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A SIMPLE COMPUTER PROGRAMME FOR THE CALCULATIONS OF REACTOR CHANNEL TEMPERATURE DISTRIBUTION

Farag Muftah BSEBSU* and Gábor BEDE**

Department for Energy
Technical University of Budapest
H-1521 Budapest, Hungary
Fax: +36 1 463-3272
* Phone + 36 1 463-2183, email: Walid@eta.enrg.bme.hu
** Phone +36 1 463-2594

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Abstract

This paper presents the results of THERMAL computer programme and checks its validity using KFKI reactor fuel channel as a sample problem of the implementation. The simple computer programme has been developed to perform the necessary calculation of axial temperature distribution along a multiregion channel, and other hydraulics parameters for any reactor coolant channel. The criteria of the temperature are such that the surface boiling should be avoided at any part of the cladding surface of the coolant channel. The results of the THERMAL compared with the results recorded in the reactor safety report are in good agreement. Finally, the results of this calculation at equilibrium and starting core with maximum and average core loading are given.

Keywords: thermohydraulics, research reactors, fuel channel temperatures.

Nomenclature

T_{f1} = Reactor coolant inlet temperature, C
$q''_{i\max}$ = maximum heat flux rate, W/cm^2
K_i = wet and heated contours, cm
H_e = extrapolated length, cm
C_p. = coolant specific heat, J/kg.K
$\dot{m}_i$ = coolant mass flow rate, kg/sec
α = heat transfer coefficient, W/cm^{2}K
λ = Thermal conductivity coefficient, W/cm.K.
R = channel thermal resistance, K/W
Z = axial distance, cm
D_e = equivalent heated diameter, cm
$T_{cl}(z)$ = axial cladding surface temperature, C
$T_m(z)$ = axial centre line fuel temperature, C
$T_f(z)$ = axial coolant temperature, C

A_{Al+U} = cross sectional area of Al and Uranium ring, cm^2
A_{H_2O} = cross sectional area of water ring, cm^2

1. Introduction

The main purposes of heat transfer analysis are accomplished by choosing the operating temperatures within the detailed and precise limits; by determination of the temperature field in a coolant channel, and determination of the parameters governing the heat transport rate at the channel walls. Then these parameters can be used to choose materials and flow conditions that maximise heat transport in the channel.

The basis of the world-wide LWR safety research programmes is to develop computer codes to analyse abnormal events. These codes are to be based on physical understanding, yet these codes have become so large and complex that few people understand all of the models employed or the numerical techniques. There are a lot of computer codes, of which I'll mention the following:

· RELAP5 code is an advanced, one dimensional, code based on a nonhomogeneous, non equilibrium mode (RANSOM et al, 1978).

· TRAC-P1A (PRYOR and SICILIAN, 1978) embodies the current state of the art and features a nonhomogeneous, multidimensional fluid dynamic treatment.

In the Western reactor design a channel with single passage is usually used (*Fig. 1*). In the Eastern reactor design, however, a channel design with a multiple passage is common, usually, the latter type is modelled as a single equivalent channel if the famous thermo hydraulic computer programmes such as RELAP are to be used. The error introduced by this can be avoided by taking geometry into consideration. Therefore, it is desirable to have a programme suitable for the calculations of the thermohydraulics of a channel with multisections (*Fig. 2*). In addition the use of big programmes is time consuming . In design and operation simple methods are desirable.

2. Heat Transfer Mathematical Model [1]

The fuel channel can be divided into four hydraulic regions. The most inner channel is a tube, marked by 'A ', followed by a channel with an annulus cross section, marked by 'B '. The outer wall of the next channel is hexagonal, while its inner wall is a circle. This channel is marked by 'C'. The last region, marked by 'D' is that part of the channel, which belongs both to the given assembly and its neighbouring assemblies.

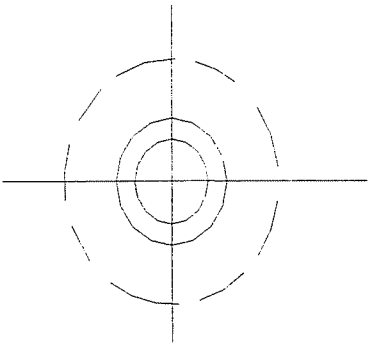


Fig. 1. Single passage fuel channel

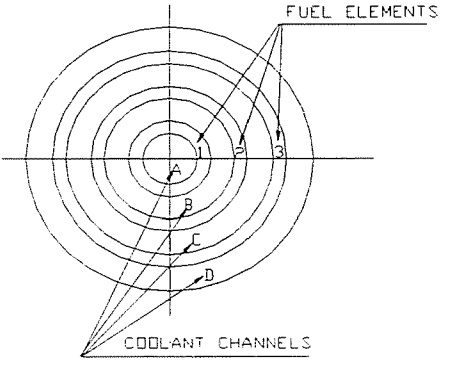


Fig. 2. Multiple passage fuel channel

The starting point of the modelling is to simulate the system as presented before the axial temperature distribution along the channels 'A' and 'D' is given by:

