THERMAL-HYDRAULIC FORTRAN PROGRAM FOR STEADY-STATE CALCULATIONS OF PLATE-TYPE FUEL RESEARCH REACTORS

by

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The safety assessment of research and power reactors is a continuous process covering their lifespan and requiring verified and validated codes. Power reactor codes all over the world are well established and qualified against real measuring data and qualified experimental facilities. These codes are usually sophisticated, require special skills and consume a lot of running time. On the other hand, most research reactor codes still require much more data for validation and qualification. It is, therefore, of benefit to any regulatory body to develop its own codes for the review and assessment of research reactors.
The present paper introduces a simple, one-dimensional Fortran program called THDSN for steady-state thermal-hydraulic calculations of plate-type fuel research reactors. Besides calculating the fuel and coolant temperature distributions and pressure gradients in an average and hot channel, the program calculates the safety limits and margins against the critical phenomena encountered in research reactors, such as the onset of nucleate boiling, critical heat flux and flow instability. Well known thermal-hydraulic correlations for calculating the safety parameters and several formulas for the heat transfer coefficient have been used. The THDSN program was verified by comparing its results for 2 and 10 MW benchmark reactors with those published in IAEA publications and a good agreement was found. Also, the results of the program are compared with those published for other programs, such as the PARET and TERMIC.

Key words: research reactor, steady-state thermal-hydraulic calculation, safety parametars, Fortran program

INTRODUCTION

Research reactors (RR) are used all over the world for various purposes, *e. g.* research, experiments, education, training, radioisotope production, neutron radiography, material tests, *etc*. Most of these RR are open-pool, light-water cooled and material testing reactor (MTR) type fuel elements. They usually operate in the single-phase, liquid-water regime, at low pressure and temperature. A safe operation of these reactors requires some limits on the variables of the major process in order to protect the reactor barriers and prevent uncontrolled radioactivity releases. These limits are established during the design stage and should be verified at any reactor modification, upgrading, core conversion, and so on.

The review and assessment process by the regulatory body for nuclear facilities usually requires qualified codes covering the different fields of importance to safety. The qualification of these codes depends on the hazard that may result in case of a transient or accident in the said facility. For example, huge amounts of financial and human resources have been invested into the improvement and development of qualified codes for power reactors. On the other hand, available codes for the review and assessment of research reactors still

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require much more effort for their qualification. In addition to this, the in-depth concept of defense requires that the safety analyses are independently assessed by the operating organization and by the regulatory body [1]. Consequently, it is preferable for the regulatory body to have its own codes or to use internationally qualified codes, rather than those used by the designer or the operating organization.

The verification of RR design goals during normal operation is one of the regulatory body's functions. Most international thermal-hydraulic (TH) qualified codes applicable for RR safety analysis, such as PARET and the new versions of RELAP, are transient TH system codes [2, 3]. These codes are usually sophisticated and require skills during the preparation of their input deck, nodalization qualifications and output processing [4, 5]. Although such packages are accurate and have capabilities to simulate a wide range of TH transients, different research institutes have different approaches to the build-up of simplest transient programs [6-9]. It is, therefore, not rational to use such large packages for steady-state calculations. The present paper is focused on the build-up of a thermal-hydraulic Fortran program to be used in the safety assessment of plate-type fuel research reactors.

It is well known that the fuel clad represents the most important barrier against the release of radioactive materials [10]. To protect this barrier, from the thermal point of view, some safety limits on the variables of the said process have to be implemented. In addition, safety margins against the appearance of any critical/undesirable phenomena that may lead to a rapid increase in the cladding temperature and, consequently, damage in fuel elements are to be observed, also.

The most critical TH phenomena which, if approached too closely, will present a safety problem regarding fuel-plate integrity, are the departure from nucleate boiling (DNB) and the onset of flow instability (OFI). In addition to this, coolant velocity beyond which the fuel plates will collapse (critical velocity) and the core pressure drop, are among the safety parameters that are of utmost importance. Although the onset of nucleate boiling (ONB) does not correspond to any critical phenomenon, for design purposes, it is considered as a conservative constraint.

Flow instability (FI) is a critical phenomenon because the associated flow oscillations affect the local heat transfer characteristics and may induce premature burnout. Burnout heat flux occurs under unstable flow conditions well below the burnout heat flux under stable flow conditions (DNB). For practical purposes, the heat flux that leads to the OFI may be more limiting in the design of plate-type fuel elements than that of a stable burnout [11]. In addition, there are different types of FI encountered in heated, forced convection channels. The most common one encountered in RR, where the operating system pressure is low and the inlet coolant temperature is much lower than the saturation temperature, is the flow excursion or Ledinegg instability [11].

The proposed program is a Fortran-type program called THDSN that calculates the TH and safety parameters of plate-type RR during the build-up of steady-state conditions. In this program, the reactor core is represented by two channels, an average and a hot channel. An analytical solution of the energy equation is used to calculate the temperature and power distribution in the two channels. Well-known correlations for core pressure drop, ONB, DNB, OFI, and critical velocity are also established in this program. There are different correlations for the heat transfer coefficient and an access for the user to choose between them. This program is simple, fast and provides an oversight of all TH and safety parameters during the steady-state operation of research reactors. For sake of verification, best estimate safety limits and margins for a typical 2 MW or 10 MW MTR IAEA reactor were calculated at steady-state conditions and the results are compared with those published in IAEA TECDOC-233, Appendix A. The program was also verified against other TH codes, such as PARET and TERMIC.

MODELING

In this section, the different correlations used in the THDSN program to calculate the axial distribution of heat flux, coolant temperature, clad and fuel temperatures and core pressure drop are presented. Also, the correlations used to calculate the heat transfer coefficient, the heat flux at ONB, the heat flux at DNB, and the heat flux at OFI are introduced.

