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Steady State Thermal Hydraulic Model for PWR Light Water Reactors

El-sheikh, B.M

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E.mail:badawymel@yahoo.com

ABSTRACT

Thermal-hydraulic analysis of nuclear reactors is very important to predict the temperature distribution in the reactor core elements, such as fuel cladding and fuel rods. Power reactors cores are composed of cylindrical fuel rods that contain fuel pellets, gap and cladding. Our goal will be to calculate the fuel and coolant temperature distribution and pressure gradient in an average and hot channel and others safety limits.

The present paper introduces a simple one dimensional Fortran program called PWRTH for steady state Thermal Hydraulic (TH) calculations and fuel elements heat conduction for Pressurized Water Reactors. A parametric analysis for obtaining the maximum possible reactor power respecting safety limits is performed.

KEYWORDS

Power Reactors;
Thermal hydraulic;
Safety limits.

1. Egyptian Nuclear and Radiological Regulatory Authority, Cairo, Egypt.

INTRODUCTION

Pressurized water reactors (PWRs) constitute the large majority of the world's nuclear power plants and are one of three types of light water reactor (LWR). Two things are characteristic for the pressurized water reactor (PWR) when compared with other reactor types: coolant loop separation from the steam system and pressure inside the primary coolant loop. The pressure in the primary coolant loop is typically 15–16 mega pascals (150–160 bar), which is notably higher than in other nuclear reactors. Light water is used as the primary coolant in a PWR. The pressurized water reactor has three new Generation III reactor evolutionary designs: the AP-1000, VVER-1200, ACPR1000+.

The modular modeling of the plant components is used to develop a system model and perform the simulation. Theoretical models of the components are obtained by formulating the describing differential or algebraic equations, usually derived from conservation of mass, energy, and momentum relationships. Accurate design and measurement data are used to determine the model parameters and inputs.

This thermal feedback requires the solution of the non-linear heat transfer equation in the fuel rods, to retrieve the effective temperature of heavy nuclei observed by the coming neutrons **(Rowlands, 1962; de Kruijf, 1994 & Goltsev *et al*., 2003)**. In the heat transfer equation, non-linearity arises with the dependence of the physical properties on the unknown temperature, and with possible radiative sources. Different forms of heat transfer needs as well specific physical properties, being thermal conductivities, contact resistances, specific heat coefficients, mass densities, radiative emissivities and others **(Carslaw and Jaeger,J. 1986 &Tomatis, D., 2013).**

PWRTH is a program that perform steady state thermal hydraulic PWR core calculations and heat conduction with the capability of calculating the required safety parameters like the Departure from Nucleate Boiling DNB, and the Fuel limit FL.

THERMAL HYDRAULIC AND HEAT TRANSFER MODELING

Discretization of the fuel rod

The fuel rod will be divided into nodes. Nodalization of fuel rod is radially and axially. In axial direction, the fuel is divided into ten slices. The radial discretisation the rod covers four nodes in the fuel, one node in gap, and two nodes in the cladding as shown in Fig. 1.

The physical properties, such as thermal conductivity and specific heat, are all temperature dependent.

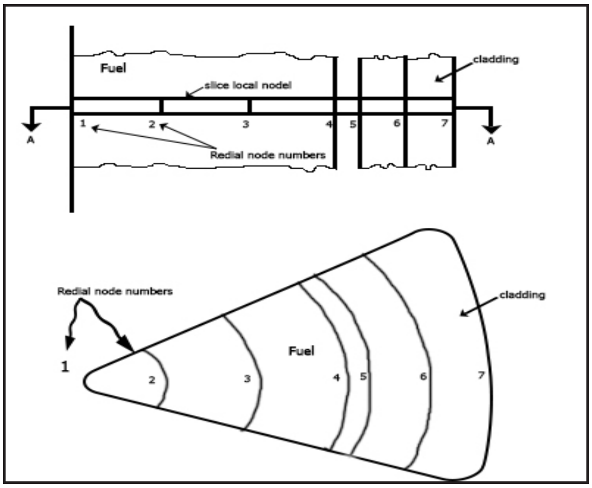

Fig. (1): Radial fuel rod discretisation.

