ENERGETICS

ЕНЕРГЕТИКА

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MODEL FOR CALCULATING THE TEMPERATURE IN THE FUEL RODS OF THE FA-X FUEL ASSEMBLY PRODUCED FOR SUBCRITICAL INSTILLATION AND REACTOR WWR-M

І.О. Чернов А.В. Куштим. **Модель для розрахунку температури в стрижневому твелі паливної збірки ТВЗ-Х при використанні в підкритичній установці та реакторі ВВР-М.** Наведено опис моделі стрижневого твела паливної збірки ТВЗ-Х, спроектованої в НТК ЯПЦ ННЦ ХФТІ, як альтернативного палива для ядерної установки, заснованої на підкритичній збірці (ПКУ, ННЦ ХФТІ, м. Харків) і дослідницького реактора (ВВР-М, ІЯД, м Київ). Модель являє собою програму, яка дозволяє проводити розрахунок розподілу температур по радіусу і висоті стрижневого тепловидільного елемента, що містить як таблеткове оксидне паливо UO_2, так і паливо на основі дисперсійної композиції UO_2+Al з різним вмістом паливної фази в матриці, а також різних геометричних характеристик твелів і значеннях параметрів теплоносія: температури на вході в гідравлічний канал твела і його швидкості. Проведені порівняльні розрахунки розподілу температур при експлуатації. В результаті чого показано, що для умов роботи в ПКУ (лінійна потужність твела 2,62 кВт/м) температура в центральній частині паливного сердечника становить ~140 °С для UO_2 і ~112 °С для композиції UO_2+Al. Для умов експлуатації в дослідницькому реакторі ВВР-М (лінійна потужність твела 12,1 кВт/м) температура в центральній частині паливного сердечника досягає ~626 °С для UO_2 і ~381 °С для металокерамічного (UO_2+Al) палива. Розрахунками показано значний вплив типу матеріалу паливного сердечника (UO_2 або дисперсійної композиції UO_2+Al) на температуру в його центрі з урахуванням умов роботи в ПКУ і дослідницькому реакторі ВВР-М. Максимальна температура поверхні оболонки твела для умов роботи реактора ВВР-М склала 86,5 °С, а максимальна температура поверхні оболонки для умов роботи Підкритичної установки становить 27 °С, що не перевищує температуру кипіння (пароутворення) при номінальних умовах експлуатації. Розрахована площа прохідного перетину твелів, коефіцієнт тепловіддачі та розподіл температур теплоносія. Програмний модуль дозволив проводити оцінку розподілу температур твелів з різним типом ядерного палива для умов дослідницьких ядерних установок.
Ключові слова: стрижневі твели, модель, теплопровідність, діоксид урану, дисперсійне паливо, об'ємна частка паливної фази

I. Chernov, A. Kushtym. **Model for calculating the temperature in the fuel rods of the FA-X fuel assembly produced for subcritical instillation and reactor WWR-M**. The TVS-X fuel rod model designed by NSC KIPT as an alternative fuel for subcritical assembly (SCA, KIPT, Kharkov) and research reactor (WWR-M, INR, Kiev) is described. The model is a program that allows calculating the temperature distribution on the radius and height of the fuel element containing both uranium oxide pellets and dispersion fuel based on the UO_2+Al composition with different contents of the fuel phase, as well as the different geometric characteristics of the fuel element and the values of the coolant parameters: the temperature at the entrance to the hydraulic channel and the coolant speed. Comparative calculations of temperature distribution during operation are carried out. As a result, it has been shown that for conditions of operation in the SCA (linear power of fuel rod is 2.62 kW/m), the fuel center temperature reaches ~140 °C for UO_2 and ~112 °C for the UO_2+Al composition. For operating conditions in the WWR-M reactor (linear power of fuel rod is 12.1 kW/m), the fuel center temperature reaches ~626 °C for ceramic (UO_2) and ~381 °C for metal-ceramic fuel (UO_2+Al). The calculations show a significant effect of the type of fuel material (UO_2 or UO_2+Al dispersion composition) on the fuel center temperature, taking into account the operating conditions in the subcritical assembly and the WWR-M research reactor. The maximum temperature of the cladding for the WWR-M operating conditions was 86.5 °C, and the maximum temperature of the cladding for the SCA operating conditions is 27 °C, which does not exceed the boiling point (vaporization) under the nominal conditions of their operation. Cross-section area of fuel rods, heat transfer coefficient and temperature distribution of the coolant are calculated. The software module allowed to estimate the temperature distribution of fuel element with different types of nuclear fuel for the conditions of research nuclear assemblies.
Keywords: fuel rods, model, thermal conductivity, uranium oxide, dispersion fuel, volume fraction of the fuel phase

Introduction

Evaluation of temperature distribution in the fuel rod is the most important factor that affects many internal processes and determines overall reliability and performance of the fuel rod.