Heat flux

The core axial power distribution is considered to follow a cosine shape given by

$$q\ (i, z) \quad q_c(i) \cos\frac{\pi z}{l_e} \tag{1}$$

where $q'(i, z)$ is the power density of fuel at axial location z in channel i (i equals 1 for an average channel and 2 for a hot channel), and l_e is the extrapolated length. $q_c(i)$ is the maximum power density in channel i at half of the fuel plate length, where the axial coordinate (z) is put equal to zero and is given by

$$q_c(i) \quad F(i)\, q_a \tag{2}$$

where q_a is the core average power density which equals core power divided by the fuel meat volume. In an average channel $F(i)$ equals the axial power factor (F_A), in a hot channel $F(i)$ equals the nuclear power factor (F_{NUC}) which is the multiplication of the axial power factor F_A and the radial power factor F_R.

The heat flux axial distribution in channel i as a function of the local power density is given by

$$q(i,z) = \frac{q'''(i,z)}{SVR} \tag{3}$$

where SVR is the fuel meat surface to volume ratio.

Temperature distribution

At steady-state, the heat generated in the fuel is transferred to the coolant through the clad. The coolant axial temperature distribution in channel i is given by [12]

$$T_{\mathrm{f}}(i,z) = T_{\mathrm{f1}} + 0.001\frac{q_{\mathrm{c}}(i)A_{\mathrm{m}}l_{\mathrm{e}}}{\pi C_{\mathrm{p}}m_{\mathrm{ch}}}\left[\sin\frac{\pi z}{l_{\mathrm{e}}} \pm \sin\frac{\pi l_{\mathrm{h}}}{2l_{\mathrm{e}}}\right] \tag{4}$$

where T_{f1} is the core coolant inlet temperature, A_{m} – the fuel meat cross-section area (= $w_{\mathrm{m}}t_{\mathrm{m}}$), w_{m} – the meat width, t_{m} – its thickness, m_{ch} – the channel mass flow rate, l_{h} – the active fuel-plate length or heated length of the channel, and C_{p} – the coolant specific heat. The sign ± is for upward or downward core flow, respectively.

The clad surface axial temperature distribution (T_{c}) in channel i is given by [12]

$$T_{\mathrm{c}}(i,z) = T_{\mathrm{f}}(i,z) + \frac{q_{\mathrm{c}}(i)t_{\mathrm{m}}}{2h(i)}\cos\frac{\pi z}{l_{\mathrm{e}}} \tag{5}$$

where h is the heat transfer coefficient.

The meat centerline axial temperature distribution in channel i is [12]

$$T_{\mathrm{m}}(i,z) = T_{\mathrm{c}}(i,z) + 100.0\,q_{\mathrm{c}}(i)\cos\frac{\pi z}{l_{\mathrm{e}}}\left(\frac{t_{\mathrm{m}}^2}{8k_{\mathrm{m}}} + \frac{t_{\mathrm{m}}t_{\mathrm{c}}}{2k_{\mathrm{c}}}\right) \tag{6}$$

where t_{c} is the clad thickness, k_{m} – the meat thermal conductivity, and k_{c} – the clad thermal conductivity.

Mass flow rate

The total core coolant flow is divided into two parts. Active flow passes through the fuel elements and non-active flow passes through the by-pass channels, such as the control rod channels. Active core flow is determined by multiplying the total core flow in a factor less than one; in this study, this factor is 0.9. Therefore, the m_{ch} is equal to

$$m_{\mathrm{ch}} = \frac{F_{\mathrm{D}}W_{\mathrm{T}}\rho}{3600\,FPTN} \tag{7}$$

where F_{D} is a factor less than unity, W_{T} – the total core flow rate, and $FPTN$ – the total number of fuel plates.

Water density ρ is evaluated at the core inlet temperature.

Channel velocity V_{ch} is

$$V_{\mathrm{ch}} = 10^4\frac{m_{\mathrm{ch}}}{\rho A_{\mathrm{ch}}} \tag{8}$$

where A_{ch} is the channel cross-sectional area. Water density ρ is evaluated at the channel average temperature. In some cases, the channel velocity instead of the core flow is given as input data. In such cases, the channel mass flow rate is calculated from eq. (8) and the volume flow rate is calculated from eq. (7).

Pressure drop

Most plate-type research reactors were designed for single-phase flow under normal operation [11]. Core pressure losses are divided into two parts: pressure losses in fuel element channels and pressure losses in the fuel element nozzle. The channel pressure losses (ΔP_{ch}) are: entrance losses ΔP_{en}, friction losses ΔP_{f}, and exit losses ΔP_{ex} *i. e.*

$$\Delta P_{\mathrm{ch}} = \Delta P_{\mathrm{en}} + \Delta P_{\mathrm{f}} + \Delta P_{\mathrm{ex}} \tag{9}$$

These three parts are given in bar by [11]

$$\Delta P_{\mathrm{en}} = 10^{-5}K\frac{\rho V_{\mathrm{ch}}^2}{2}$$

$$\Delta P_{\mathrm{f}} = 10^{-5}\frac{4fL_{\mathrm{p}}}{D_{\mathrm{y}}}\frac{\rho V_{\mathrm{ch}}^2}{2}$$

$$\Delta P_{\mathrm{ex}} = 10^{-5}\left(1-\frac{V_{\mathrm{o}}}{V_{\mathrm{ch}}}\right)^2\frac{\rho V_{\mathrm{ch}}^2}{2}$$

where K is the entrance losses coefficient considered to be 0.5, L_{p} – the total length of the fuel plate, V_{o} – the water velocity in fuel element plenums, and D_{y} – the channel hydraulic diameter

$$D_y = \frac{4\ \text{cross-section area}}{\text{weted perimeter}}$$

The nozzle pressure losses (ΔP_{noz}) consist of two parts: entrance losses (ΔP_{en}) and friction losses (ΔP_{f}) *i. e.*

$$\Delta P_{\mathrm{noz}} = \Delta P_{\mathrm{en}} + \Delta P_{\mathrm{f}} \tag{10}$$

These two parts are given in bar by

$$\Delta P_{\mathrm{f}} = 10^{-5}\frac{4fL_{\mathrm{noz}}}{D_{\mathrm{y}}}\frac{\rho V_{\mathrm{noz}}^2}{2}$$

$$\Delta P_{\mathrm{en}} = 10^{-5}K\frac{\rho V_{\mathrm{noz}}^2}{2}$$

where V_{noz} is the coolant velocity in the fuel element nozzle, K – a constant equal to one, and L_{noz} – the nozzle length.