The heat flux and Temperature distribution in the axial direction

The amount of heat generated along the fuel channel follows the neutron flux and fission reaction distribution along the fuel channel. The natural axial variation of the neutron flux is given by the following relation **(Nikola K. and Popov , 2015):**

$$\varphi(z) = \varphi_{\max} \cos\left(\frac{\pi z}{L_e}\right), \quad (1)$$

Where Le is the effective fuel length and $\varphi_{\max}$ is the maximum neutron flux in the middle core.

In coordinate system, the power distribution can be expressed by the following equation:

$$q^{\text{"}}(z) = (q^{\text{"}})_{max}\, sin\left(\frac{\pi z}{L_e}\right) \qquad (2)$$

Where $(q^{\text{"}})_{max}$ is the surface heat flux at the middle of the fuel rod in the axial direction.

In a steady-state situation, the sensible heat gain by the passing coolant (assuming no phase change) is equal to the heat generated in the differential fuel rod (Cp is the specific heat capacity at a given pressure [kJ/kg oC]):

$$wc_p \int_{T_{f1}}^{T_f} dT_c = (q^{\text{"}})_{max} \int_{z=0}^{Z=L} sin\left[\frac{\pi z}{L_e}\right] dz, \qquad (3)$$

The axial variation of the fluid temperature Tc can be obtained by substituting Eq. (2) into Eq. (3) and integrating **(El-Wakil ,M.M.,1978)**:

$$T_c = T_{c,in} + \frac{((q^{\text{"}})_{max})\, L_e}{\pi w c_p} \left[1 - \cos\left[\frac{\pi z}{L_e}\right]\right] \qquad (4)$$

The subscript "in" designates the entry point in the fuel meat. In this equation , the equivalent core length is approximated by the actual core length, i.e L=L_e

The coolant temperature is measured in the middle of the fuel rod for z= L/2 and has the following form:

$$T_{c,mid} = T_{c,in} + \frac{(q^{\text{""}})_{max}\, A_f\, L}{\pi w c_p}, \qquad (5)$$

Heat transfer coefficient correlation

For the heat transfer coefficient , the Dittus-Boelter correlation were implemented (Dittus, F. W, and Boelter ,L.M. K**, 1930,1985),(Winterton , R. H. S.,1998**):

$$h = 0.0023 \left[\frac{\dot{m}.D_h}{\mu.A}\right]^{0.8} \left[\frac{\mu.c_{pa}}{k_a}\right]^{0.4} \frac{k_a}{D_h} \qquad (6)$$

Where h is the heat transfer coefficient , the $\dot{m}\dot{m}$ and μ are the mass flow and viscosity of the coolant in kg/s, respectively , D_h *is the hydraulic diameter in m,* C_{pa} is the specific heat at constant pressure in $J/kg.oC$ *and* k_a *is the* thermal conductivity of water in $W/m.oC$.

Radial heat conduction in the fuel meat

The (time independent) conduction equation for an infinite cylindrical fuel pin **(Anglart , H. , 2010):**

$$-\frac{1}{r}\frac{d}{dr}\left[\lambda r \frac{dT}{dr}\right] = \dddot{q}, \qquad (7)$$

After integration of Eq. (7) and applying the boundary conditions, the solution in each region (centerline, fuel gap, gap-clad, clad-fluid boundary) is found.

Pressure distribution in channels with single phase flow

The total pressure drop over the coolant circulation loop has to be known in order to determine the needed pumping power.

