STHIRP-1: COMPUTER CODE FOR THERMAL-HYDRAULIC ANALYSIS OF RESEARCH REACTOR CORE

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Abstract:. *The computer program STHIRP-1, which fundaments are descript in this work, uses the principles of the subchannels analysis and has the capacity to simulate, under steady state and transient conditions, the thermal and hydraulic phenomena which occur inside the core of a water-refrigerated research reactor under a natural convection regime. The models and empirical correlations necessary to describe the flow parameters which can not be descript by theoretical relations were selected according to the characteristics of the reactor operation. To demonstrate the analytical capacity of STHIRP-1, comparisons were made between the calculated and measured results in the research reactor TRIGA IPR-R1 of CDTN/CNEN. Thermal hydraulic calculations were carried out for the reactor operating at the steady-state power levels of 50, 70 and 100 kW. The results indicate that the program reproduces the experimental data with good precision.*

Keywords: *subchannel analysis, subchannel codes, research reactors, TRIGA research reactor*

1. INTRODUCTION

The STHIRP-1 (Veloso, 2004) computer code uses the principles of the subchannels analysis and has the capacity to simulate, under steady state and transient conditions, the thermal and hydraulic phenomena which occur inside the core of a water-refrigerated research reactor under a natural convection regime.

Although the primary objective is the calculation of research reactors, the formulation used to describe the fluid flow and the thermal conduction in the heater elements is sufficiently generalized to extend the use of the program for applications in power reactors and other thermal systems with the same features represented by the program formulations.

IPR-R1, a TRIGA Mark I reactor manufactured by General Atomic (1959) for the Institute for Radioactive Research (IPR), currently called Nuclear Technology Development Center (CDTN) was dedicated on November 11, 1960. Since its start up, the reactor has been continuously operated for the following purposes: production of radioisotopes for educational and scientific use, scientific experiments, training of research and power reactor operators, experiments with materials and minerals, and neutron activation analysis

The reactor core is located at the bottom of a water filled cylindrical tank with internal diameter of 1.92m and the depth of 6.625m. The height of the water column is of approximately 6.1 m. The reactor core is surrounded by an annular graphite neutron reflector and consists of an array of fuel elements, graphite dummy elements and control rods.

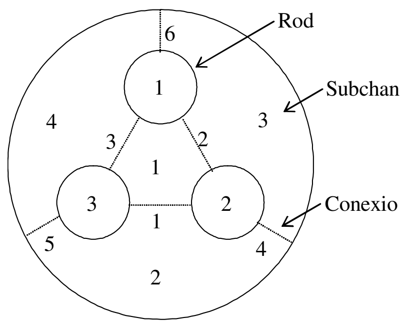
Reactor cooling occurs predominantly by natural convection of pool water, with the driving forces supplied by the reactor itself as it transfers heat to the coolant which creates a buoyant head. The heat generated through fission in the fuel material is conducted through the fuel, through the fuel-cladding interface and through the cladding to the coolant. The removal of the heat generated in the core for nuclear fissions is effected by forced circulation of the demineralized water through an external heat exchanger

Thermal hydraulic calculations were carried out for the IPR-R1 TRIGA reactor (CDTN/CNEN) operating at steady-state power levels of 50, 70 and 100 kW. The models and empirical correlations necessary to describe the flow phenomena which can not be described by theoretical relations were selected according to the characteristics of the reactor operation. The analytical capacity of the code will be tested against measurements for subchannel exit and fuel rod temperatures taken in the IPR-R1 TRIGA core at several reactor power levels.

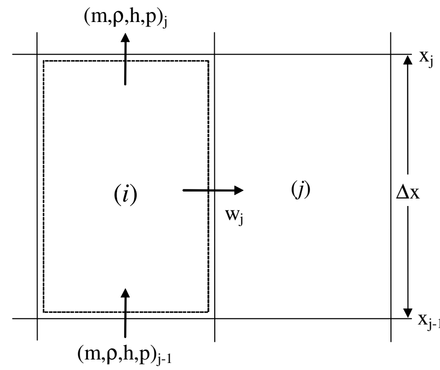
2. BASIC PRINCIPLES OF SUBCHANNELS ANALYSIS

In the subchannel techniques the rod cluster, cooled by a fluid passing axially along the rods, is considered to be subdivided into a number of parallels interacting flow subchannels between the rods. The channel length is divided into a number of smaller intervals for calculations purposes.

