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THERMAL-HYDRAULIC ANALYSIS OF A RESEARCH REACTOR ON PERSONAL COMPUTER WITH TARR CODE

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SUMMARY. The code has been created for thermal-hydraulic calculation of stationary regime of nuclear research reactor, using personal computer. The main objective of the code is to compute the thermal parameters in the reactor core in order to avoid any accident. The code can be applied for many fuel assamblies available in research reactors.

The Dalat research reactor has been put into operation with 500kW thermal power since March 1984. In studies related with upgrading of the reactor power as well as to ensure its safe operation, it is necessary to have computer codes to make quantitatively predictions of the thermal-hydraulic behaviour of the reactor during the operation. In the framework of a research contract with the International Atomic Energy Agency (IAEA) the TARR code was developed by a group of the Dalat Nuclear Research Institute and the Institute of Mechanics, Hanoi. The code was written for thermal-hydraulic analysis of a research reactor in steady-state operating conditions and used on IBM PC/AT or compatible personal computers. In writting it, the main emphasis has not been laid on originality but rather on using a formalism which is thoroughly tested [1].

The purpose of the analysis is to establish the range of parameters in the reactor core which would meet the safety criteria: to avoid nucleate boiling and to avoid any damage to the clad integrity. So important outputs are [2]:

- Peak fuel temperature, excessive value of which could lead to long-term fuel growth or eventual fuel melting and cause high stress in the clad.
- Cladding temperature, excessive value of which could lead to degradation of the clad mechanical properties.
- Departure from Nucleate Boiling (DNB) ratio. Experiencing DNB could lead to flow blockage, burn-out and rod failure.

Code TARR is a finite difference solution of the one-dimensional momentum and energy transport equations along the cooling channels (or subchannels) with given axial distribution of power generation. Radial and azimuthal distributions of core power generation are taken into account by dividing the core to the channel groups in each from which one representative channel is considered. For ceramic fuel (rod cluster type assemblies) most of the temperature drop from the center of the fuel rod to the coolant takes the place in the fuel, a significant drop occurs in the fuel-cladding interface. The analysis of heat conduction in the fuel rod, therefore, is included in TARR code. The code handles single-phase flow which is typical in most research reactor cores [3]. However, in order to study limiting conditions which would meet the safety criteria, several two-phase heat transfer and CHF (Critical Heat Flux) correlations are considered. These correlations identify the location where the ONB (Onset of Nucleate Boiling) or the Developed Subcooled Nucleate Boiling occurs and determine the DNB ratio. Three general cathegories of reactor primary coolant system [2] are treated in TARR code: upward flow, gravity flow and pressurized closed loop.

Detailed algorithm of TARR code is performed in [4]. The reactor core is considered to be made up of a number of paralleled cooling channels connected to the same inlet and exit plenium. Each fuel or non-fuel assembly is assumed to be a cooling channel which in turn may be divided to the paralleled subchannels (in the case of concentric annular fuel assemblies, fuel tubes divide the channel to separated subchannels). The channels are of identical geometry. For a fixed value of core flow rate G (or coolant flow rate accross channel), actual core flow distribution (or distribution among subchannels) will be found as the solution of the system of non-linear algebraic equations [1]:

$$
\begin{aligned}
&\Delta P_i(G_1, G_2, \ldots, G_n) = \Delta P \qquad (i = \overline{1,n}) \\
&\sum_{i=1} G_i = G
\end{aligned}
\tag{1}
$$

where ΔP_i and G_i is the pressure drop and coolant mass flow rate accross channel (or subchannel) i; ΔP is the pressure drop accross the core; n is the number of paralleled channels (or subchannels). The system of Eqs. (1) can be solved by a numerical iteration. This iteration proved to converge very fast in practice. To calculate pressure drops ΔP_i, and then axial variations of coolant parameters, when flow distribution $\{G_i\}$ is fixed, a system of momentum and energy transport Eqs. for one-dimensional single-phase flow. is considered. For channel (subchannel) i, it can be written as:

$$
\begin{aligned}
\frac{dP_i(z)}{dz} =& - \rho_i(z) g \sin\alpha - \Big(\frac{G_i}{S_i}\Big)^2 \frac{d}{dz}\Big[\frac{1}{\rho_i(z)}\Big] - \\
&- \sum_{m=1}^{M} \delta_\varsigma(z - z_m)\Big(\frac{G_i}{S_i}\Big)^2 \frac{1}{2\rho_i(z)} - \\
&- \Big(\frac{G_i}{S_i}\Big)^2 \frac{\varsigma_i(z)}{2\rho_i(z) d_i} \qquad (i = \overline{1,n}) \\
\frac{dh_i(z)}{dz} =& \frac{q_i(z)\Pi_i}{G_i} \qquad (i = \overline{1,n})
\end{aligned}
\tag{2}
$$