The pressure gradient along an arbitrary channel with constant cross-section A(z) and with mass flux G can be represented by the following equation **(Anglart , H., 2005)** :

$$-\frac{dp}{dz} = \frac{1}{A}\frac{d}{dz}\left(\frac{G^2}{\rho}A\right) + \frac{p}{A}\frac{dA}{dz} + \frac{P_w \tau_w}{A} + \rho g sin\varphi \qquad (8)$$

Where τ_w *is wall shear stress* , P_w*watted channel perimter*, ρ *is the coolant density .*

For a vertical channel (φ= 90[0]) with constant cross-section (assumption valid for most of LWR fuel assemblies, Eq. (8) becomes,

$$-\frac{dp}{dz} = \frac{P_w \tau_w}{A} + \rho g \qquad (9)$$

For vertical channel with length L, the last equation can be integrated from z=0 to z=L to obtain the total pressure drop between the inlet and the outlet of fuel assembly:

$$-\Delta p_{tot} = -\int_0^l \frac{dp}{dz} dz = -[p(L) - p(0)]$$

$$-[p(L) - p(0)] = \frac{P_w \tau_w}{A} L + \rho g L \qquad (10)$$

As can be seen, the total pressure drop consists of two terms: the friction loss term and gravity term.

Where: $-\left[\frac{dp}{dz}\right]_{fric} = C_f \frac{P_w}{A} . \frac{G^2}{2\rho} \qquad (11)$

Where the friction coefficient for laminar flow can be written in a general form as,

$$C_f = a.Re^{-b} \tag{12}$$

Where a and b are constant.

Departure from Nucleate Boiling (DNB)

The heat flux to cause DNB depends on T_b , G,and P $\rightarrow \bar{\bar{q}}\,(T_b\,, G, P)$

$$DNBR(z) = \frac{\bar{\bar{q}}_{DNB}}{\bar{\bar{q}}} \tag{13}$$

Where T_b is the bulk temperature.

MDNBR≡ min. DNBR > 1.3.

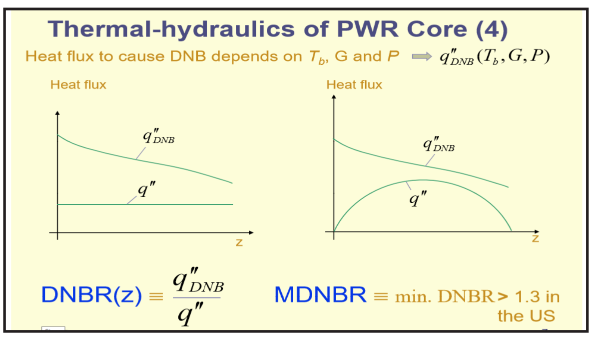

Fig. (2): Typical relation shapes of heat flux and $\bar{\bar{q}}_{DNB}$ distribution along the fuel axis.

The calculation of $\bar{\bar{q}}_{DNB}$ *in program was* according to the Tong 68 correlation [Buongiorno, J.,2010]

$$\bar{\bar{q}}_{DNB} = K_{Tong} \frac{G^{.4}\,\mu_f^{.6}\,h_{fg}}{D_e^{.6}} \tag{14}$$

Where :

$$K_{Tong} = [1.76 - 7.433x_e + 12.222x_e^2] \tag{15}$$

$$x_e = \frac{c_{P,l}(T_{sat} - T_b)}{h_{fg}} < 0 \text{ in PWR} \tag{16}$$

RESULTS

For typical PWR data tabulated in Table .1 , axial power density, fuel rod central temperature , cladding temperatures ,radial temperature profile in fuel rod and the temperature distribution throughout the coolant when the simulation has reached a steady state in Figures : 3, 4, 5, 6, 7, respectively. And to test the safety parameter we drow the effects of changing reactor power to upper and lower than the typical reactor power (1940 MWT) on fuel temperature and DNBR as in Figures 8 to 11.

It is found that if we respect the maximum fuel temperature (1500 0C) , the maximum allowed reactor power will be ~2390MW. For DNBR two limits are presented in literature: 1.3 and 1.17. If we consider 1.3 the maximum power is 1940 MW , if we consider 1.17 the maximum power is ~2250 MW.