The cross-sectional areas of subchannels are conventionally defined by lines joining rods centers (Figure 1a). To each lateral connection k corresponds a pair of adjacent subchannels [i(k), j(k)]. The subchannels are opened laterally so that the transport of mass, energy and momentum can occur between adjacent subchannels. The conservation equations can be derived in differential form by considering the transport in and out of a control volume (Figure 1b).



(a) Cluster Rods Section



(b) Control Volume

Figure 1. Subchannel configuration

By making use of some simplifying assumptions and appropriate boundary conditions, the differential equations of mass, energy, and momentum conservation are written in finite-difference form and solved numerically to give the distributions of fluid density (ρ), specific enthalpy (h), pressure (p), axial mass flow rate (m), and lateral mass flow rate per unit length (w).

The finite-difference equations for the conservation of mass, energy, and momentum are summarized below:

<u>Continuity equation</u>

$$m_{i,j} = m_{i,j-1} - \Delta z \sum_{k \in i} e_{ki} w_{k,j} - \frac{\Delta z}{\Delta t} A_{i,j} (\rho_{i,j} - \hat{\rho}_{i,j}) \tag{1}$$

<u>Energy equation</u>

$$h_{i,j} = \left(1 + \frac{\Delta z}{\Delta t} \frac{1}{u''_{i,j}} \right)^{-1} \left[h_{i,j-1} + \frac{\Delta z}{\Delta t} \frac{1}{u''_{i,j}} \hat{h}_{i,j} + \left(\frac{\Delta h}{\Delta z} \right)_{i,j} \Delta z \right] \tag{2}$$

<u>Axial momentum equation</u>

$$p_{i,j} = p_{i,j-1} + F_{i,j} \Delta z + \sum_{k \in i} (r_{ik} w_k)_j \Delta z \tag{3}$$

<u>Lateral momentum equation</u>

$$w_{k,j} = D^{-1}_{k,j} Q_{k,j} + (s / \ell) D^{-1}_{k,j} (p_{i(k)} - p_{j(k)})_{j-1} \tag{4}$$

<u>Pressure field equation</u>

Substitution of Eq. (4) into Eq. (3) results the following system of linear equations for the subchannel pressure field:

$$[I + M_j] \{ p_{j-1} \} = \{ p_j \} - \{ b_j \} \Delta z \tag{5}$$

where [I] denotes the identity matrix.

This set of equations is closed by defining an additional equation for the physical state of the coolant and empirical correlations for turbulent mixing coefficients, subcooled steam quality, void fraction, friction factors and heat transfer coefficients. STHIRP-1 uses the same fluid flow conservation equations as described by Rowe (1973).

The boundary conditions for forced convection cooling are: (1) the core outlet pressure must be uniform; (2) the total inlet flow to the core is fixed. It is thus necessary to find an inlet velocity distribution which will satisfy these boundary conditions. In the case of natural convection it is necessary to reformulate these boundary conditions because the mass flow rates are governed by the fluid behavior through the channel.

2.1. Physical Model for Natural Convection Cooling

In the TRIGA reactors the coolant channel extends from the bottom grid plate to the top grid plate. The coolant enters the channel through special holes in the bottom grid plate, passes the unheated lower part of fuel element, enters then the active part of the core, passes the upper axial reflector and top end-fixture and leaves the channel through the holes in the upper grid plate.

Inlet coolant temperature T_1 and density ρ_1 are equal to the bulk fluid temperature and density in the reactor tank. When passing through the active core, the coolant is heated and then leaves the upper grid plate with lower density (ρ_{N+1}) and higher temperature (T_{N+1}) in relation to the surrounding bulk water. Due to this density difference buoyant head is created providing the driving force for natural circulation. Countering this force are the contraction and expansion losses at the entrance and exit to the subchannel, the acceleration and potential energy losses, and the friction losses in the coolant subchannel itself. The height of the virtual chimney above the top grid plate depends on the reactor power. At the top of the chimney, the water temperature and density are equal to the bulk values and it is assumed that the fluid is locally motionless.

Figure 2 illustrates schematically the natural convection process in a typical subchannel in the TRIGA reactor core.

