Regional Meeting: Nuclear Energy in Central Europe
Present and Perspectives
Portorož, SLOVENIA, 13.-16. June 1993

TRISTAN
A Computer Program for Calculating Natural Convection Flow Parameters in TRIGA Core

I. Mele
"J. Stefan" Institute, Ljubljana, Slovenia

Abstract

TRISTAN is a computer program for calculating the flow parameters in a coolant channel of reactor core, cooled by natural convection. The code is based on simplified physical model and provided by its own data base of material properties. It is very simple to use and can run on any personnal computer. The code was developed at "J. Stefan" Institute for simple steady-state thermohydraulic analysis of low power *TRIGA* research reactors.

In this paper the physical model is briefly discussed and some results of parametric study of steady-state thermal performances of *TRIGA Mark II* reactor in Ljubljana are presented.

Povzetek

TRISTAN je računalniški program za izračun temperature in hitrosti hladila v stacionarnem stanju v hladilnem kanalu reaktorske sredice, hlajene s naravno konvekcijo. Razvit je bil na institutu "J. Stefan", namenjen pa je predvsem ocenam termohidravličnih lastnosti v *TRIGA* reaktorjih majhnih moči. Uporablja preprost fizikalni model, zato teče na kateremkoli osebnem računalniku. Ker je opremljen tudi s potrebno bazo materialnih podatkov, je njegova uporaba zelo preprosta.

V tem prispevku je zelo kratko opisan fizikalni model in predstavljeni nekateri rezultati termohidravličnega izračuna hlajenja sredice *TRIGA Mark II* v Ljubljani.

1 Introduction

TRISTAN is designed for steady-state thermohydraulic analysis of *TRIGA* research reactors operating at low power level of 1-2 MW in an open pool or tank, where normal pressures are not exceeding 2 bars.

A single flow channel with no cross flow between adjacent channels is considered in the analysis. Triangular coolant channel in *TRIGA* reactor is schematically presented in Fig. 1. It is bounded by three fuel elements of the same type in equilateral triangular arrangement. Coolant channel extends from the bottom grid plate to the

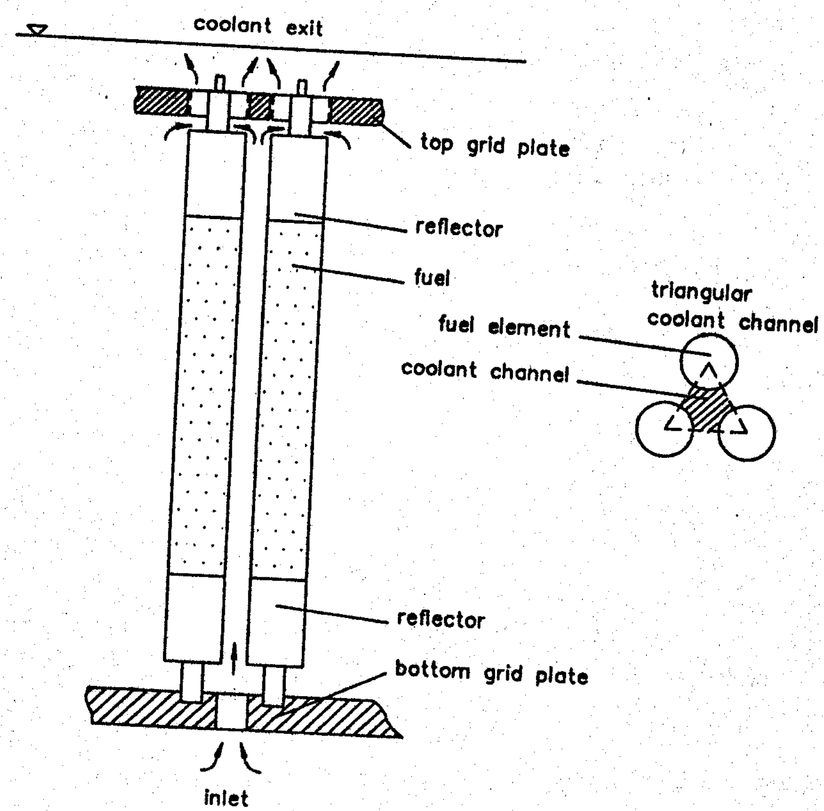

Figure 1: Triangular coolant channel in TRIGA reactor, side and top view.

top grid plate. A real or virtual chimney above the channel can also be taken into account.