Table (1) : *Screening of fungi for tri-calcium phosphate solubilization [Positive means halo zone formation around the colony].*

Reactor Power (P)	**1940 MWt**
Core height [m]	**3.658**
Fuel rod radius [m]	**4.75E-03**
Clad thik [m]	**0.57E-03**
Gap thik. [m]	**7.87E-05**
No. fuel elements	**145**
No. of rod in fuel elements	**264**
Inlet core temperature [k]	**552.5**
Outlet core temperature [k]	**588.6**
P1	**2300 MWt**
P2	**2800 MWt**
P3	**1500 MWt**
P4	**1100 MWt**

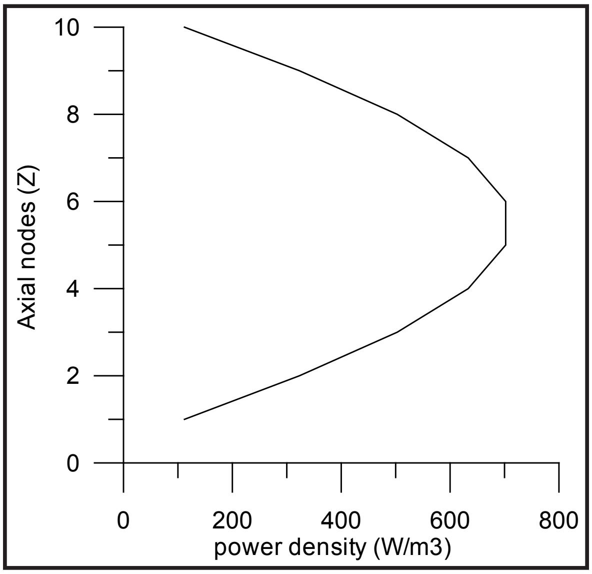

Fig. (3): Axial power density distribution.

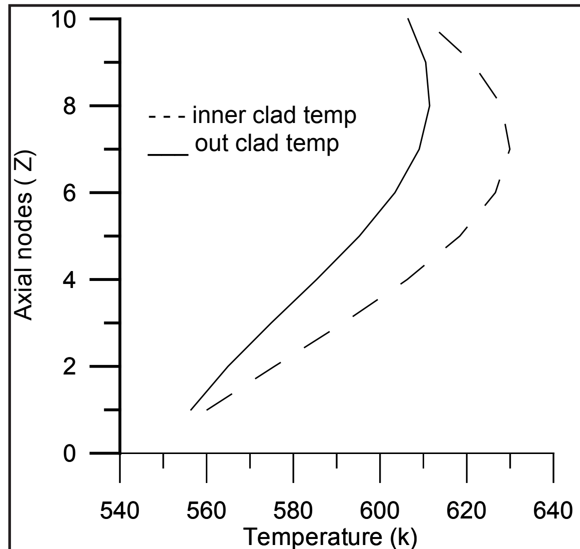

Fig. 5 Internal and external clad temperature (k)

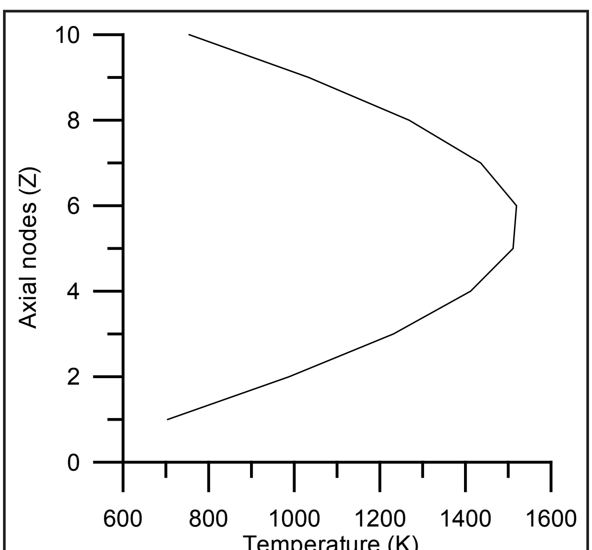

Fig. 4 Fuel rod centeral temperature (k)