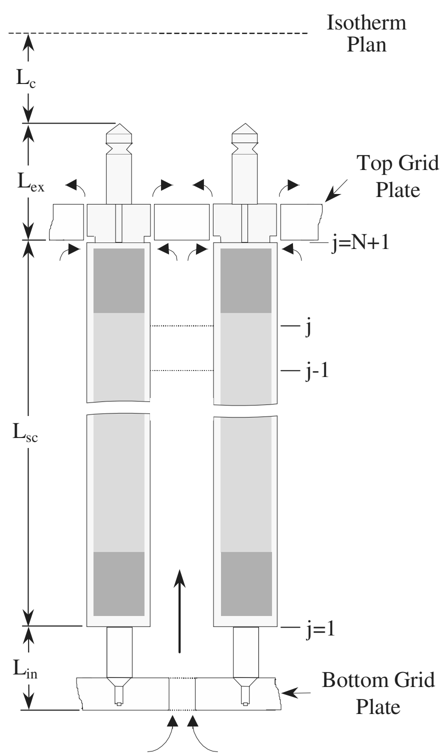


Figure 2. TRIGA coolant subchannel

Steady state flow in a coolant channel is governed by the equation

$$\sum_{j=2}^{N+1} (\Delta p_i)_j = \rho_\infty (T_\infty) g (L + L_C) \tag{6}$$

where $(\Delta p_i)_j$ represent the total pressure drop at axial level j, ρ_∞ is the average water density at temperature average T_∞, L is the total length of the core and L_C is the chimney length

For a subchannel i, it is assumed that the difference between the total pressure drop and the gravitational pressure drop is proportional to square of inlet mass flow rate:

$$\Delta p_i - \Delta p_{gi} = C(m_i')_1^2 \tag{7}$$

For a new inlet mass flow rate m' that gives a specified pressure drop, Eq. (7) becomes

$$\Delta p_{esp} - \Delta p_{gj} = C(m_i')_1^2 \tag{8}$$

By elimination of the constant C, results:

$$(m_1')_1 = (m_i)_1 \left(\frac{\Delta p_{esp} - \Delta p_{gi}}{\Delta p_i - \Delta p_{gi}} \right)^{1/2} \tag{9}$$

The pressure drop Δp_{esp} is just the second member of Eq. (6):

$$\Delta p_{esp} = \rho_\infty (T_\infty) g (L + L_C) \tag{10}$$

The pressure drop is calculated in each axial segment. The contributions are summed over all segments in order to give the total pressure drop:

$$\Delta p_i = \sum_{j=2}^{N+!} (\Delta p_i)_j = - \sum_{j=2}^{N+1} [(p_i)_j - (p_i)_{j-1}] \tag{11}$$

The gravitational term is given by:

$$\Delta p_{gi} = \sum_{j=2}^{N+1} (\rho g \Delta x)_j + \frac{1}{2} (\rho_{N+1} + \rho_\infty) g L_C \tag{12}$$

The pressure drops at the inlet and outlet of the core can be determined whit the relations:

$$\left(\Delta p_{si} \right)_1 = \frac{1}{2} (\varsigma_i)_1 \frac{(v_i')_1 (m_j)_1^2}{(A_j)_1^2} \tag{13}$$

$$\left(\Delta p_{si} \right)_{N+1} = \frac{1}{2} (\varsigma_i)_{N+1} \frac{(v_i')_{N+1} (m_j)_{N+1}^2}{(A_j)_{N+1}^2} \tag{14}$$

where v' is the specific volume, A is the subchannel flow area and $(\varsigma)_1$ and $(\varsigma)_{N+1}$ represent the hydraulic resistance coefficients in the inlet and outlet of the subchannels.

Concerning the numerical procedure, in a typical calculation, coolant variables are computed simultaneously for all subchannels by starting at the bottom of the rod bundle and sweeping downstream to the bundle exit. Iterative calculations are carried out until the convergence of the flow solution is obtained. Convergence is achieved when the changes in the distributions of both axial and lateral flows are less than prescribed tolerances. If overall pressure drop is the boundary condition (e.g. for natural convection problems), the inlet flow rates are adjusted after each iteration to force the calculated pressure drop to the desired value.

3. THERMAL-HYDRAULIC CALCULATIONS

TRIGA fuel was developed around the concept of inherent safety. The U-ZrH alloy is a homogeneous mixture of uranium and zirconium, where the zirconium acts as fuel-dispersion agent and hydrogen is the moderator: zirconium fixes hydrogen in the alloy. A core composition was sought that had a large prompt negative temperature coefficient of reactivity so that if all the available excess reactivity were suddenly inserted into the core, the resulting fuel temperature would automatically cause the power excursion to terminate before any core damage resulted.