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During the design stages, preliminary evaluation of the axial and radial temperature distribution is required for the developed container and coupled fuel rods. At the starting design stages in the developed container and coupling fuel rods (FR) performance, a preliminary assessment of the temperature distribution across height and radius of the fuel rods is required.

A similar problem was also considered by many authors who used both numerical and analytical modeling of the temperature distribution in nuclear fuel [1 – 5].

Theoretical basis for the thermal analysis of fuel rods in pressurized water reactors (WWER, PWR) and description of the finite element method for solving the heat transfer equation by HT_Frod computer program are given in [6].

The main purpose of this work is to develop a simple model for the FA-Kh fuel rod, designed for calculation of stationary distribution of the fuel stack axial and radial temperatures with variation of such parameters:

– geometry of the cladding and fuel stack;

– fuel material type (uranium dioxide – UO_2 and dispersion UO_2+Al fuel compositions with different fuel content);

– inlet coolant temperature and flow rate.

Also, the aim of the work is comparative temperature analysis for fuel rods with different fuel types (UO_2 and UO_2+Al) for the subcritical assembly (KIPT, Kharkov) and WWR-M research reactor (INR, Kiev) operating conditions.

1. Analysis of publications and statement of the problem

Analysis of publications on this issue have shown that there are several ways of solving the problem of evaluation the heat transfer in fuel rods of nuclear reactors. The first way is based on the development of solid-state models of fuel rods with the use of powerful modern complexes of the 3D design. So in [7] was carried out numerical simulation of fuel assemblies at the supercritical parameters of the coolant of the WWER-SCP, developed in OKB “Gidropress” using Design Molder program complex that is included into ANSYS. As a result of the simulation, the temperature distribution in the fuel assembly and the speed of the coolant were obtained. The paper [5] presents the results of calculations of temperature distribution of fuel rods of the reactor PIK which have a cross-shaped profile and length of 500 mm and self-distancing heterogeneous fuel core based on UO_2+Al. For this fuel composition was used the formula of thermal conductivity coefficient, including the value of the fuel and matrix components and their individual values of conductivity. Calculation of heat transfer in the fuel rods with such complex geometry was carried out using the program COSMOS/M. The second way of evaluating the heat engineering state of the fuel rods of nuclear reactors is based both on quantitative and analytical modeling of temperature distribution using mathematical programs such as MathCAD and MathLAB. In [2] it is shown a comparison of calculations of a heat generation and temperature distributions in the fuel based on UO_2 pellets for fuel rods of the WWER-1000 reactor. It is concluded that the analytical method is the most effective for small fluctuations in the change of the thermophysical parameters and at transients it is more efficient to use a numerical method.

However, solving problems of conceptual design of fuel rods and the estimation of the temperature using the complexes of the 3D modeling and mathematical packages are not always appropriate by the reason of their significant cost.

The aim of this work is to develop a simple model (a computer program) to calculate the stationary distribution of the temperature across the radius and the height of the fuel pellet column in the fuel rods with a different type of fuel (UO_2 and UO_2+Al) and with the possibility of variation of parameters among which are the geometric dimensions of the cladding and fuel mandrel, the temperature and the coolant flow rate at the inlet to the fuel assembly.

2. Description of the FA-Kh fuel rod, calculation model and program algorithm

FA-Kh [8] is designed to be compatible and interchangeable with the basic WWR-M2 FA according to the main performance characteristics: neutronic and thermal-hydraulic and overall and connecting dimensions. The cross section of the FA-Kh is maintained the same as in WWR-M2. In contrast to the fuel rods of the basic variant – WWR-M2 FA, which consists of three tubular fuel rods with a dispersion fuel composition in the form of UO_2+Al, the developed at FA-Kh includes 6 fuel rods. These fuel rods are based on uranium dioxide pellets placed in the cladding with a diameter of 9.1×0.7 mm, and a length of ~ 565 mm, made from zirconium alloy E110 (Fig. 1).