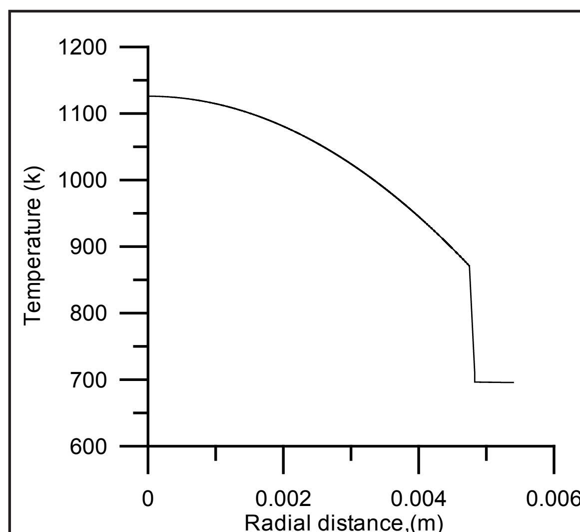

Fig. 6 Radial temperature distribution in a fuel rod

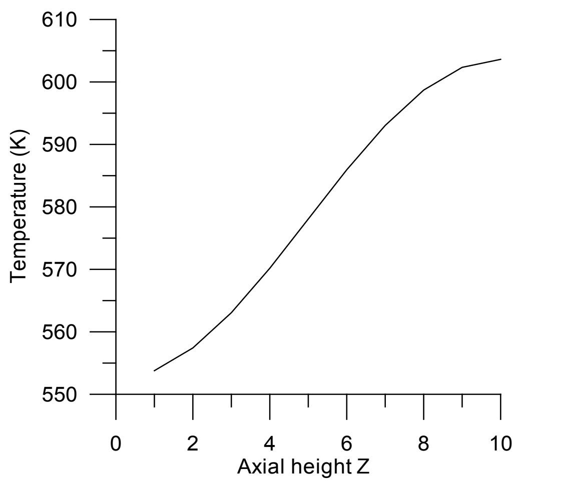

Fig. 7 : Temperature distribution along the coolant channel

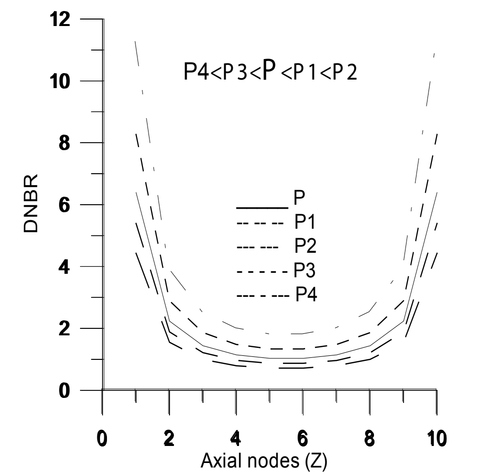

Fig. (10): Change of DNBR according to changing of reactor power

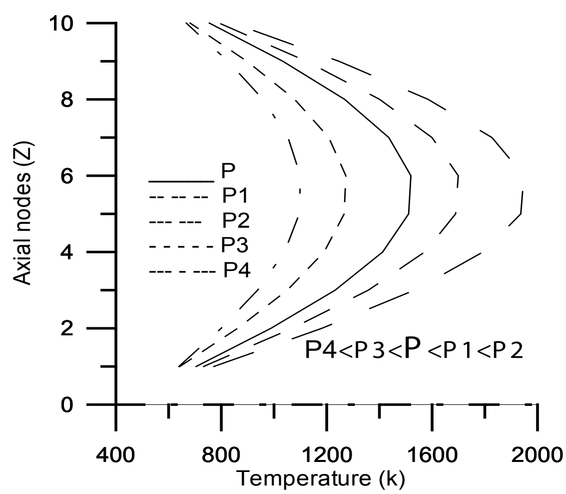

Fig. (8): Change of fuel central temperature according to changing Reactor power

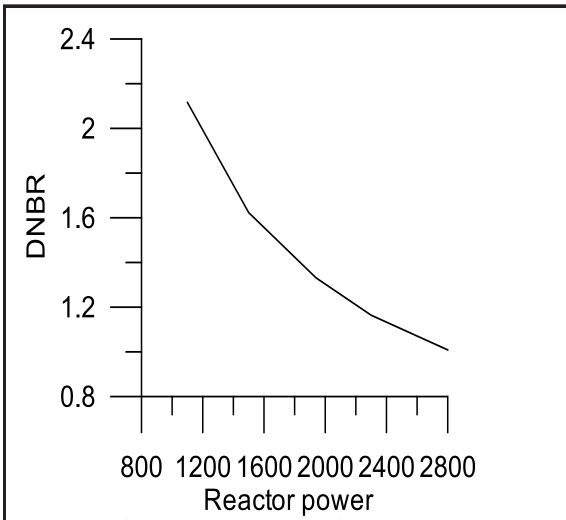

Fig. (11): DNBR versus different reactor power

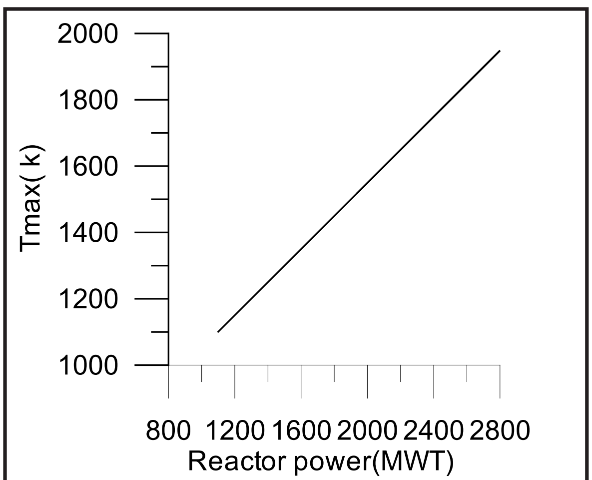

Fig. (9): Fuel rod central temperature versus different reactor power

CONCLUSIONS

PWRTH a Fortran program has been designed for PWR thermal hydraulic. Heat transfer radially and vertically in average and hot channel are presented. Some thermal hydraulic limits are presented. The results has shown that the program to be already an effective tool for PWR thermal hydraulic analysis. It's found that the max. allowable power for fuel temperature is 2390MW and the max. allowable for two DNBR limits 1.3, 1.17 are 1940 MW and 2250 MW respectively. Hence it is shown that the DNBR is the governing factor in determining the maximum operating power.