The IPR-R1 core is loaded with two types of fuel elements: (1) elements with aluminum clad and (2) elements with stainless steel clad. In the aluminum clad element, the percentage in weight of uranium, zirconium and hydrogen in the mixture are, respectively, 8%, 91% and 1% (U-$ZrH_{1.0}$). The fuel mixture in the stainless-steel clad element contains 8.5% in weight of uranium, 89.9% of zirconium and 1.6% of hydrogen (U-$ZrH_{1.6}$). The uranium is 20% enriched in ^{235}U.

A cross-sectional view of the IPR-R1 TRIGA reactor core discretized in subchannels is shown in Figure 3. The reactor's neutronic parameters of vital importance for thermo-hydraulic analysis are the power distribution and power-peaking factors, and axial and radial power distribution inside the fuel rod. The radial power factors were calculated by Dalle (1998) using the codes WIMSD4C and CITATION. The axial power profile was assumed to be described by a chopped cosine curve with peak-to-average ratio equal to 1.25.

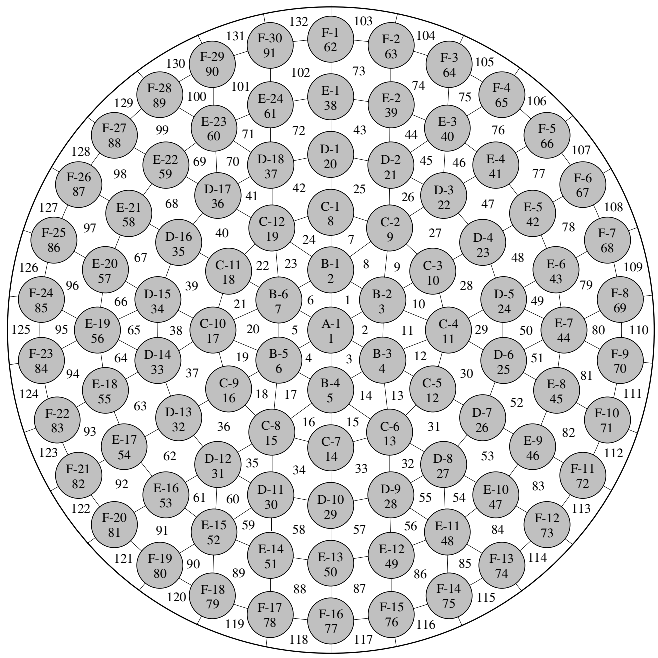


Figure 3. IPR-R1 core subchannel layout

The solution of the conservation equations requires the specification of constitutive relations for the thermodynamic properties of the fluid transport, such as void fraction, heat transfer coefficients, pressure drop, hydraulic resistances, mixing coefficients etc. The parameters and correlations listed in Table 1 had been used in the simulations of reactor with STHIRP-1.

Table 1. Calculation parameters and models

Laminar friction factor	$f = 64\ Re^{-1}$
Turbulent friction factor	$f = 0.316\ Re^{-0.25}$
Transverse hydraulic resistance	0.5
Ratio of gap width to crossflow length (s/ℓ)	0.5
Turbulent momentum factor, f_t	1.0
Turbulent mixture coefficient	$\beta = 0.0062\ (D/s)\ Re^{-0.1}$ Rowe (1967)
Bulk void fraction	Jordan and Leppert model (1962)
Subcooled steam quality	Levy model (1967)
Two-phase friction multiplier	Jordan and Leppert model (1962)

The exit temperatures of the subchannels 8, 11, 18, 28, 36, 46, 52 and 65 calculated by program STHIRP-1 for the 50, 70 and 100 kW power levels are given in Figures 4, 5 and 6.