where z is axial coordinate; $P_i(z)$, $\rho_i(z)$, $h_i(z)$ are axial variations of coolant pressure, density and enthalpy along channel (subchannel) i; g is free fall acceleration; α is the angle of flow direction and z axis; δ_ς is δ - function performing hydraulic resistant of spacer grids, specified as input data in TARR code; z_m is spacer grid location coordinate; ς_i and d_i are the friction factor and hydraulic diameter of channel (subchannel) i; Π_i, q_i are the heated perimeter and surface heat flux in channel (subchannel) i.

The system of Eqs. (2) supposed some simplification: heat conduction along z axis is neglected, azimuthal variations of parameters are not taken into account and heat transfer between channels is absent. Hence, the boundary conditions can be expressed as follow:

$$
P_i(z)\big|_{z=0} = P_{in}; \quad h_i(z)\big|_{z=0} = h_{in} \quad (i = \overline{1,n}) \tag{3}
$$

where P_{in} and h_{in} are coolant pressure and enthalpy at the channel inlet and defined as input data.

The surface heat flux q_i can be defined after solving steady-state heat conduction in fuel cylindrical tubes:

$$-\text{div}\,\left(\lambda(T)\text{ grad } T\right) = q_v \tag{4}$$

with boundary condition:

$$\lambda \text{ grad } T\big|_S = \alpha\left(T\big|_S - T_i\right) \tag{5}$$

where S denotes the cladding surface; T_i is coolant temperature in channel i; T is radial temperature field inside fuel tube; α is the heat transfer coefficient to the coolant at cladding surface; λ is the thermal conductivity of fuel tube. Equation (4) with (5) can be solved analitically after being linearized by introducing new variable [5].

$$\Lambda(T) = \int_{273}^{T} \lambda(\xi)d\xi \tag{6}$$

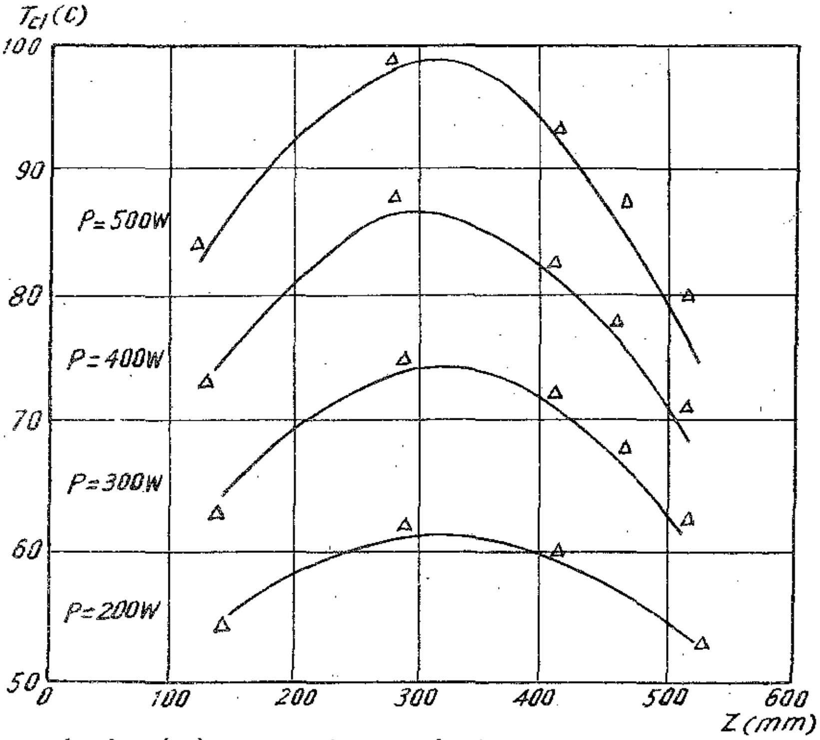
The system of equations (2), (3) is integrated by Euler method and equation (4) is solved for each channel increment. It is obvious that in the case of concentric annular fuel assemblies, the value of surface heat flux in one subchannel depends on heat transfer conditions not only in this subchannel but also in the adjacent ones, thus in all the others. Hence it will be found at each increment by an iteration. In the case of subcooled convection, heat transfer coefficient, and then cladding temperature, is calculated by Ditus - Boelter or Collier correlations, depending on the Reynolds number [6, 7]. Cladding temperatures at the ONB and at the developed nucleate boiling are defined by Bergles - Rosenow and Mc Adams correlations respectively [2]. Actual cladding temperature is the minimum of these temperatures. The use of different heat transfer correlations may cause the divergence in iterative calculation of surface heat flux. Then the heat flux will be considered as averaged value and the adequacy of this simplification is proved only by practice. A literature survey of CHF correlations applicable to low-pressure research reactor found that DNB data are very limited. However several experimentally-deduced CHF correlations are included in TARR code following [2, 8] among them the Lund, Mirshak, Labuntsov, Mactbeth and Bernath correlations. Heat flux at onset of flow instability is calculated by the Forgan empirical correlation [2]. The available core flow rate G is calculated with respect to concrete primary coolant system as solution of an equation based on the assumption that the net pressure difference is null in a closed loop. This equation is solved numerically by the halving method [9]. Correlations for thermophyscal propertes of coolant included in the code are verified by comparison with standard values presented in [10].

