (statistically average) fitting gap $\delta_{act}$ formed in some FAs. Assuming that the gaps in the bundle have resulted from elastic bending deformations of the fuel, the relation between $\delta_{act}$ and $\delta_{th}$ was approximated by the linear dependence

$$\delta_{act} = 0.11 + 0.40\delta_{th},$$

which is true for the investigated range of $0.22 \text{ mm} \leq \delta_{th} \leq 1.45 \text{ mm}$. This invariably spans the fitting gap variation range for the standard FAs of the BN-600 and BN-800 reactors ($0.27 \text{ mm} \leq \delta_{th} \leq 1.02 \text{ mm}$).

A real (experiment-based) model of the bundle makes it possible to identify three channel types: triangular channels in the central region, triangular channels in the closing row (these contact the wall-adjacent fuel elements), and wall-adjacent rectangular channels (Fig. 1).

The computational model uses the channel flow area values averaged axially over the core, $\Omega_i^{av}$. The mathematical expectation $M(\Omega_i^{av})$ for the set of the cross-sections of the channels of each type in the bundle (universal mean) is denoted as $\Omega_i^{av\text{ef}}$.

Representation of the experimental results in the form of deviation sweeps, $\Delta\Omega_i^{av}$, as the difference of the $i$th channel

\begin{figure}[h]
\centering
\includegraphics[width=\textwidth]{Fig_1.png}
\caption{Typical channels in the cross-section of the BN-type reactor FAs: 1 – triangular channels in the central region; 2 – triangular channels in the closing row contacting the wall-adjacent fuel elements and 3 – wall-adjacent rectangular channels.}
\end{figure}
Плотность вероятности, \( \varphi(x) = \text{Probability density} \), \( \varphi(x) \)

Fig. 2. Probability density function of the Weibull distribution.

<table>
<thead>
<tr>
<th>Channel type</th>
<th>Calculation formula</th>
</tr>
</thead>
<tbody>
<tr>
<td>Triangular, central region</td>
<td>( \Omega^{\text{av}} = (0.212(\text{act}) + 0.986)\Omega^{\text{cal}}_{\text{min}} )</td>
</tr>
<tr>
<td>Triangular, closing row</td>
<td>( \Omega^{\text{av}} = (0.314(\text{act}) + 0.964)\Omega^{\text{cal}}_{\text{min}} )</td>
</tr>
<tr>
<td>Rectangular, wall-adjacent</td>
<td>( \Omega^{\text{av}} = (0.812(\text{act}) + 0.958)\Omega^{\text{cal}}_{\text{min}} )</td>
</tr>
</tbody>
</table>

To calculate the average flow areas for channels of different types, the following relationships have been obtained by approximation from the Weibull distribution for two types of the relative channel characteristics, \( \Delta \Omega^{\text{rel}} \):

\[ \Delta \Omega^{\text{rel}} = \frac{\Omega^{\text{av}} - \Omega^{\text{av min}}}{W_{\text{cal}}^{\text{min}}} \quad (5) \]

\[ \Delta \Omega^{\text{rel}} = \frac{\Omega^{\text{av}} - \Omega^{\text{av min}}}{W_{\text{cal}}^{\text{min}}} \quad (6) \]

Processing of statistical data (5) for all channel types made it possible to obtain the linear dependences of the average flow area, \( \Omega^{\text{av}} \), on the fitting gap value, \( \delta_{\text{act}} \), in a dimensional form, which are presented in Table 2 (gaps in mm, channel flow areas in mm\(^2\)). The dependences are valid in a range of \( 0.20 \leq \delta_{\text{act}} \leq 0.70 \text{ mm} \) and have a scatter band of \( \pm (1.0-1.9)\% \) for the central and the wall-adjacent regions respectively, with

a confidential probability of 95%. The values of the constant coefficients have been obtained based on the approximation of experimental data.

The processing of statistical data (6) for each channel type revealed the dependence between the average channel flow area, \( \Omega^{\text{av}} \), and the Weibull distribution mathematical expectation:

\[ \Omega^{\text{av}} = \Omega^{\text{av min}} + M_{\text{s}} \Omega^{\text{cal min}} \quad (7) \]

where \( M_{\text{s}} = M(\Delta \Omega^{\text{rel}}) \) is the mathematical expectation of the distribution.