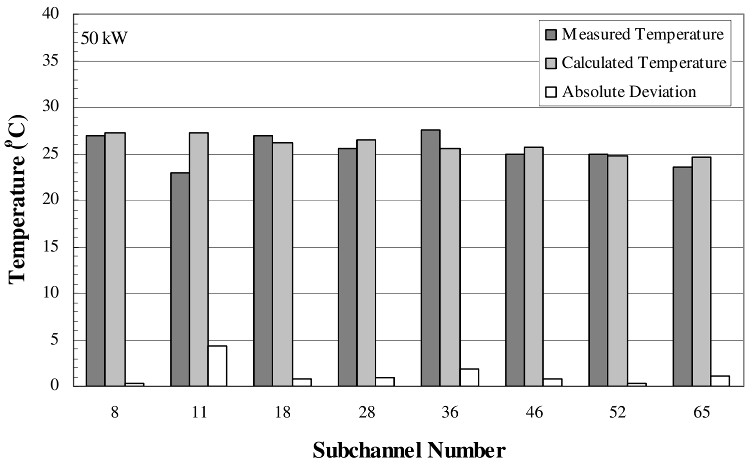


Figure 4. Measured and predicted channel exit temperatures at 50 kW

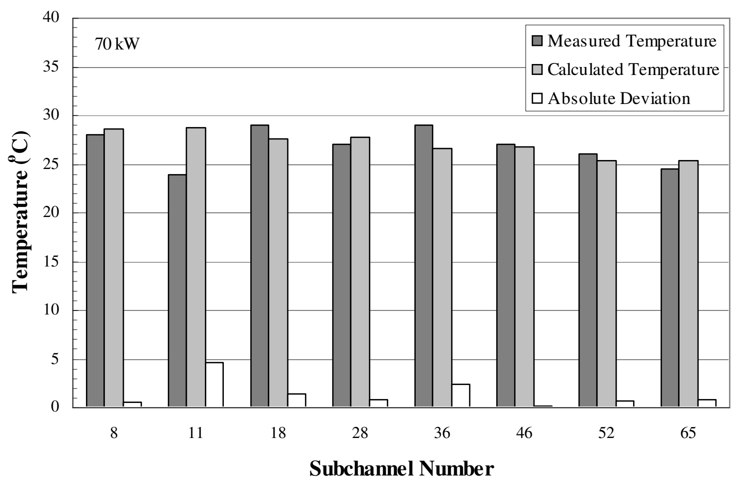


Figure 5. Measured and predicted channel exit temperatures at 70 kW

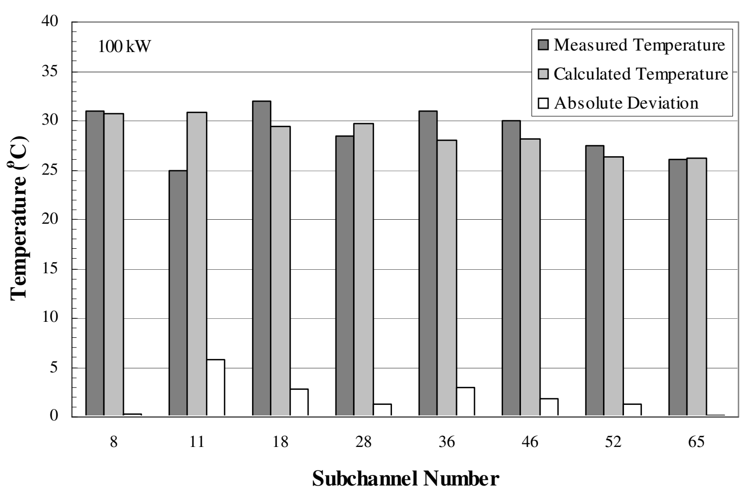


Figure 6. Measured and predicted channel exit temperatures at 100 kW

In the calculations the water inlet temperature was taken as uniform and equal to 23^{o}C, the same as the arithmetic mean of the temperatures measured at the inlets of the mentioned eight subchannels. Analysis of figures reveals a good concordance between measured and calculated temperatures.

Deviations between calculated and measured exit temperatures for suchannels 8, 28, 52, and 65 were found to be of the same order of magnitude of the experimental errors. The greatest discrepancies occur consistently for subchannel 11. Since subchannels 8 and 11 were symmetrical, as shown in Figure 3, it was to be expected that both the subchannels presented comparable exit temperatures. In principle, at least from the theoretical viewpoint, there is no way to justify such significant differences.

4. CONCLUSIONS

A computer code for thermal hydraulics analysis of research reactors cooled by natural convection has been presented. The subchannel approach have been used for the thermal hydraulic evaluation of IPR-R1 reactor at 50, 70 and 100 kW power level. The good agreement between the STHIRP-1 predictions and the in-core temperature measurements reveal the soundness of the proposed subchannel flow model.

5. ACKNOWLEDGEMENTS

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