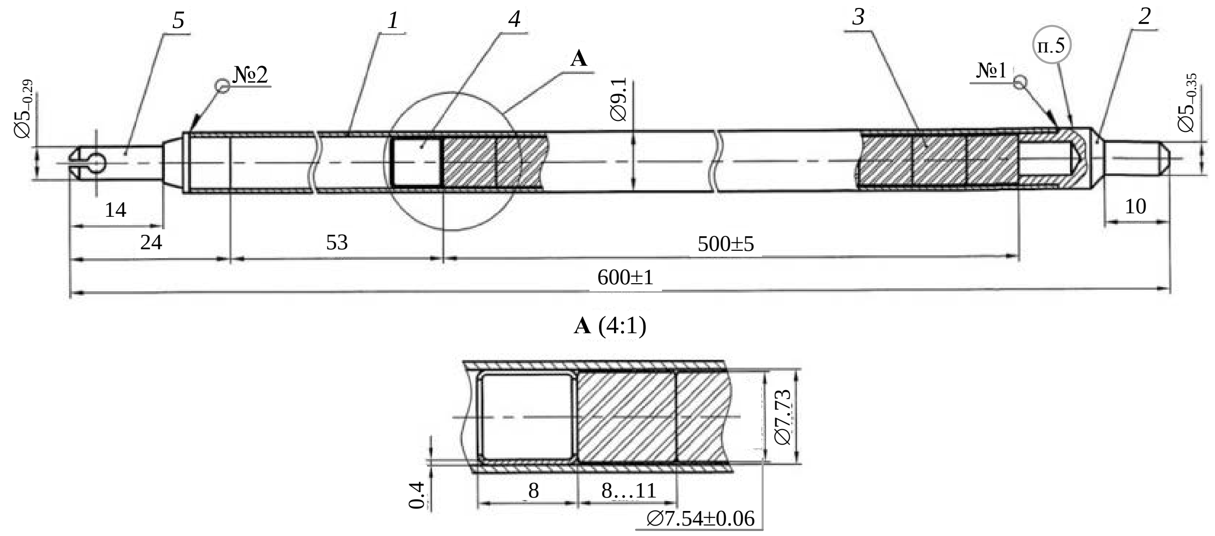

Fig. 1. FA-Kh fuel rod and pellet design

The cladding is sealed with lower plug and upper plug. Inside the cladding there are fuel pellets with a height of 500 mm. UO_2 pellets have a diameter of 7.54±0.06 mm and a height of 8…11 mm. In the upper part, the fuel stack is fixed by retainer 5. The free volume inside the cladding is filled with helium with a pressure of 0.1 MPa. The total length of the fuel rod is 600 mm. As fuel, both uranium dioxide and dispersion composition UO_2+Al with different content of the fuel phase can be used.

The fuel rod model is a program written in C language, which allows calculating temperatures along the radius and height of the fuel stack taking into account various types of fuel material, as well as various geometric characteristics of fuel elements and coolant parameters: inlet temperature, hydraulic channel and coolant velocity.

The main input data of the model given in the text file InputData.txt, are:

– geometry characteristics of the cladding and pellets;

– fuel rod pitch in the fuel assembly;

– cladding material (zirconium alloy);

– fuel material (UO_2, UO_2+Al, in the latter case, the volume fraction of the fuel phase Frac_UO_2 is set);

– linear heat rate (LHEAT) taking into account axial (k_z) and radial (k_r) non-uniformity;

– inlet coolant temperature (TCOOL);

– coolant flow rate through the fuel rod cross section.

The gap conductance and the heat transfer coefficient between the coolant and the fuel cladding can be specified in the input-file, as well as calculated in the program itself (HGAP = 0.0).

General view of the input-file (for example, SCA fuel rod, nominal power 260 kW) without a description of the radial profile of the heat generation across the radius of the pellets in each axial segment is shown in Fig. 2.

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LHEAT=2.34     ;  linear heat rate [Wt/mm]
PITCH=12.750   ;  fuels rod pitch in the fuel assembly h     [mm]
HGAP=0.0       ;  gap gas conductance [Wt/mm^2*ºC)]     0.0035
H_ROD=50.0     ; fuel stack length                 [см]
ALPHA=0.0      ;  heat transfer coefficient [Wt/(mm^2*ºC)]  0 – calculated by program (0.05)
TCOOL=25.0     ;  inlet coolant temperature             [ºC]
RIF=0.0        ;  fuel pellet inner radius                  [mm]
ROF=3.770      ;  fuel pellet outer radius                  [mm]
RIC=3.860      ;  clad pellet inner radius                  [mm]
ROC=4.550      ;  clad pellet outer radius                  [mm]
Gmass=0.0000923   ;  mass flow rate              [kg/sec]
MAT_Pellet=0  ; fuel material type (thermal conductivity eq.):
0-UO2.
1-UO2+Al.
Frac(UO2)= 0.32   ;  volume fraction UO2 in dispersion composition UO2+Al
MAT_Claddig=5 ; cladding material type (thermal conductivity eq.):
5 - Zirconium alloy Zr1Nb
6 - stainless steel (06X18H10T)
RIP=0.1        ;  rod internal pressure [МПа]
t_exp=0;
1 – thermal expansion fuel material taken into account:
0 – thermal expansion fuel material do not taken into account.