$$T_f(z) = T_{f1} + \frac{q''_{imax} K_i H_e}{\pi C_p \dot{m}_i} \left[1 - \cos\left(\frac{\pi z}{H_e}\right)\right] .$$

The axial temperature distribution of the cladding surface along both channels 'A' and 'D' is given by:

$$T_{cl}(z) = T_{f1} + \frac{q''_{i\max} K_i H_e}{\pi C_p \dot{m}_i} \left[1 - \cos\left(\frac{\pi z}{H_e}\right)\right] + \frac{q''_{i\max}}{\alpha_i} \sin\left(\frac{\pi z}{H_e}\right) .$$

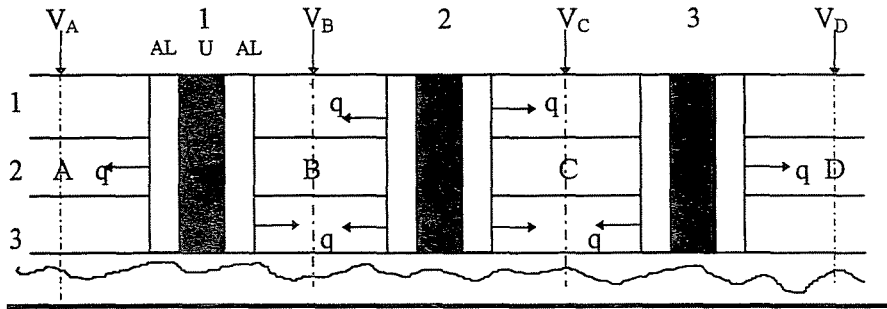
The axial temperature distribution of the centre of fuel meat is given by:

$$T_m(z) = T_{f1} + \frac{q''_{i\max} K_i H_e}{\pi C_p \dot{m}_i} \left[1 - \cos\left(\frac{\pi z}{H_e}\right)\right] + q''_{i\max} R K_i \sin\left(\frac{\pi z}{H_e}\right) .$$

The above algorithm is valid only for the two regions of our model as mentioned before 'A' and 'D' as shown in *Fig. 3*.

As for the other two regions B' and 'C': the temperature distribution along these channels is not the same as in channels 'A' and 'D' because each channel gets the heat from both sides. For example channel 'B' gets the heat from the fuel elements (1) and (2). The temperature distribution function is given by:

$$T_f(z) = T_{f1} + \left[\frac{(q''_{i\max} K_i + q''_{j\max} K_j) H_e}{m_{i/j} \pi C_p}\right] \cos\left(\frac{\pi z}{H_e}\right) .$$



r*	0											18.38
r_{AL}		3.0		5.5		8.5		11.0		14.18	16.80	
Δr_U			0.7				0.7				0.74	
Δr_{H2O}		3.0			3.0			3.18			1.58	
d_e		6.0			6.0			6.176			6.274	
K		18.85		34.56		53.41		69.12		93.53		110.85
		18.85			87.97				162.65		110.85	
A_{AL+U}			66.76				153.15				255.50	
A_{H2O}		28.27			131.90			251.20			174.1	

* All dimensions are in mm and mm^2

Fig. 3. The axial cross section of a channel

For engineering analysis, the difference between the wall temperature and the bulk flow temperature is obtained by defining the heat transfer coefficient (α) through the dimensionless Nusselt number [2]:

$$Nu = f\left(Re, Pr, Gr, \frac{\mu_w}{\mu_b}\right) = \frac{\alpha D_H}{\lambda} .$$

The Nusselt number shows in both experiment and theory what can be given for almost all non-metallic fluids given by the Seider and Tate equation [3]:

$$Nu = 0.023\mathrm{Re}^{0.8}\mathrm{Pr}^{0.4}\left(\frac{\mu_w}{\mu_b}\right)$$

For more accurate calculations and for cases when $\mu_w = \mu_b$ the Dittus-Boelter equation is the most universally used correlation in the reactor calculations [4]:

$$Nu = 0.023\mathrm{Re}^{0.8}\mathrm{Pr}^{0.4}$$

For $0.7 < Pr < 120$, $Re > 10000$, and $L/D > 60$. All fluid properties are evaluated at the arithmetic mean bulk temperature. By transforming