The friction factor f for turbulent flow in smooth channels is calculated as follows [13]:

$$f \quad 0.316\,\mathrm{Re}^{\;\;0.25} \qquad \mathrm{Re} \quad 1.0 \;\; 10^{5}$$

$$\frac{1}{\sqrt{f}} \quad 2\log_{10}(\mathrm{Re}\sqrt{f}) \;\; 0.8 \qquad \mathrm{Re} \quad 1.0 \;\; 10^{5} \tag{11}$$

The total core pressure drop (ΔP_{cor}) is given by

$$\Delta P_{\mathrm{cor}} \quad \Delta P_{\mathrm{ch}} \quad \Delta P_{\mathrm{noz}} \quad \Delta P_{\mathrm{st}} \tag{12}$$

The plus sign in the downward flow and the negative sign in the upward flow. ΔP_{st} is the static pressure difference between the core inlet and outlet, given by

$$\Delta P_{\mathrm{st}} \quad 10^{\;\;7}\gamma(L_{\mathrm{p}} \quad L_{\mathrm{noz}}) \tag{13}$$

where γ [Nm^{-3}] is the water specific weight .

Heat transfer coefficient

The program contains different correlations for calculating the heat transfer coefficient (h) and the user has options to choose any of them. The flow in the RR at normal operation is usually a turbulent flow; consequently, the correlations mentioned here cover this flow regime. These correlations are summarized as following:

– Dittus-Boelter correlation [14]

There are two forms of this correlation. The first one is the usual formula which is used when the structure is cooled by a fluid, as in reactor core cooling, and takes the form

$$\mathrm{Nu} \quad 0.023\,\mathrm{Re}^{0.8}\,\mathrm{Pr}^{0.4} \tag{14}$$

The second one is used when the structure is heated by a fluid, as in a heat exchanger primary side, and takes the form

$$\mathrm{Nu} \quad 0.023\,\mathrm{Re}^{0.8}\,\mathrm{Pr}^{0.3} \tag{15}$$

where

$$\mathrm{Nu} \quad \frac{hD_{\mathrm{h}}}{k}, \;\; \mathrm{Re} \quad \frac{G_{\mathrm{ch}}D_{\mathrm{h}}}{100\mu}, \;\; \mathrm{Pr} \quad 10^{3}\,\frac{\mu C_{\mathrm{p}}}{k},$$

k is the water thermal conductivity [Wm^{-1}°C^{-1}), μ – the water viscosity [Pa s], and G_{ch} – the channel mass flux ($=\rho V_{\mathrm{ch}}$).

All the water properties in these two correlations are evaluated at bulk temperature. It is clear that the heat transfer coefficient predicted by correlation (15) is lower than that predicted by correlation (14). So, for more conservative calculations, some references such as TECDOC-233 [11] used correlation (15) for calculating the heat transfer coefficient into the core channels.

– Sieder-Tate correlation [12]

$$\mathrm{Nu} \quad 0.023\,\mathrm{Re}^{0.8}\,\mathrm{Pr}^{0.4}\;\; \frac{\mu_{\mathrm{w}}}{\mu}^{\;\;0.14} \tag{16}$$

All properties are evaluated at bulk temperature, except μ_{w} which is evaluated at surface temperature.

– Colburn correlation [14]

$$\mathrm{Nu} \quad 0.023\,\mathrm{Re}^{0.8}\,\mathrm{Pr}^{0.33} \tag{17}$$

All properties are evaluated at film temperature (arithmetic mean between the fluid bulk and surface temperatures), except for the specific heat which is evaluated at fluid bulk temperature.

– Petukhov correlation [15]

$$\mathrm{Nu} \quad \frac{\frac{f}{8}\mathrm{Re}\,\mathrm{Pr}}{1.07 \quad 12.7\sqrt{\frac{f}{8}}(\sqrt[3]{\mathrm{Pr}^{2}} \quad 1)} \;\; \frac{\mu_{\mathrm{b}}}{\mu_{\mathrm{w}}}^{\;\;0.11} \tag{18}$$

where

$$f = \frac{1}{(1.82\log_{10}\mathrm{Re} \quad 1.64)^{2}}$$

All properties are evaluated at film temperature (arithmetic mean between the fluid bulk and surface temperatures), except for μ_{b} and μ_{w} which are evaluated at bulk and wall temperature, respectively.

Onset of nucleate boiling

The ONB is not a limiting criterion in the design of a fuel element, but is considered as a conservative statement for design purposes. The heat flux that initiates ONB is frequently used as a thermal design constraint. This constraint is determined by equating the clad surface temperature at the ONB which is calculated from the Bergles and Rohsenow correlation with the clad surface temperature calculated at the hot channel's worst condition [11]. Therefore, the heat flux (q_{ONB}) at the ONB, is calculated from the following correlation

$$T_{\mathrm{sat}} \quad \frac{5}{9}\;\; \frac{9.23F_{\mathrm{A}}F_{\mathrm{R}}q_{\mathrm{av}}}{P^{1.156}}^{\frac{P^{0.0234}}{2.16}}$$

$$T_{\mathrm{f1}} \quad \frac{20F_{\mathrm{R}}q_{\mathrm{av}}l_{\mathrm{h}}w_{\mathrm{h}}}{G_{\mathrm{ch}}t_{\mathrm{w}}C_{\mathrm{p}}w} \quad \frac{F_{\mathrm{A}}F_{\mathrm{R}}q_{\mathrm{av}}}{h} \tag{19}$$

where w_h is the channel heated width, w – the channel width, P – the absolute pressure, T_{f1} – the core inlet temperature, and q_{av} – an average heat flux.