q_z – axial profile:
bottom FR - q_z(1).
top FR q_z(24)

=1.0; =1.0; =1.0; =1.0; =1.0; =1.0;
=1.0; =1.0; =1.0; =1.0; =1.0; =1.0;
=1.0; =1.0; =1.0; =1.0; =1.0; =1.0;
=1.0; =1.0; =1.0; =1.0; =1.0; =1.0;
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Fig. 2. Structure of the input-file (InputData.txt)

A schematic view of the fuel element (fuel stack and clad) divided into axial and radial segments is shown in Fig. 3. Profiling step of fuel stack 20.83 mm.

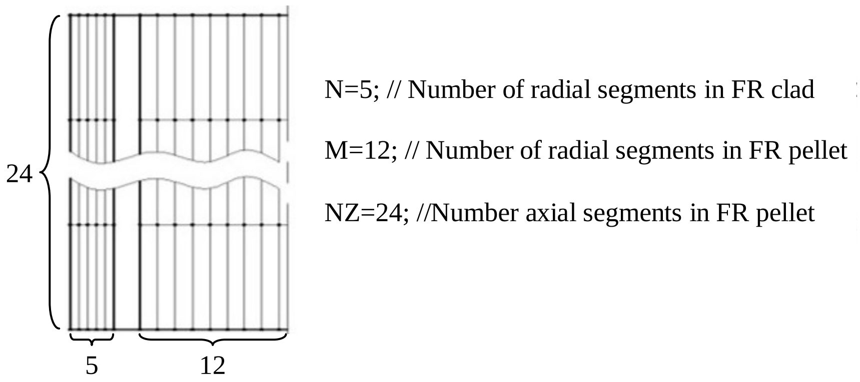

Fig. 3. Schematic view of the fuel element

3. Assumptions and boundary conditions accepted for calculations

The following assumptions are made in the calculations:

1. Heat flow in the axial direction is absent.

2. Deformation of the clad due to the difference of internal and external pressure is not considered.

3. Coolant – single-phase liquid – water. (boiling on the surface of fuel rods is absent).

4. The temperature on the pellet surface is calculated by the program, and r temperature distribution along the fuel radius is calculated as:

$$T(r) = T_{N+1} + \frac{1}{4\lambda_N}(r_{N+1}^2 - r^2)q_N - \left(\frac{1}{4\pi\lambda_N}q_N - \frac{1}{2\lambda_N}r_N^2 q_N\right)\ln\left(\frac{r}{r_{N+1}}\right),$$

where q_N – volume heat generation current segment fuel pellet,

r – current fuel pellet radius,

λ_N – thermal conductivity coefficient in fuel pellet material in the current layer.

This expression is derived from the heat-transfer equation in the layer j:

$$\frac{d}{rdr}\left(\lambda_j r \frac{dT}{dr}\right) = -q_j .$$

General solution:

$$T(r) = -\frac{1}{4\lambda_j}r^2 q_j + C_j \ln(r) + D_j ,$$

where C and D are constants selected from the boundary conditions. The boundary condition on the outer boundary of the layer, i.e. $r = r_{j+1}$:

$$T(r_{j+1}) = T_{j+1} ,$$

$$-\lambda_j \left.\frac{dT}{dr}\right|_{r_{j+1}} = J_{j+1} .$$

If the temperature is set at the pellet surface, constants for the last layer, i.e. for layer N, are obtained from the equations:

$$C_N = -\frac{1}{2\pi\lambda_N}Q_N + \frac{1}{2\lambda_N}r_{N+1}^2 q_N ,$$

$$D_N = T_{N+1} + \frac{1}{4\lambda_N}r_{N+1}^2 q_N - C_N \ln(r_{N+1}) .$$

4. Basic equations and program flow diagram

According to the developed model, at the first stage, the input data is read from the InputData.txt file and written into program variables and arrays.

The coolant temperature is calculated with account of the coolant flow rate and FR cross-section area. Equivalent diameter (hydraulic channel) is calculated as:

$$De = \frac{4 \cdot Srod}{P},$$

where *Srod* – area cross section FR,

P – wet perimeters, $P = \pi \cdot 2 \cdot Roc$,

Roc – outer cladding radius.