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مجلة
التقنيات النووية
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نمذجة السلوك الاستاتيكي الهيدرلوكي الحراري لمفاعلات الماء المضغوط

بدوي محمود الشيخ

إن التحليل الهيدروليكي الحراري للمفاعلات النووية له أهمية عظمى، نظرا للحاجة إليه للتنبؤ بتوزيع درجة الحرارة في العناصر الأساسية للمفاعل ، مثل قضبان الوقود وأغلفتها، حيث يتكون قلب مفاعلات الطاقة من عناصر وقود اسطوانية تحتوي على كريات الوقود وأغلفة الوقود والفرجة التي بينهما. ويهدف البحث إلى حساب توزيع درجة حرارة الوقود وسائل التبريد وتدرج الضغط في القناة الحارة والمتوسطة وحدود أمان الأخرى. تقدم هذه الورقة البحثية برنامج حسابي بسيط أحادي الأبعاد بلغة لفورتران يسمى PWRTH لحساب الحالة الحرارية الهيدروليكية الثابتة (TH) والتوصيل الحراري لعناصر الوقود لمفاعلات الماء المضغوط. كما تم اجراء تحليل حدودي للحصول على الحد الأقصى من قدرة المفاعل المحتملة التي تتوافق مع حدود السلامة.

1. هيئة الرقابة النووية والإشعاعية.