The coolant enters the coolant channel through special holes in the bottom grid plate, passes the unheated lower part of fuel element (e.g. bottom end-fixture and axial reflector), the active part of the core, the upper axial reflector and top end-fixture and through the holes in the upper grid plate leaves the channel. Inlet coolant temperature and density T_0 and ρ_0 are equal to the bulk coolant temperature in reactor tank. When passing the active core, the coolant is heated, leaving the upper grid plate holes with the density ρ_N and temperature T_N, which is higher than the temperature of surrounding bulk water. Due to this density difference buoyant head is created, providing the driving force for natural circulation.

In *TRISTAN* the heat source in fuel element is input data and axial power distribution is specified as chopped cosine. The coolant in a coolant channel is treated as a single phase system, which means that the coolant temperature is below the water boiling point and no change in enthalpy because of evaporation occurs. On the other hand the possibility of localized nucleate boiling at the clad surface is taken into account.

On the basis of coolant channel geometry and heat source data *TRISTAN* calculates the temperature of coolant, flow velocity and *DNB* ratio as a function of height. Iterative procedure is applied. Material properties like water density, specific heat, thermal conductivity, viscosity and boiling temperature are built into the program. Data are taken from steam tables [1] or Physics handbook [2] and tabulated in dependence of coolant temperature and pressure. Automatic calculation is also provided for different friction factors.

2 Physical model for natural convection cooling

Steady-state flow in a coolant channel is governed by the equation [3], [4]:

$$\Delta p_{ln} + \Delta p_{lr} + \sum_{i=1}^{N} \Delta p_i + \Delta p_{ur} + \Delta p_{un} + \Delta p_{chimney} = \rho_0(T_0) \cdot g \cdot H, \qquad (1)$$

where the left-hand members represent the pressure drops through the flow channel due to entrance, exit, friction, acceleration, and gravity losses, and the right-hand member represents the driving pressure due to the static head in the pool. Δp_{ln} and Δp_{un} denote the pressure drops at the entrance and exit, Δp_{lr} and Δp_{ur} pressure drops in lower and upper reflectors and $\Delta p_{chimney}$ pressure drop in real or virtual chimney. The active core is divided into N equidistant axial segments. Pressure drops Δp_i are calculated for each axial segment i and summed over all segments. H is the coolant channel height:

$$H = h_{ln} + h_{lr} + \sum_{i=1}^{N} \Delta h_{c,i} + h_{ur} + h_{un} + h_{chimney} \qquad (2)$$

and is the sum of contributions of lower element end-fixture, lower reflector, N axial segments of active core, and upper reflector, upper element end-fixture and chimney.

Since for a given channel height and bulk coolant temperature the static driving pressure is fixed, pressure drops on the left-hand side, which depend on the flow rate, are defined in iterative procedure. Initial flow velocity is assumed and pressure drops over the channel are evaluated. The result of left-hand side of equation (1) is then compared with the right-hand side until the equality is achieved.

With given flow velocity pressure losses in unheated regions can be calculated from Bernoulli equation. Details about these contributions are given in the manual [5] or in different text books [4], [3]. Pressure drop in axial segment i of active core is calculated as:

$$\Delta p_i = \frac{1}{2}\rho_i v_i^2 - \frac{1}{2}\rho_{i-1} v_{i-1}^2 + \rho_i g h_{c,i} - \rho_{i-1} g h_{c,i-1} + \frac{f_{fric}^i \cdot \rho_i \cdot v_i^2 \cdot (h_{c,i} - h_{c,i-1})}{2 \cdot d_h}. \qquad (3)$$

ρ_i is coolant density in i-th segment, g is gravitational constant, d_h hydraulic diameter of coolant channel, f_{fric}^i friction factor and v_i flow velocity in segment i and $h_{c,i}$ the dimension from the bottom of active core to the top of axial segment i.