Coolant velocity significantly depends on the mass flow rate and coolant density, as well as on the flow area and is calculated as:

$$W = \frac{Gmas}{DenW \cdot Srod},$$

where *Gmas* – coolant mass flow rate,

DenW – density of the coolant.

Thus, coolant temperature in each axial segment of the FR is calculated based on the coolant temperature in the previous segment (for the first segment, the inlet temperature of the coolant, hydraulic channel of the FR T_{cool} is taken into account), and also based on coolant heating in the current axial fuel segment:

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$$T_{w\_i+1} = T_{w\_i} + \frac{q_{v\_i} \cdot h_i}{W \cdot Srod},$$

where q_{v_i} – volume heat generation in the axial segment,

h_i – height of the FR axial segment.

The relationship between linear and volumetric heat generation is determined by the expression:

$$q_v = \frac{q_l}{\pi} \cdot (R_{of}^2 - R_{if}^2),$$

where R_{if} – fuel pellet inner radius.

To calculate temperature of the cladding surface in the axial segment, except for the coolant temperature and linear heat rate, it is necessary to calculate the coolant – cladding surface heat transfer coefficient (if not specified in the file *InputData.txt*).

$$\propto = 0.023 \frac{\lambda_w}{De} \cdot \mathrm{Re}^{0.8} \cdot \mathrm{Pr}^{0.33},$$

where λ_w – thermal conductivity coefficient of water,

De – equivalent diameter,

Re – Reynolds ($\mathrm{Re} = (W \cdot De) / \nu$, ν – kinematic viscosity of water),

Pr – Prandtl ($Pr = (\mu \cdot \mathrm{Cp}) / \lambda_w$, μ – dynamic viscosity of water, Cp – heat capacity of water).

Having determined the heat transfer coefficient from cladding to coolant (α), the temperature of the fuel cladding is:

$$T_{oc\_i} = T_{w\_i} + \frac{q_l\_i}{2\pi \cdot R_{oc} \cdot \alpha},$$

where q_l – linear heat rate,

R_{oc} – outer clad radius.

The temperature of the inner surface of the cladding in each FR segment is based on the expression:

$$T_{ic\_i} = T_{oc\_i} + \frac{q_l\_i}{2\pi \cdot \lambda_{cl}} \cdot \log\left(\frac{R_{oc}}{R_{ic}}\right),$$

where λ_{cl} – thermal conductivity coefficient of cladding,

R_{ic} – inner clad radius.

GAP conductance (if not specified in the input data file InputData.txt) is defined as:

$$HGAP = T_{ic} + \frac{\lambda_{He}}{GAP},$$

$$GAP = R_{ic} - R_{of},$$

where λ_{He} – thermal conductivity coefficient of helium.

The temperature of the outer surface of the fuel (pellet):

$$T_{of} = T_{ic} + \frac{q_l}{2\pi \cdot R_{of} \cdot HGAP},$$

where R_{of} – outer fuel radius.

The general block-diagram of the program is shown in Fig. 4.

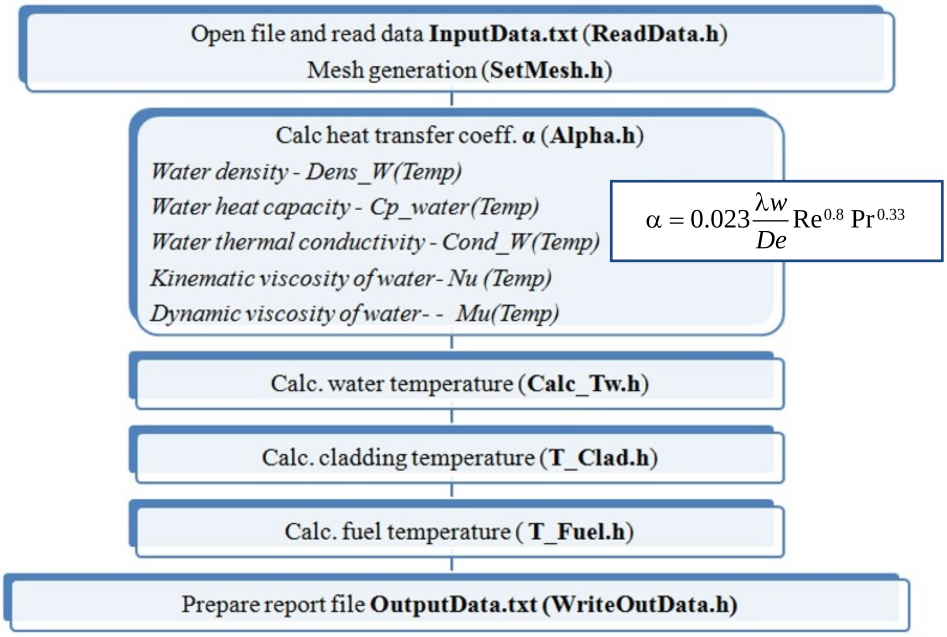

Fig. 4. B*lock-diagram of the program*

The output text file OutputData.txt contains the calculation results:

– the flow area and heat transfer coefficient of the coolant,

– temperature distribution of the coolant, shell, outer surface and center of the fuel core along the height of the fuel rods,

– temperature distribution along the radius of the fuel stack.

5. Temperature dependence of the thermal conductivity of UO_2 and UO_2+Al pellets

As a temperature dependence of the thermal conductivity coefficient of UO_2, a new standard correlation (model No 21) for UO_2 and (U, Gd) O_2 of the TRANSURANUS code was used [9]:

$$\lambda = \left(\frac{1}{a + a_1 bu + a_2 Gd + b_1 buT_p + b_2 GdT_p + bT} + \frac{c}{T^2} e^{d/T} \right) (1 - P)^{2.5},$$

where a, b, c, d – fitting coefficients,

bu – local burnup (MW·d/kgU),

Gd – local content of gadolinium, %wt.,

T – local temperature (K),

P – pellets porosity (0.05 or 5 %).

a=0.0375	b=2.165·10^{-4}	c=4.751·10^{9}
a_1=0.38360·10^{-2}	b_1=–0.90849·10^{-6}	d=16361
a_2=0.84476·10^{-2}	b_2=0.22149·10^{-5}	

For the dispersion fuel composition UO_2+Al with different contents of the fuel phase, the empirical dependence of the calculation of the thermal conductivity for heterogeneous systems (Odelevsky) was used (Fig. 5) [10, 11]:

$$\lambda_e = \frac{1}{4}((3\varphi_2 - 1)\lambda_2 + [3(1-\varphi_2) - 1]\lambda_1 + \sqrt{[(3\varphi_2 - 1)\lambda_2 + (3\{1-\varphi_2\} - 1)\lambda_1]^2 + 8\lambda_1\lambda_2}),$$

where λ_1 – thermal conductivity of uranium dioxide λ (UO_2),

λ_2 – thermal conductivity of aluminum λ (Al)=207 W/(m·K),

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φ_1 – volume fraction of fuel phase (UO_2),
φ_2 – volume fraction of aluminum (Al).

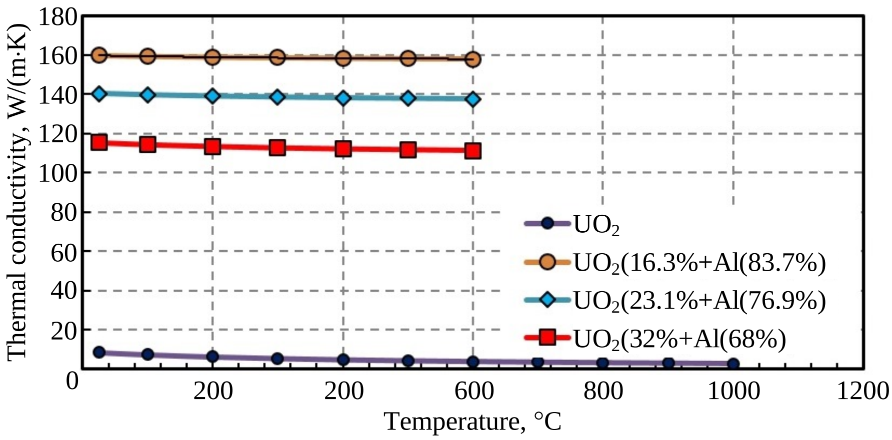

Fig. 5. Temperature dependences of the thermal conductivity of UO_2+Al pellets with different content of the fuel phase as compared with UO_2 fuel pellets

6. Input Data and calculation results

The basic data for calculating temperatures in the FA-Kh fuel assemblies are fuel rod geometric characteristics, types of materials, heat generation (calculated from the total nominal power of the SCA and WWR-M cores), inlet coolant temperature and coolant flow rate (Fig. 6 and Table 1).