The sample problem presented is steady-state thermal hydraulic analysis of Dalat reactor. It is a swimming pool type reactor (TRIGA reactor) with Soviet fuel assemblies VVRM-2. The fuel assembly is concentric annular tubes, the outer "tube" is a hexagonal prism and is considered as a round tube with equivalent radii. Primary coolant system is based on free convection upward flow through the reactor core supported by a drawing well set above the core. All data needed for analysis by the code were prepared by the Dalat Nuclear Research Institute [11]. Some comparisions between values computed by TARR code and measured in experiments on Dalat reactor [12] have been done and they showed acceptable agreement. The following table shows the computed and measured temperatures for the hottest fuel assembly i.e. the nearest to the central neutron trap (the channel No 9-6 in Fig. 2).

Reactor Power (kw)	Max coolant temperature		Max cladding temperature	
	Measured (°C)	Computed (°C)	Measured (°C)	Computed (°C)
50	32.0	32.68	41.0	40.1
100	36.5	35.90	47.0	46.3
200	44.0	41.67	62.0	59.5
300	48.0	47.42	74.0	72.1
400	53.0	53.11	86.0	84.75
500	57.0	57.52	98.0	96.91

Cladding temperature is the most important parameter to verify the adequacy of the code. Axial variations of cladding temperature T_{cl} at outer surface (Fig. 2) of the fuel assembly No 9-6 at different core power levels are presented in Fig. 1. In this case all four contol rods are equally immersed in the core. The experiments give a little high values. It is plausible due to the fact that the fuel assembly equipped with thermal couples is the new one while the others are with 3 years burn-up.

At present the next version of TARR code is under development. It will include thermal hydraulic analysis during transients conditions, the purpose of which is safety analysis in the cases of loss of coolant flow accident (LOCA). In this case some two-phase flow heat transfer regimes will be taken into account:



Δ - Measured values [12]; ——— - Computed values

Fig. 1. Axial variations of cladding temperature at outer surface of the fuel assemby No 9-6 at 200W, 300W, 400W and 500W power levels.

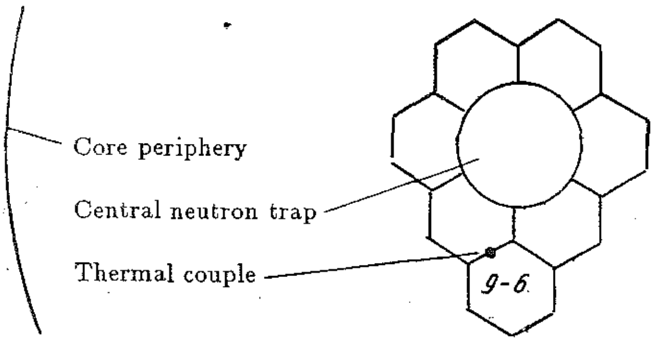


Fig. 2. Location of the fuel assembly No 9-6 in reactor core nucleate boiling, stable film boiling, flow transition boiling, pool film boiling, forced convection vaporazation and superheated convection...

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TARR-CODE - PHÂN TÍCH CHẾ ĐỘ NHIỆT-THỦY ĐỘNG CỦA LÒ PHẢN ỨNG NGHIÊN CỨU TRÊN MÁY TÍNH CÁ NHÂN

Trên cơ sở mô hình nhiệt-thủy lực của lò phản ứng nghiên cứu đã tạo ra được bộ chương trình trên máy tính cá nhân để tính toán các thông số nhiệt, thủy lực của lò trong chế độ dừng. Mục đích chính của chương trình là thiết lập các thông số trong vùng hoạt để tránh các sự cố. Chương trình cho phép dùng tính cho một số dạng cấu hình thanh nhiên liệu phổ biến trong các lò phản ứng.