Friction factor f^i_{fric} depends on regime by which the heat is transferred from fuel element to the coolant. If cooling is provided by convection, convective friction factor is used, if it is provided by nucleate boiling, boiling friction factor is applied [5]. At each axial increment the test is performed to establish the heat transfer regime and the appropriate friction factor is selected automatically.

In each heated axial increment the initial flow velocity is corrected for the density difference in that increment:

$$v_i = v_{i-1} \cdot \frac{\rho_{i-1}(\overline{T}_{i-1})}{\rho_i(\overline{T}_i)}. \qquad (4)$$

$\overline{T_i}$ is average coolant temperature of axial segment i, calculated from heat generation at that segment.

In reactor, operating at P_{therm}, the energy, released by one fuel element in time interval Δt_i and interval Δz_i around elevation z_i, is:

$$\Delta E(z_i) = p(z_i) \cdot \Delta z_i \cdot \Delta t_i, \qquad (5)$$

where $p(z_i)$ is linear power per element in W/cm. It is assumed, that axial linear power distribution is chopped cosine.

The volume unit of coolant, traveling through the channel with velocity $v(z)$, up to the elevation z_n receives the energy $E(z_n)$:

$$E(z_n) = \kappa \cdot \sum_{j=1}^{n} p(z_j) \cdot \Delta z_j \cdot \frac{\Delta z_j}{v_j}. \qquad (6)$$

κ is number of fuel elements, heating one coolant channel (in case of triangular channel $\kappa = 0.5$).

In single phase approximation all generated energy is used for heating the coolant.

$$E(z_n) = m_n \cdot c_{p,n} \cdot \Delta T_n. \qquad (7)$$

m_n is mass of coolant volume unit around z_n and $c_{p,n}$ temperature dependent coolant heat capacity at the same elevation. Taking into account that mass flow rate ϕ_m along the channel is constant:

$$\phi_m = \rho_0 \cdot S_{in} \cdot v_0 \qquad (8)$$

(9)

then the temperature increment at interval i can be expressed by:

$$\Delta T_i = \frac{\kappa \cdot p(z_i) \cdot \Delta z_i}{\phi_m \cdot c_{p,i}} \qquad (10)$$

$$T_i = T_{i-1} + \Delta T_i. \qquad (11)$$

From average coolant temperature at given elevation $\overline{T_i}$ water density is estimated from tabulated values. The value $\rho_i(\overline{T_i})$ is then corrected for the void fraction α at that segment, which is tabulated in the code in dependence on heat flux from fuel rod j_{out} (W/cm^2), temperature difference between water boiling point and average coolant temperature and flow velocity:

$$\rho_{pi} = \rho_i \cdot (1 - \alpha_i). \qquad (12)$$

If void fraction is different from zero, than ρ_i is set to ρ_{pi} and the calculation of v_i, α_i and ρ_{pi} is repeated until the stable operating point is found.

The calculation of *DNB* ratio in each axial segment is also performed, defined as the ratio between the critical heat flux j_{crit} and real heat flux j_{out} from fuel element:

$$DNBR_i = \frac{j_{i,crit}}{j_{out}(z_i)}. \qquad (13)$$

Critical heat flux is taken directly from Safety report [7] or calculated by Bernath correlation [3].

3 Verification of the code

The code was tested by comparing the results of *TRISTAN* to the calculated values of coolant temperature in dependence of power density in *TRIGA Mark II* in the natural convection operating mode, calculated by "General Atomics" and given in Safety Report [6]. Both results are in good agreement (the differences are less than 3-4 %), if power density is not exceeding 20 kW/element. For higher power densities the model of *TRISTAN* becomes inadequate and the results are unreliable.