№	Parametr	Unit of meas	Value
1	**Cladding – Zirconium alloy E110**		
2	Outer diameter	mm	9.1
3	Inner diameter	mm	7.72
4	**Fuel – UO_2 or 32%UO_2+Al pellets**		
5	Pellets diameter	mm	7.54
6	Fuel stack	mm	500
7	Pitch	mm	12.75

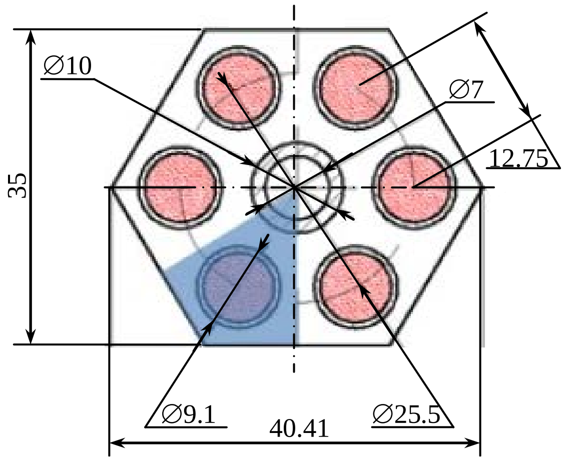

Fig. 6. Geometric characteristics of fuel rods and FA-Kh

Table 1

Operational characteristics of the SCA and WWR-M

№	Parameter	unit of meas.	SCA	WWR-M
1	Number of FAs		37	260
2	Power (nominal)	kW	260	8400
3	Coolant inlet temperature	ºC	25	38
4	Volume flow rate through the core	m^3/hour	82	1200
5	Volume flow rate through cross section, FA	m^3/hour	2	4.6
6	Mass flow rate through cross section FR	kg/sec	$9 \cdot 10^{-5}$	$21.4 \cdot 10^{-5}$
7	Average linear heat rate FR	W/mm	2.34	10.8
8	Maximum linear heat rate FR, (Kz=1.12)	W/mm	2.62	10.8

A single FA-Kh FR cross-section is 0.757 cm^2, equivalent diameter is 1.06 cm, and depending on the coolant flow rate through the core and the amount of FAs, the flow rate through FR will vary.

An example of the FR axial power shape (*q_z*): 0.146; 0.290; 0.429; 0.560; 0.682; 0.792; 0.889; 0.970; 1.035; 1.082; 1.110; 1.120; 1.110; 1.082; 1.035; 0.970; 0.889; 0.792; 0.682; 0.560; 0.429; 0.290; 0.146; 0.112.

Calculations using the developed model showed that heating of the coolant along the FA-Kh fuel stack length (500 mm) was ~3 °C for reactor WWR-M. The temperature distribution of the coolant along the height of the fuel stack for the SCA and WWR-M operating conditions are shown in Fig. 7.

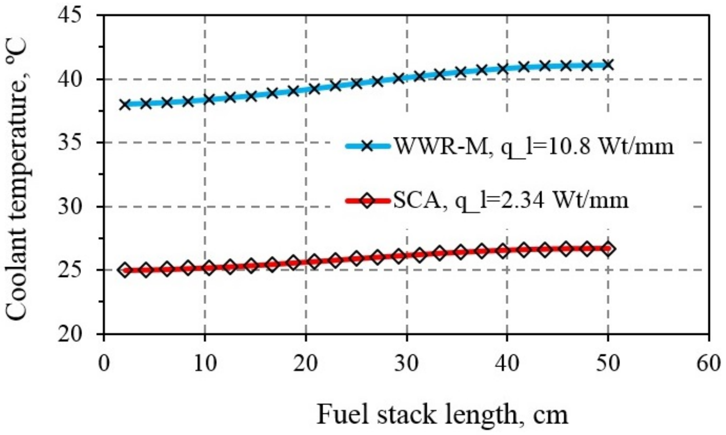

Fig. 7. Distribution of coolant temperature along the fuel stack

The temperature distribution (central part of fuel pellets with different types of material – UO_2 and 32 % UO_2+Al) along the height of the fuel stack for the SCA and WWR-M operating conditions are shown in Fig. 8.