Besides, all basic characteristics of the Weibull distribution (parameters \( a, b \), and \( M_{\text{s}} \), and the mean-root-square deviation \( \sigma_{\text{s}} \)) were expressed through the actual fitting gap value \( \delta_{\text{act}} \) to calculate the probability density for the dimensions of each channel type. The relationships of the components in expression (7) are presented in Fig. 3.

The regularities obtained for the statistics-based fuel bundle model made it possible to present the fuel assembly as a set of three channel types in two ways:

- with the average flow area \( \Omega^{\text{av}} \) being constant for each group (Table 2);
- with flow areas in each group (the value \( \Omega^{\text{av}} \) being average) varying in a range from the smallest dimension \( \Omega^{\text{av min}} \) to the largest dimension \( \Omega^{\text{av max}} \) according to the Weibull distribution with their own distribution parameters. However, no location of the particular channel in a group is specified.
Option 1 (statistically average) (see Table 2) is basically similar to the model with distributed gaps, but the average channel dimensions have been adjusted in accordance with the gap value on the FA periphery. It makes it possible to calculate correctly the nominal coolant heating values $\Delta t_{\text{cool}}^n$ and, accordingly, the fuel cladding temperatures at all design stages. A random deviation of the heating temperature from the nominal value due to the dispersion of the channel dimensions can be taken into account using the channel dimension scatter in a range from the average value $\Omega_{\text{av}}^\text{ref}$ to the minimum value $\Omega_{\text{av}}^\text{min}$ as one of the overheating factors.

Option 2 gives the probabilistic distribution of the channels by dimensions (Fig. 3), making it possible to determine the number of the channels falling within any given interval of the flow area values. Such way to present the fuel bundle allows calculate, apart from the nominal heating temperature ($\Delta t_{\text{cool}}^n$), its scatter through the dispersion of the channel dimensions, with regard for the interchannel heat and mass transfer taking place during the longitudinal motion of sodium in the bundle. This process smooths to a great extent the differences in the heating temperatures of individual channels, while reducing the core outlet coolant temperature irregularity inside the FAs, as compared to the standard calculation (see option 1) of the sodium overheating under the influence of the channel dimension dispersion. The comparison results will be given below.

Since the location of each particular channel in the bundle is arbitrary, the model implementation consists in the repeated structural design of a fuel assembly based on the Monte-Carlo method using the predetermined laws of the channel dimension distribution.

The random dimension of the channel flow area is written as:

$$\Omega_{\text{av}} = \Omega_{\text{av}}^\text{min} + x \cdot \Omega_{\text{cal}}^\text{min},$$

where $x = \Delta \Omega_{\text{rel}}$ is the random value distributed according to the Weibull distribution (4). Having transformed $x$, using the relation

$$x = -a[\ln(1 - y)]^{1/b},$$

into a random value $y$ distributed uniformly in the interval (0; 1), we find the random dimension of the channel’s flow area:

$$\Omega_{\text{av}} = \Omega_{\text{av}}^\text{min} + \Omega_{\text{cal}}^\text{min}a[\ln(1 - y)]^{1/b}.$$  

(10)

Values $y$ are obtained using a random number generator, and the flow area values obtained from expression (10) are used to fill in a line-by-line manner the whole of the fuel assembly region occupied by channels of this type. Then, thermal-hydraulic calculation for the FA is performed. Multiple repetitions of the operation make it possible to obtain the key statistical data of the coolant temperature field for any channel, namely the mathematical expectation $M(\Delta t_{\text{cool}})$ and the mean-root-square deviation $\sigma(\Delta t_{\text{cool}})$ used further to calculate the fuel cladding temperature with regard for the overheating factors.

Both options of the statistical model give one and the same value of the average coolant heating $M(\Delta t_{\text{cool}}) = \Delta t_{\text{cool}}^n$ in the channels of the group under consideration (Table 1, column 4), however option 2 demonstrates the heating scatter in accordance with the dispersion of the flow area dimensions.