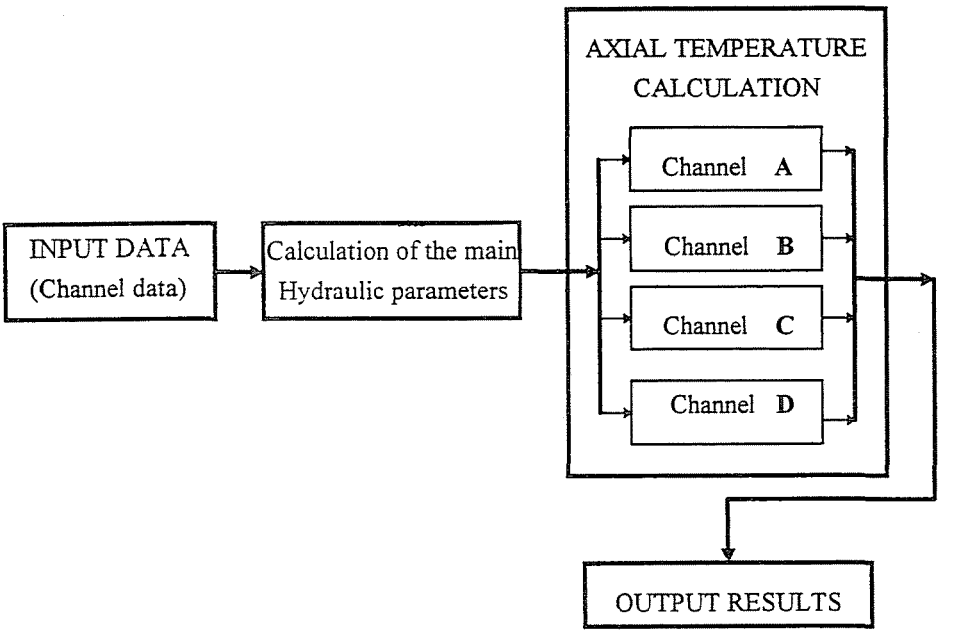


Fig. 4. Thermal code flow diagram

the mathematical relations to the PASCAL computer programme (THERMAL), and the main flow diagram they are as shown below.

3. Programme Validation

The implementation of the computer programme is done by using the Budapest research reactor (KFKI reactor) as a testing sample problem. The Budapest Research Reactor is a tank type reactor, moderated and cooled by light water. The reactor is in a cylindrical reactor tank, made of a special aluminium alloy with a diameter of 2300 mm, and a height of 5685 mm. The heavy concrete reactor shielding block is situated in a rectangular semihermetically sealed reactor hall. The area of the reactor hall is approximately 600 m^2. It is ventilated individually.

The fuel of the research reactor is that of the VVR-SM type (Russian product). It is an alloy of aluminium and uranium-aluminium eutectic with aluminium cladding. The uranium enrichment is 36%, the average U^{235} content is 39 g/fuel element. The fuel element contains three fuel tubes, the outer tubes are of hexagonal shape, while the two inner ones are cylindrical. The active length of fuel elements is 600 mm.

The equilibrium core consists of 223 fuel assemblies, with a lattice pitch of 35 mm. The core is surrounded radially by a solid beryllium reflector. The reactor is equipped with boron carbide safety and shim rods. There is a stainless steel rod for the purpose of automatic power control [5].

The axial cross-section of the fuel channel of the KFKI reactor is as shown in *Fig. 3*. The most important parameters of the ring shaped U-Al fuel, of the aluminium cladding, and of the cooling channels are $(r, \Delta r)$. Moreover, the wet and heated contours (K), the equivalent diameter (d_e), cross-section area (A), and the heated surfaces are presented and the main geometrical data of the VVR-SM type assembly are also presented in *Fig. 3*.

4. Calculation and Results

The axial temperature and heat flux distribution along each channel of the fuel element for both calculations (THERMAL code and KFKI group) are shown in *Figs. 5 - 8*, respectively. The maximum cladding temperature for each layer of the fuel element at equilibrium and starting core (maximum loading and average loading) is given in the *Tables 1–3*.