Iterations are made on q_{av}, starting from the average heat flux at nominal power (q_a). The value of q_{av} at which the two sides are equal is the average heat flux at which the onset starts (q_{ONB}). The ratio between q_{ONB} and q_a is denoted by ONBR. This ratio is usually used to define the margin against ONB and its value must be greater than unity. The value of q_{ONB} results from eq. (19) is lower than the actual value because it is calculated at the channel's worst conditions. Consequently, its value will be multiplied by 1.15 [11].

Departure from nucleate boiling

Two correlations are recommended for the calculation of burnout heat flux (critical heat flux) at research reactor conditions of low pressure and exit sub-cooling. These correlations are Mirshak and Labuntsov correlations [11]. These two correlations are:

– Mirshak correlation

$$q_c \quad 151.0(1 \quad 0.1198 V_{ch}) \quad (1 \quad 0.00914\Delta T_{sub})(1 \quad 0.19P) \tag{20}$$

– Labuntsov correlation

$$q_c \quad 145.4\,\theta(P) \;\; 1 \quad \frac{2.5V_{ch}^2}{\theta(P)}^{0.25} \quad 1 \quad \frac{15.1 C_p \Delta T_{sub}}{\lambda P^{0.5}} \tag{21}$$

where

$$\theta(P) \quad 0.99531 P^{1/3}(1 \quad P/P_{cr})^{4/3},$$

P_{cr} – the water critical pressure, and λ [kJkg^{-1}] – the water latent heat.

The water exit sub-cooling (ΔT_{sub}) is given by

$$\Delta T_{sub} \quad T_{sat} \quad T_{f1} \quad \frac{20 l_h w_h q_c}{\rho C_p t_w W_A V_{ch}} \tag{22}$$

The burnout, or critical heat flux, is determined by making iterations on the value of q_c in eqs. (21) and (22), starting from the maximum heat flux in the hot channel. When the difference between two subsequent iterations is zero, the value of q_c is equal to the critical heat flux. This value should be greater than the maximum heat flux in the hot channel and the ratio between them represents the margin against DNB, denoted by DNBR. DNB is a critical phenomenon and must be prevented to preserve the fuel element.

Mirshak and Labuntsov correlations must be used at positive sub-cooling. When the exit sub-cooling becomes negative, the burnout heat fluxes can be reasonably estimated by using these two correlations extrapolated with ΔT_{sub} equal to zero. The lower limit for this extrapolation is the Rohsenow and Griffith pool-boiling critical heat flux correlation given by [11]

$$q_c \quad 1.21 \;\; 10^{\;3} \rho_v \lambda \quad \frac{\rho_l \quad \rho_v}{\rho_v}^{0.6} \tag{23}$$

where ρ_l and ρ_v [kgm^{-3}], are the liquid and steam densities, respectively.

Onset of flow instability

Flow instabilities are undesirable in heated channels because flow oscillations affect the local heat transfer characteristics and may induce premature burnout [11]. The burnout heat flux occurring under unstable flow conditions was well below the burnout heat flux for the same channel under stable flow conditions. Consequently, the critical heat flux leading to the onset of flow instability may be more limiting than that of stable burnout.

The most common flow instabilities encountered in heated channels under nearly atmospheric condition with forced convection are flow excursions of the Ledinegg-type. Experiments by Whittle and Forgan for sub-cooled water flowing (upward and downward) in narrow heated channels lead to their estimate of flow instability occurring at an average heat flux of [11]

$$\bar{q}_{FI} \quad R \rho C_p \frac{w}{w_h} \frac{t_{ch}}{l_h} V_{ch}(T_{sat} \quad T_{f1}) \tag{24}$$

where

$$R \quad \frac{1}{1 \quad \eta \dfrac{D_{he}}{l_h}},$$

t_{ch} – the channel thickness, D_{he} – the heated equivalent diameter of the channel given by

$$D_{he} \quad 4\frac{\textit{channel}\text{ flow area}}{\textit{channel}\text{ heated perimeter}} \quad \frac{2t_{ch}w}{t_{ch} \quad w_h},$$

T_{sat} – the water saturation temperature, and η – (experimental fit parameter) equal to 25.

Another correlation derived by Winkler for the average heat flux at onset of flow instability [11]

$$\bar{q}_{FI} \quad 29.35 \quad (128.15 \quad 1.104 T_{f1})V_{ch}^{0.8} \tag{25}$$

The peak heat flux q_{FI} is obtained by multiplying $\bar{q}_{FI}$ by the axial power factor (F_A).

Flow instability is a critical phenomenon and must be prevented by adequate TH design. The ratio between q_{FI} and the maximum heat flux in hot channel q_c represents the margin against FI. This ratio must be greater than unity.

Critical velocity

For a given plate assembly there is a critical flow velocity at which the plates become unstable and large deflections of the plates can occur. A critical velocity formula derived by Miller [11] is

$$V_{crit} = \left[\frac{2}{3}\frac{15 \cdot 10^5 E(t_p^3 - t_m^3)t_{ch}}{\rho w^4 (1 - \nu^2)}\right]^{0.5} \qquad (26)$$

where E [bar] is the Young's modulus of elasticity, t_p – fuel plate thickness, and ν – Poisson's ratio.