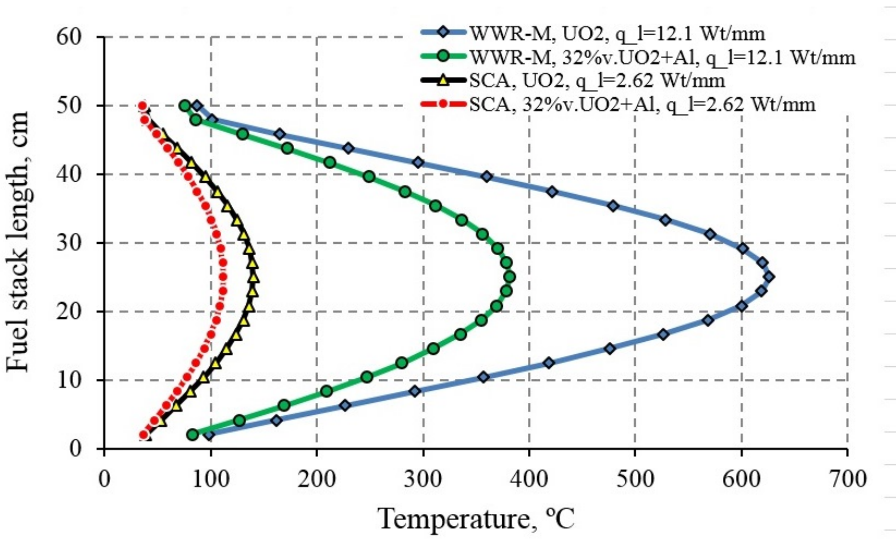

Fig. 8. Axial temperature distribution in center of fuel pellets

Radial temperature distribution for pellets with different types of fuel material: UO_2 and 32 % UO_2+Al in the segment with maximum heat rate and temperature is shown in Fig. 9.

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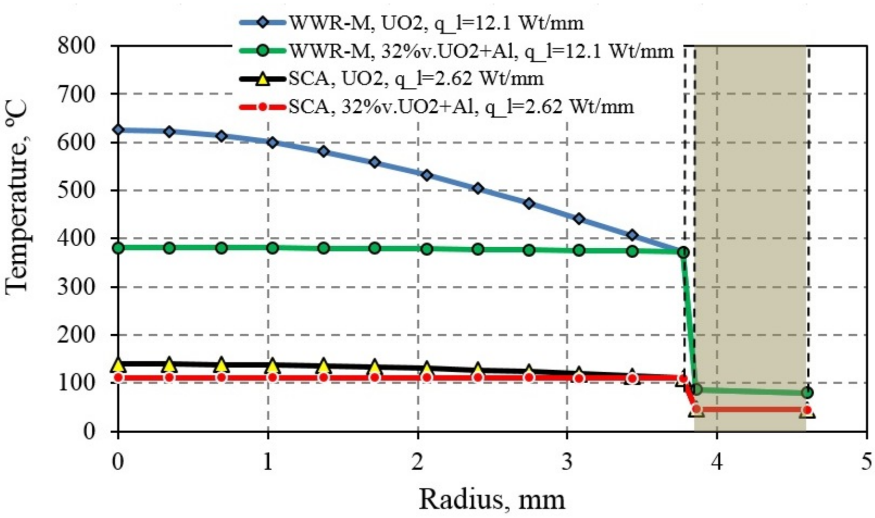

Fig. 9. Temperature distribution along the radius of the fuel rod

It is well known that low thermal conductivity causes high temperature gradients along the cross section of the fuel pellets. In our case, low heat rate in FA-Kh fuel rods, for example, in SCA (*q_l* = 2.62 W/mm), causes low temperature in the center of the fuel ~140 °C. In case of using dispersion fuel UO_2+Al, the temperature was 112 °C with a minimum (several degrees) temperature gradient.

For the case of WWR-M research reactor, the temperature in the center of the fuel was ~626 °C (UO_2 fuel) and ~381 °C for the dispersion composition UO_2+Al.

Thus, the calculations show a significant effect of the fuel material type (UO_2 or dispersion composition UO_2+Al) on the fuel center temperature as applied to the conditions of operation in the subcritical assembly SCA and the research reactor WWR-M.

It should be noted that the maximum temperature of the cladding surface (WWR-M operating conditions, *q_l*=12.1 W/mm) was 86.5 °C, which does not exceed the boiling point (vaporization), under nominal operating conditions of the WWR-M reactor.

Conclusions

A description of the fuel rod model for the FA-Kh fuel assembly developed as alternative fuel for the “Neutron Source” nuclear facility based on the sub-critical assembly and WWR-M research reactor is given.

Comparative temperature calculations for fuel rods with different types of fuel (UO_2 and UO_2+Al) were carried out. It is shown that for SCA operating conditions (linear power of fuel rod is 2.62 kW/m), the temperature in the central part of the fuel reaches 140 °C for UO_2 and 112 °C for the dispersion composition 32%UO_2+Al.

For WWR-M operating conditions (linear power of fuel rod is 12.1 kW/m), the temperature in the central part of fuel reaches 626 °C for UO_2 and 381 °C for the dispersion composition 32 % UO_2+Al which is explained by higher thermal conductivity.

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