Table 1
Maximum heat flux and cladding surface temperature for layer number One

State of the CORE	Fuel Loading	Variables	KFKI First inner	Result Layer outer	Thermal First inner	Result Layer outer
Equilibrium Core	Max. Loading	q_{max} [W/cm^2]	65.13	53.39	65.13	53.39
		$T_{cl.max}$ [C]	96.43	88.94	97.05	84.02
	Average Loading	q_{max} [W/cm^2]	36.31	29.84	36.31	29.84
		$T_{cl.max}$ [C]	76.39	72.17	76.85	69.53
Starting Core	Max. Loading	q_{max} [W/cm^2]	75.79	60.99	75.79	60.99
		$T_{cl.max}$ [C]	96.44	88.33	98.80	83.94
	Average Loading	q_{max} [W/cm^2]	62.21	50.06	62.21	50.06
		$T_{cl.max}$ [C]	88.35	81.65	90.35	78.10

5. Conclusion

In this paper, we have implemented a simple computer programme THERMAL and checked its vitality by using KFKI reactor fuel channel as a sample problem. This programme was developed to determine the cladding

Table 2
Maximum heat flux and cladding surface temperature for layer number Two

State of the CORE	Fuel Loading	Variables	KFKI Second inner	Result Layer outer	Thermal Second inner	Result Layer outer
Equilibrium Core	Max. Loading	q_{max} [W/cm^2]	60.73	55.05	60.73	55.05
		$T_{cl.max}$ [C]	93.09	90.50	90.69	84.53
	Average Loading	q_{max} [W/cm^2]	32.65	30.21	32.65	30.21
		$T_{cl.max}$ [C]	73.70	71.85	72.44	69.67
Starting Core	Max. Loading	q_{max} [W/cm^2]	70.34	63.35	70.34	63.35
		$T_{cl.max}$ [C]	92.90	90.08	91.44	84.63
	Average Loading	q_{max} [W/cm^2]	55.58	50.86	55.58	50.86
		$T_{cl.max}$ [C]	84.23	81.30	83.02	78.30

Table 3
Maximum heat flux and cladding surface temperature for layer number Three

State of the CORE	Fuel Loading	Variables	KFKI Third inner	Result Layer outer	Thermal Third inner	Result Layer outer
Equilibrium Core	Max. Loading	q_{max} [W/cm^2]	79.27	80.93	79.27	80.93
		$T_{cl.max}$ [C]	103.9	97.72	101.4	100.5
	Average Loading	q_{max} [W/cm^2]	30.74	29.15	30.74	29.15
		$T_{cl.max}$ [C]	72.08	70.84	70.66	71.13
Starting Core	Max. Loading	q_{max} [W/cm^2]	91.77	92.69	91.77	92.69
		$T_{cl.max}$ [C]	103.7	97.03	102.1	101.9
	Average Loading	q_{max} [W/cm^2]	52.24	49.13	52.24	49.13
		$T_{cl.max}$ [C]	81.85	79.90	80.18	81.21

surface axial temperatures, and other thermalhydraulics parameters. In view of the simplicity of the written programme, the results are in good agreement with those mentioned in the reactor safety report published by KFKI research centre safety group. *Figs. 5* and *6* exhibit the axial temperature distribution in the cladding surfaces of the fuel element (see *Fig. 3*). Maximum deviation of the calculated temperatures from those given in the reactor safety report *Figs. 7* and *8* is less than 2%. In view of the simplicity of the calculation the agreement is considered to be good. The differences may be attributed to the following reasons:

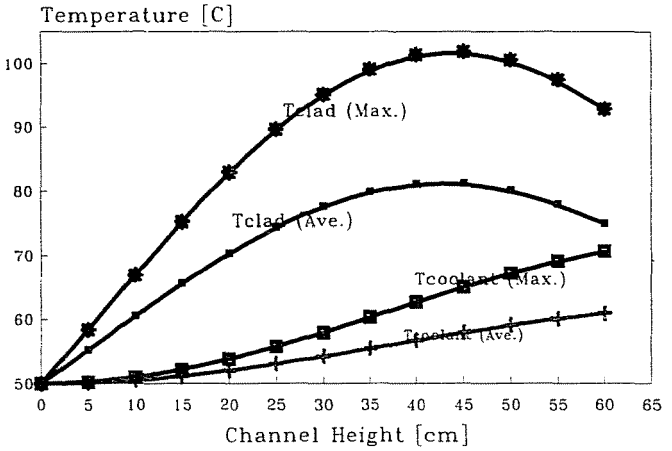


Fig. 5. Axial temperature distribution for KFKI fuel channel calculated by THERMAL code at equilibrium core with (max. and average) loading