For design purposes, it is recommended that the coolant velocity be limited to 2/3 of the critical velocity.

PROGRAM STRUCTURE

The THDSN is a Fortran program built to operate on PCs. The program simulates the reactor core as two parallel channels, an average and a hot channel. The hot channel represents a single channel and the average one represents all other core channels. The THDSN consists of a main part and a number of subroutines, each one calculating a certain parameter, as illustrated in the program structure shown in fig. 1. As shown, there are subroutines to calculate the temperature distribution in the coolant, clad, and fuel meat. Other subroutines for calculating the heat transfer coefficient, core pressure drop and power and heat flux distribution are included. There are also subroutines for calculating the ONB, FI, DNB, and critical velocity. All the data required for these calculations are put into an input unit and the results are saved in an output unit. The program is simple and easily applicable for calculating TH parameters and safety limits and margins during normal reactor operation.

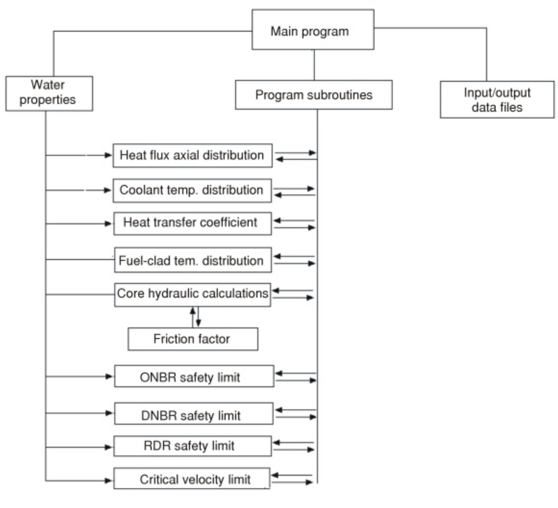

Figure 1. THDSN program structure

PROGRAM VERIFICATION

2 MW and 10 MW MTR IAEA reactors

The 2 MW and 10 MW MTR reactors described in the IAEA document TECDOC-233, Appendix-A [11] are typical plate-type RR present all over the world. All the data regarding these two types of reactors, such as core dimensions, neutron information, operating conditions, safety limits, margins and so on, that may be required for the application and verification of new programs are available and documented in that reference. Consequently, these two reactors are used for our own program verification. Table 1 shows the IAEA reactors' data required for this verification [11]. Water properties such as polynomial correlations in temperatures required for the calculations are included in the THDSN program. These correlations are derived from a curve fitted to tabulated water properties at an atmospheric pressure present in ref. [13].

Safety parameters

The THDSN program results regarding safety parameters of benchmark reactors in comparison with those mentioned in TECDOC-233 are shown in tabs. 2 and 3. In our calculations, the channel flow is considered to be downward and its exit pressure equal to the core exit pressure given in tab. 1. Table 2 shows that, except for the average heat flux and the flux at the ONB, there is a good agreement between the other calculated and referenced safety parameters. The deviation in the average heat flux returns to the heat transfer area used in the calculation. In the present study, contrary to ref. [11] that used the plate surface area, the meat surface area is the one used. This result can be checked by hand calculation of the average heat flux. With respect to ONB, there is no clear reason for the over-estimation appearing in the present calculated value. But it is important to mention that in the present study, the water properties and the heat transfer coefficient are recalculated at the new, updated temperature during the iterations on the average heat flux using eq. (22). In ref. [11], the water properties and, consequently, the heat transfer coefficient, were considered constant.

Safety margins

The safety margin is equal to the value of the safety parameter at the critical phenomenon divided by the corresponding nominal value. Therefore, tab. 3

Table 1. Input values used in the TH calculations for 2 MW and 10 MW MTR IAEA reactors [11]

Property	2 MW	10 MW	Property	2 MW	10 MW
No. of standard fuel elements (SFE)	19	23	No. of plates in SFE	19	23
No. of control fuel elements (CFE)	4	5	No. of plates in CFE	15	17
Channel flow width [cm]	6.64	6.64	Plate total length [cm]	62.6	62.5
Channel heated width [cm]	6.30	6.3	Channel thickness [cm]	0.2916	0.219
Fuel plate heated length [cm]	60.0	60.0	Clad thickness [cm]	0.0381	0.038
Core inlet temperature [°C]	38.0	38.0	Meat thickness [cm]	0.051	0.051
Core exit pressure (bar absolute)	1.961	1.566	Radial power factor	2.0	1.78
Clad thermal conductivity [W/m°C]	180.0	180.0	Axial power factor	1.58	1.4
Meat thermal conductivity [W/m°C]	53.6	53.6	Total core flow [m^3/h]	300	1000
Fuel element nozzle diameter [cm]	6.0	6.0	Channel velocity [m/s]	0.94	2.97
Fuel element nozzle length [cm]	18.0	18.0			

Table 2. Best estimate TH safety parameters

Reactor power [MW]	Pressude drop across channel [bar]		Average heat flux [W/cm^2]		Average heat flux at ONB [W/cm^2]		Burnout heat flux [W/cm^2]				Heat flux at OFI [W/cm^2]	
							Labuntsov		Mirshak			
	Present study	Ref. [5]	Present study	Ref. [5]	Present study	Ref. [5]	Present study	Ref. [5]	Present study	Ref. [5]	Present study	Ref. [5]
2	0.019	0.0186	6.28	5.8	13.58	11.4	231.4	231	230	231	102.7	102.2
10	0.199	0.193	21.54	20.54	41.97	35.9	353.2	353	265.6	266	208.0	208.8

Table 3. Best estimate TH safety margins

Reactor power [MW]	Margin to ONB eq. (23)		Margin to DNB, eq. (26)				Margin to OFI eq. (30)		Critical velocity [m/s]	
			Labuntsov		Mirshak					
	Present study	Ref. [5]	Present study	Ref. [5]	Present study	Ref. [5]	Present study	Ref. [5]	Present study	Ref. [5]
2	2.18	1.94	11.75	12.6	11.71	12.6	5.216	5.58	12.64	13
10	1.95	1.75	6.58	6.9	4.95	5.2	3.88	4.08	10.94	11

shows that there is a small deviation between the calculated and referenced safety margins. Except for the ONB margin, the slight difference between the calculated and referenced safety margins returns to the deviation in the average heat flux demonstrated in the above paragraph. The ONB margin of the present study is nearly 12% higher than the reference value. This disagreement also returns to the deviation in the ONB heat flux, as demonstrated previously. The approximate sign that appears beside the referenced values of critical velocity in tab. 3 means that values are determined by hand from figures in ref. [11].