An attempt was also made to verify the code by comparing its results to some measured values of coolant temperature. The experiment was performed at *TRIGA Mark II* reactor in Ljubljana. Axial dependence of coolant temperature was measured at several locations in the core by special termocouple, fixed at the end of long wire. Thermocouple was inserted into the core through the special holes for flux measurements in the upper grid plate and than moved along the coolant channel. Unfortunately the average power per element at full reactor power (250 kW) is only about 3.5 kW/element, which is not sufficient for full qualification of the code. However, the of results in this narrow range are in good agreement with the calculations (about 5 % differences), but for complete verification similar measurements should be performed at *TRIGA* reactor with higher nominal power.

4 Parametric study of natural convection cooling of TRIGA core

Nominal power of *TRIGA Mark II* reactor in Ljubljana is 250 kW. The possibility of upgrading power level was studied including also the termohydraulic analysis. The natural convection cooling of *TRIGA* core under various conditions was analysed using the computer program *TRISTAN*. The analysis was performed as parametric study. By varying most important parameters like power per element, inlet coolant temperature and coolant channel geometry different operating conditions were calculated.

Power per element varied from 2 kW to 20 kW. Inlet water temperature varied from 15°*C* to 35°*C*. The range includes all normally occurring water temperatures in our reactor tank. In analysis the coolant channel cross-sectional area was also varied since in reactor core with cylindrical geometry there are significant differences between the smallest and the biggest coolant channel (more than 100 %).

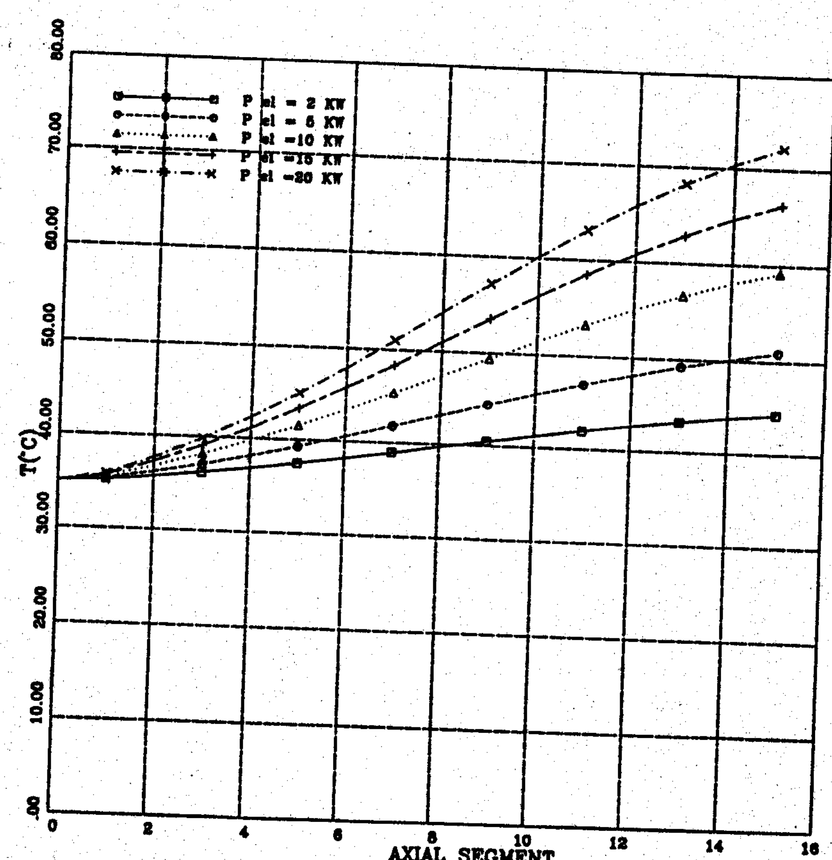

Figure 2: Axial coolant temperature distribution in average coolant channel for different power levels and inlet water temperature 35°*C*. The core was divided into 15 axial segments. No virtual chimney.

6

The influence of virtual chimney above the core on coolant temperature was also tested. Since its height is not known, the calculations at different power and at highest inlet water temperature were repeated for different heights of chimney: 0, 25 cm, 50 cm and 75 cm.