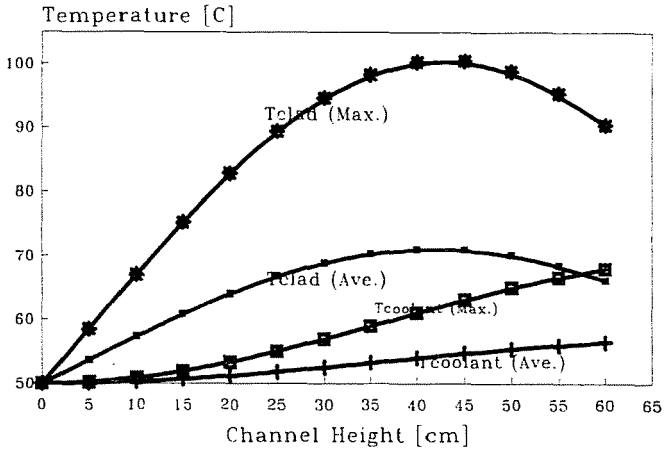


Fig. 6. Axial temperature distribution for KFKI fuel channel calculated by THERMAL code at starting core with (max. and average) loading

1. Using of Dittus–Boelter correlation for calculation of heat transfer coefficient.
2. The heat flow from inside fuel element to outside (cladding surface – coolant channel) due to using high fuel thermal conductivity.

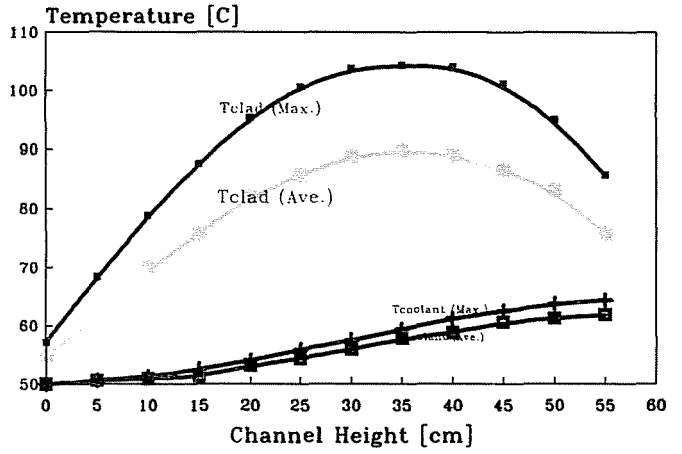


Fig. 7. Axial temperature distribution for KFKI fuel channel calculated by KFKI group at equilibrium core with (max. and average) loading

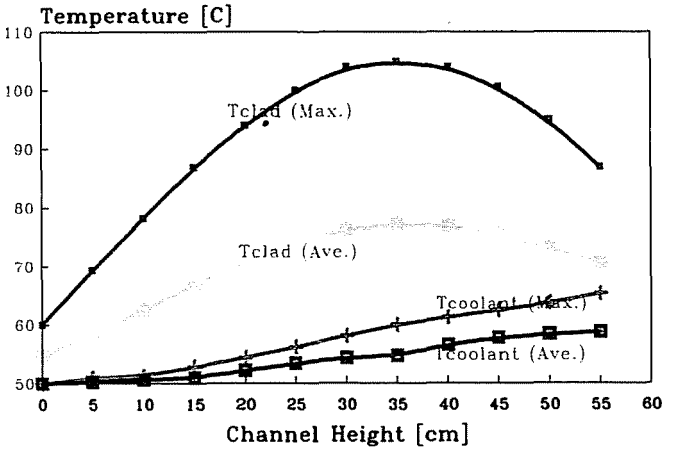


Fig. 8. Axial temperature distribution for KFKI fuel channel calculated by KFKI group at equilibrium core with (max. and average) loading

6. Doubts of Calculations

First of all, according to our results and KFKI results about Budapest research reactor: using a simple computer programme 'THERMAL', its results are quite correct for the cladding surface temperature, but there are some doubts on these results:

1. The chemical composition of reactor fuel and cladding materials, are actually unknown.
2. There are not enough data about the neutron and heat fluxes distribution inside the reactor core experimentally.
3. The velocities of the coolant distribution inside the fuel elements are not actually available at this time.
4. All our calculations or results are based on some data from KFKI research centre (oral contact).

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