Temperature distribution

TECDOC-233 doesn't contain the axial distribution of clad/coolant temperature in the hot or average channel for the 2 MW or 10 MW RR. To verify the THDSN program results of temperature distribution, the RELAP5 model used in ref. [10] for 10 MW benchmark reactor analysis is modified to simulate the reactor core as average and hot channels. The RELAP5 steady-state results for the clad and coolant axial temperature distribution in comparison with those of the THDSN program are shown in figs. 2 and 3, respectively. In this comparison, the THDSN program uses the Dittus Boelter correlation, eq. (14), for calculating the heat transfer coefficient. The figures show that the THDSN temperature results are in good agreement with the RELAP5 results.

Comparison with the PARET code

During the conversion of the Pakistan Research Reactor-1 (PARR-1) from HEU to LEU with an upgrade from 5 MW to 10 MW, all the thermal-hydraulic and safety parameters were calculated at the elevated power of 10 MW and by using the PARET code. For purposes of verification of the THMSN program, its results regarding the safety limits and margins of PARR-1 were compared with PARET results given in refs. [16, 17]. The PARR-1 data required for the pres-

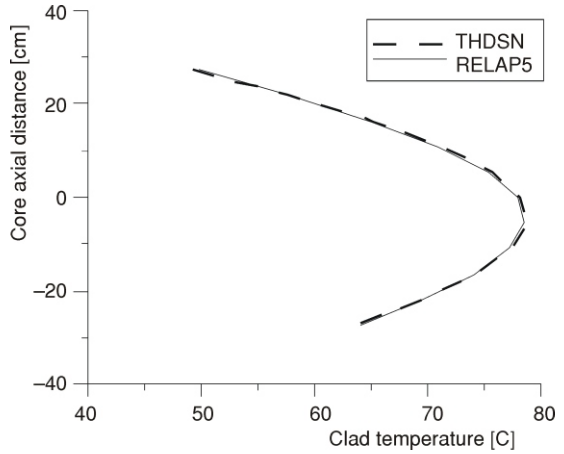

Figure 2. Clad temperature in the hot channel

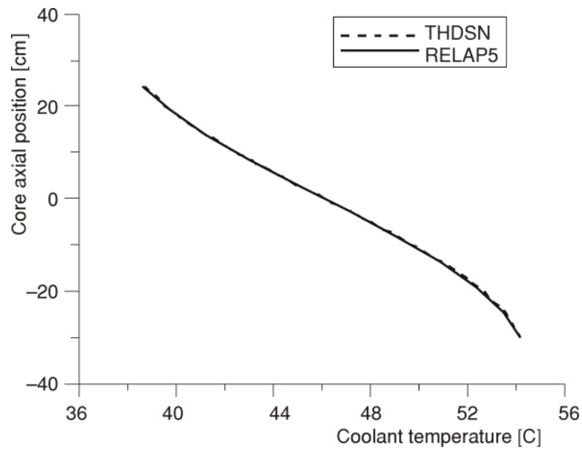

Figure 3. Coolant temperature in the hot channel

ent calculations are quoted from those references and tabulated in tab. 4.

In order to take into consideration the engineering factor shown in tab. 4, in our calculations, the radial power factor used in THDSN is taken to be equal to F_r F_{eng}. The core flow is considered to be downward. The Dittus-Boelter correlation (14) is chosen to calculate the heat transfer coefficient. Therefore, the results of THDSN compared to those of PARET are shown in tab. 5. THDSN results, with and without the engineering factor (Feng), are tabulated in tab. 5. It appears that, except for the steady-state peak clad and centerline temperatures and the ONB heat flux, there is a good agreement between the THDSN and PARET results. The agreement between the peak heat flux and the coolant temperature rise in the hot channel means that the treatment of the engineering factor in the two programs is similar. The THDSN prediction of peak clad and centerline temperatures are, respectively, 6.5% and 7.8% higher than the PARET ones. The ONB heat flux predicted by THDSN is more conservative, being 12.7% less than that of PARET, because it is calculated at worst core conditions. On the other hand, when the comparison between THDSN, RELAP5 and TECDOC-233 is considered, it can be concluded that PARET has overestimated the heat transfer coefficient and that adversely affects the peak temperatures and partially overestimates the ONB heat flux.

Comparison with the TERMIC program

The MTR reactor is an open-pool, plate-type fuel element research reactor. The core is cooled and moderated by light water and reflected by beryllium and light water. Typical MPR 22 MW research reactor data are tabulated in tab. 6 [18]. Steady-state TH calculations of the safety limits and margins regarding this MTR reactor were performed by using the TERMIC program of the MTR_PC package, as tabulated in that reference. For the purpose of verification, steady-state TH calculations for that reactor are performed with THDSN and TERMIC programs using the data in tab. 6 and the results tabulated in tab. 7.