The statistical nature of the channel dimension distribution in the bundle was confirmed by analyzing the cross-sections of an FA from the French reactor Phenix irradiated to a burn-up of 6.6% of heavy atoms [11]. Histograms of the triangular channel areas in the base (not swelled) cross-section of an FA from the Phenix reactor and in one of the cross-sections of the BN-600 reactor FA dummy were compared with the same number of partitioning intervals and with the mathematical expectations integrated. The comparison results shown in Fig. 4 (no particular channel dimensions are specified) agree well in terms of the distribution nature. The histogram for the swelled part of the Phenix FA bundle demonstrates a minor decrease in the mathematical expectation and an approximately twofold increase in the channel area scatter.

Besides, the tightly compacted model is not confirmed by the direct measurements of the coolant temperature in the channels of the instrumented Tetacouple FA in the Rapsodie reactor [12, 13].

Now we shall consider the application of the statistical model in the thermal-hydraulic calculations of the fast reactor FA fuel elements. When discussing the results, the statistical bundle will be compared against the tightly compacted bundle as the most conservative model.

**Capabilities of the statistics-based bundle model**

- As compared to the tightly compacted bundle, using the statistically average model for an FA with the fast-reactor
Плотность вероятности = Probability density

Подогрев теплоносителя = Coolant heating

Fig. 5. Distribution of the coolant heating in the central channels at an accidental flow area variation calculated with no interchannel transfer (the analytic curve and the histogram obtained by Monte-Carlo method) taken into account. \( M_1 = \Delta t_{\text{cool}}^n = 266 \, ^\circ \text{C} \).

(e.g. BN-800) parameters leads to the temperature field leveled off across the fuel assembly \((\Delta t_{\text{cool}}^n/\Delta P_{\text{nom}}^n \approx 1.15)\), as well as to a decrease in the coolant heating temperature \( \Delta t_{\text{cool}}^n \), and, accordingly, in the calculated nominal temperature of the central fuel elements by approximately 17 °C (Table 1, columns 3 and 4).

When the bundle model option built by the Monte-Carlo method is used, taking into account the interchannel heat and mass transfer reduces the accidental coolant overheating caused by the channel flow area dispersion, from \( \delta(\Delta t_{\text{cool}}) \approx 18 \, ^\circ \text{C} \) (Fig. 5) to \( \delta(\Delta t_{\text{cool}}) \approx 4-5 \, ^\circ \text{C} \) (Fig. 6).

Therefore, as compared to the option with isolated channels, the overheating decreases by approximately 13 to 14 °C.

It should be noted that the inter-channel heat and mass transfer levels off the heating scatter inside the FA and from the power density dispersion caused, for example, by the fuel load scatter into the fuel elements. This effect can be taken into account in the same way as the effect from the channel dimension scatter.

The obtained results make it possible to consider the coolant heating scatter in the statistical bundle as a new random variable with its own distribution law and scatter characteristics. With regard for the interchannel transfer, the overheating value \( \delta(\Delta t_{\text{cool}}) \), relative to the rated value, caused by the channel dimension dispersion, can be used in the calculation of the total coolant heating deviation from the random values as the overheating factor \( F_{\Delta t_{\text{cool}}} = \delta(\Delta t_{\text{cool}})/\Delta t_{\text{cool}}^n \).

And the factor allowing for the channel flow area dispersion is not used in the calculated dependences.

Частота = Frequency

Подогрев теплоносителя = Coolant heating

Fig. 6. Distribution of the coolant heating in the central channels at an accidental flow area variation calculated with no interchannel transfer (the histogram obtained by Monte-Carlo method) taken into account. \( M_1 = \Delta t_{\text{cool}}^n = 266 \, ^\circ \text{C} \).

Conclusion

The use of the calculation model based on an experimental study into the statistical distribution of one of the key hydraulic characteristics of an FA, namely the flow area of the channel in a fuel element bundle, and, as a consequence, the possibility for taking into account the interchannel heat and mass transfer lengthwise of the fuel element, offers an effective technique to increase the reliability of determining the working temperature of the fuel cladding material.

References