The results of these calculations show, that the highest coolant temperatures are achieved with the highest inlet water temperature and no virtual chimney. Coolant temperature distributions along the coolant channel with average cross-sectional area, no chimney and inlet water temperature 35°*C* are presented in Fig. 2. Corresponding coolant velocities varied from ~ 10 cm/s to 24 cm/s. *DNB* ratio as a function of channel height for this case is given in Fig. 3.

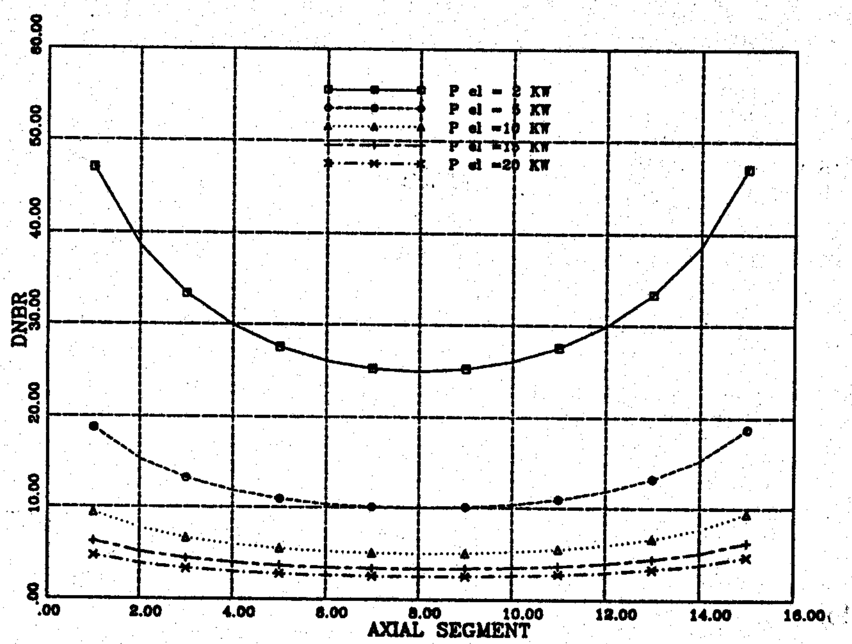

Figure 3: Axial dependence of DNBR for different powers per element. Inlet water temperature is 35°*C*, no virtual chimney. DNBR is calculated from critical heat flux 137 W/cm^2.

From these results it is seen that the highest coolant temperature at the top of coolant channel is far below the boiling point and *DNB* ratio above 2 even in case of inlet water temperature 35°*C* and power 20 kW per element. This is true even if the coolant channel with average cross-sectional area is replaced with the smallest one. It can be concluded that the efficiency of natural convection cooling in *TRIGA Mark II* reactor is sufficient even in case 20 kW per element and high inlet water temperature 35^oC.

5 Conclusions

A computer code for thermohydraulic analysis of *TRIGA* research reactors cooled by natural convection, which was developed at "J. Stefan" Institute, has been presented. The advantage of the code is that it runs on any *PC* and is very simple to use.

The program can be applied for calculating the natural convection flow parameters in low power reactors, where power level is not exceeding 1-2 MW. As an example some results of parametric study of natural convection cooling in *TRIGA Mark II* reactor in Ljubljana are given.

References

[1] M. Oprešnik, M. Opara, Termodinamične tabele in diagrami, Fakulteta za strojništvo, Ljubljana 1974

[2] Physics Handbook, American Institute of Physics Handbook, McGraw-Hill Book Company, 1982

[3] L. S. Tong, J. Weisman, Thermal Analysis of Pressurized Water Reactors, American Nuclear Society, 1970

[4] M.M. El-Wakil, Nuclear Heat Transport, An Intext Publisher, London, 1971

[5] I. Mele, B. Žefran, TRISTAN - A Computer program for Calculating Natural Convection Flow Parameters in TRIGA Core, Program Manual, IJS-DP-6548 (1992)

[6] Safety Analysis Report for the 3000 kW Forced-Flow TRIGA Mark II Reactor for the Bangladesh Atomic Energy Commission, GA E-117-990, July 1981

[7] Varnostno poročilo za reaktor *TRIGA Mark II* v Podgorici, Revizija 3, IJS-DP-5823, Junij 1992