Table 4. Pakistan research reactor PARR-1 data [11, 12]

Property	Value	Propery	Value
Core power [MW]	10	Core inlet temperature [°C]	38.0
No. of standard fuel elements (SFE)	29	Pressure at core end (bar absolute)	1.61
No. of control fuel elements (CFE)	5	Core pressure drop (bar)	0.2
No. of plates in SFE	23	Clad thermal conductivity [W/m°C]	180.0
No. of plates in CFE	13	Meat thermal conductivity [W/m°C]	53.6
Channel flow width [cm]	6.692	Radial power factor (F_r)	2.228
Channel heated width [cm]*	6.30	Axial power factor (F_a)	1.303
Fuel plate heated length [cm]	60.0	Engineering factor (F_{eng})	1.584
Fuel plate total length [cm]	62.6	Total flow rate [m^3/h]	950
Channel thickness [cm]*	0.21	Channel coolant velocity [m/s]	2.46
Clad thickness [cm]*	0.0381	Fuel element nozzle diameter [cm]	6.0
Meat thickness [cm]*	0.051	Fuel element nozzle length [cm]	18.0

* These date not present in those references and quoted from ref. [5] for 10 MW benchmark reactor

Table 5. Comparison between PARET and THDSN

Parameter	PARET, [15] $F_{eng} = 1.584$	THDSN	
		$F_{eng} = 1.584$	$F_{eng} = 1.0$
Steady-state parameters			
Average heat flux (AHF) [W/cm^2]	18.1	18.07	18.07
Peak heat flux [W/cm^2] = AHF F_a F_r F_{eng}	83.4	83.094	52.458
Coolant temperature rise [°C]			
– Average channel	9.4	9.54	9.54
– Hot channel	33.6	33.58	21.23
Peak clad surface temperature (hot channel)	102.47	109.167	84.644
Peak centerline temperature (hot channel)	104.64	112.796	86.935
Onset of nucleate boiling (ONB)			
Average heat flux [W/cm^2]			
– Bergles and Rohsenow	25.52	20.548	35.28
Onset of flow instability (OFI)			
Peak heat flux [W/cm^2]			
– Forgan (= 48)	138.0	138.297	138.193
Departure from nucleate boiling (DNB)			
Peak heat flux [W/cm^2]			
– Labunstov	326	325.469	325.469
– Mirshak	257	255.305	255.305
Critical velocity [m/s]	10.5	10.59	10.56
Safety margins			
Margin to ONB	1.4	1.137	1.953
Margin to OFI (Forgan correlation)	1.6	1.664	2.915
Margin to DNB			
– Labuntsov	3.9	3.917	6.204
– Mirshak	3.1	3.072	4.867

Table 6. MTR research reactor data [13]

Property	Value	Propery	Value
Core power [MW]	22	Core inlet temperature [°C]	40.0
No. of fuel elements (FE)	29	Core exit pressure (bar absolute)	2.09
No. of plates in FE	19	Clad thermal conductivity [W/m°C]	180.0
Channel flow width [cm]	7.0	Meat thermal conductivity [W/m°C]	13.6
Channel heated width [cm]	6.40	Radial power factor	2.22
Fuel plate heated length [cm]	80.0	Axial power factor	1.35
Fuel plate total length [cm]	84.0	Total flow rate [m^3/h]	1950
Channel thickness [cm]	0.27	Channel coolant velocity [m/s]	4.7
Clad thickness [cm]	0.04	Fuel element nozzle diameter [cm]	6.0
Meat thickness [cm]	0.07	Fuel element nozzle length [cm]	18.0

The comparison in tab. 7 shows that the THDSN results are more conservative than the TERMIC ones. The THDSN results of clad temperature are 6.44% higher, while the ONB margin is 31.48% lower than corresponding TERMIC values. Also, THDSN Mirshak calculated DNB is 21.67% under the TERMIC value. On the other hand, TERMIC overestimates the ONB margin because it uses a less conservative correlation, Foster and Greif, and overestimates the heat transfer coefficient because it calculates the coolant properties in the Dittus-Boelter correlation at film temperature and not at bulk temperature [19]. Also, TERMIC considers the exit sub-cooling (ΔT_{sub}) in terms of the Mirshak correlation, eq. (20), independent variable and that greatly overestimates the DNB margin.

CONCLUSIONS

A steady-state, best estimate FORTRAN program for TH calculations of plate-type fuel RR called THDSN has been built. The program simulates the reactor core as two channels, an average and a hot channel. The program calculates the axial distribution of coolant, clad and fuel meat temperatures, heat flux and core pressure drop. Also, it uses well-known correlations for calculating safety parameters and margins against the critical phenomena such as the onset of nucleate boiling, the departure of nucleate boiling and flow instability. The program gives the user possibility to choose one of six correlations for the heat transfer coefficient. All water properties are simulated in the program by temperature and/or pressure dependant

Table 7. Comparison between TERMIC and THDSN

Parameter	THERIC	THDSN
Steady-state parameters		
Coolant temperature rise [°C]		
– Average channel	10.0	10.85
– Hot channel	22.78	24.11
Peak clad surface temperature (hot channel)	94.2	100.27
Peak centerline temperature (hot channel)		117.79
Average heat flux (AHF) [W/cm^2]	39.0	39.0
Peak heat flux [W/cm^2] = AHF F_a F_r*	117.0	117.0
OFI peak heat flux [W/cm^2]		
Forgan	321.7	328.35
DNB critical heat flux [W/cm^2]		
– Labunstov		468.047
– Mirshak	445.3	348.819
– Bernarth	424.9	
Safety margins		
Margin to ONB		
– Bergles and Rohsenow		1.48
– Foster and Greif	2.16	
Margin to OFI		
– Forgan	2.75	2.807
Margin to DNB		
– Labuntsov		4.001
– Mirshak	4.235	2.982
– Bernarth	4.229	

polynomials produced from curve fitting to the published data. The program is constructed from a number of subroutines, each one calculating a certain distribution or safety limit, all of them called from the program main.

The THDSN program is verified by comparing its results against commercial and customized programs. Also, the program is verified by comparing its results for 2 MW and 10 MW MTR IAEA reactors with those published in IAEA TECDOC-233, Appendix-A. A good agreement has been found with IAEA TECDOC, with the exception of the ONB limit which was overestimated by a narrow margin. Also, good agreement with PARET, except for the clad temperature and the ONB limit, was established. The THDSN clad temperature turned out to be 6.5% over and the ONB 12.7% under that calculated by PARET. This underestimation in the ONB may be returned to the THDSN if the estimated value is at the channel worst condition. Another explanation may lie in the fact that the PARET overestimates the heat transfer coefficient which affects the clad temperature and the ONB limit. The comparison to TERMIC also shows that THDSN uses more conservative correlations than those used by TERMIC. Also, TERMIC overestimates ONB and DNB values.

NOMENCLATURE

A_{ch} – channel cross-sectional area, [cm^2]
C_p – coolant specific heat, [kJkg^{-1} °C^{-1}]
D_{he} – heated equivalent diameter of the channel, [cm]
D_y – channel hydraulic diameter, [cm]
F_A – axial power factor, [–]
F_D – factor less than unity, [–]
F_{NUC} – nuclear power factor (=F_1 F_2), [–]
F_R – radial power factor, [–]
FPTN – fuel plate total number, [–]
f – friction factor [–]
G_{ch} – channel mass flux, [kgm^{-2}s^{-1}]
h – heat transfer coefficient, [Wm^{-2}°C^{-1}]
k_c – clad thermal conductivity, [Wm^{-1}°C^{-1}]
k_m – meat thermal conductivity, [Wm^{-1}°C^{-1}]
l_e – extrapolated length, [cm]
l_h – fuel heated length, [cm]
L_{noz} – nozzle length, [cm]
L_p – total fuel plate length, [cm]
m_{ch} – channel mass flow rate, [kgs^{-1}]
Nu – Nusselt number, [–]
P – absolute channel pressure, [bar]
P_{cr} – coolant critical pressure, [bar]
ΔP_{ch} – channel pressure losses [bar]
Pr – Prandtl number, [–]
q – heat flux, [Wcm^{-2}]
q_C – critical heat flux at DNB, [Wcm^{-2}]
q_{FI} – peak heat flux at flow instability, [Wcm^{-2}]
q' – local power density, [Wcm^{-3}]
q_c – maximum power density, [Wcm^{-3}]
Re – Reynolds number, [–]
ρ – water density, [kgm^{-3}]
SVR – fuel meat surface to volume ratio, [cm^{-1}]
T_c – clad temperature, [°C]
T_f – coolant temperature, [°C]
T_m – meat centerline temperature, [°C]
T_{sat} – coolant saturation temperature, [°C]
t_c – clad thickness, [cm]
t_{ch} – channel thicknes, [cm]
t_m – fuel meat thickness, [cm]
t_p – fuel plate thickness, [cm]
t_w – water channel thickness, [cm]
V_{ch} – channel coolant velocity, [ms^{-1}]
V_{noz} – coolant velocity in the fuel element nozzle, [ms^{-1}]
V_o – water velocity, [ms^{-1}]
W_A – active core flow rate (=F_D W_T), [m^{3}h^{-1}]
W_T – total core flow rate, [m^{3}h^{-1}]
w – total channel width, [cm]
w_h – channel heated width, [cm]
w_m – meat width, [cm]
z – axial location, [cm]

ABBREVIATION

DNB – departure of nucleate boiling
DNBR – departure of nucleate boiling ratio
FE – fuel element
FI – flow instability
HEU – highly enriched uranium
LEU – low enriched uranium
MTR – material test reactor
OFI – onset of flow instability
ONB – onset of nucleate boiling
ONBR – onset of nucleate boiling ratio
RR – research reactor
TH – thermal hydraulic

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Ахмед КЕДР

ФОРТРАНСКИ ПРОГРАМ ЗА ТЕРМОХИДРАУЛИЧКЕ ПРОРАЧУНЕ СТАБИЛНОГ СТАЊА ИСТРАЖИВАЧКИХ РЕАКТОРА СА ПЛОЧАСТИМ ГОРИВОМ

Сигурносна процена истраживачких и енергетских реактора обавља се непрекидно током њихове употребе и захтева проверене и ваљане програме. У читавом свету програми за енергетске реакторе добро су утемељени и утврђени мерењима и подацима са одговарајућих експерименталних постројења. Ови кодови обично су софистицирани, а од корисника захтевају посебне вештине и троше много рачунарског времена. С друге стране, већини програма намењених истраживачким реакторима још увек недостају подаци ради проверавања и потврђивања. Отуда је од користи за свако регулаторно тело да развије своје сопствене кодове за преглед и оцену истраживачких реактора.

У овом раду представљен је THDSN – једноставан једнодимензионални фортрански програм за термохидрауличке прорачуне стабилног стања истраживачких реактора са горивом плочастог типа. Поред прорачуна расподела температура и градијената притиска горива и хладиоца у средњем и врућем каналу, програмом се прорачунавају сигурносне границе и маргине на основу појава критичности које се сусрећу код истраживачких реактора, као што су: феномен кључања, критични топлотни флукс и нестабилан ток. Коришћење су познате термохидрауличне корелације за прорачун сигурносних параметара и више формула за коефицијенте преноса топлоте. Програм THDSN потврђен је поређењем његових резултата за бенчмарк реакторе од 2 MW и 10 MW, са резултатима објављеним у МААЕ издањима, а са којима је нађена добра сагласност. Такође, упоређени су резултати овог програма са објављеним резултатима других програма, као што су PARET и TERMIC.

Кључне речи: истраживачки реактор, термохидраулички прорачун, стабилно стање, сигурносни параметри, Фортран програм