2017 International Nuclear Atlantic Conference - INAC 2017
Belo Horizonte, MG, Brazil, October 22-27, 2017
ASSOCIAÇÃO BRASILEIRA DE ENERGIA NUCLEAR– ABEN

Agência Brasileira do ISBN
ISBN 978-85-99141-07-6

A THERMAL HYDRAULIC ANALISYS IN PWR REACTORS WITH UO_2 OR (U-Th)O_2 FUEL RODS EMPLOYING A SIMPLIFIED CODE

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ABSTRACT

In order to project a nuclear reactor, the neutronic calculus must be validated, so that its thermal limits and safety parameters are respected. Considering this issue, this research aims to evaluate the APTh-100 reactor thermal limits. This PWR is a project developed in Universidade Federal do ABC (UFABC) using fuel composed of Uranium and Thorium oxide mixed (U,Th)O2. For this purpose, a simplified, although conservative, code was developed in a MATLAB environment named STC-MOX-Th" "Simplified Thermal-hydraulics Code-Mixed Oxide Thorium". This code provides axial and radial temperature distribution, as well as DNBR distribution over the hottest channel of the reactor core. Moreover, it brings other hydraulic quantities, such as pressure drop over the fuel rod, considering any fuel proportion of (U,Th)O2.The software uses basic laws of conservation of mass, momentum and energy, it also calculates the thermal conduction equation, considering the thermal conductive coefficient as a temperature function. In order to solve this equation, the finite elements method was used. Furthermore, the proportion of 36% of UO2 was used to evaluate the temperature over the fuel rod and DNBR minimum in three burn conditions: beginning, middle and ending. The program has proven to be efficient in every condition and the results evidenced that the APTh-1000 reactor, in an initial analysis, has its thermal limits within the recommended security parameters.

Keywords: nucleate boilling, mixed oxides, heat transfer, nuclear fuel.

1. INTRODUCTION

According to [1], in the design of a nuclear reactor, the neutron calculations must be validated to ensure that the thermal limits - the melting temperature of the fuel and the coating of the rod, as well as the DNB rate - are not violated. There are, in the academy, consolidated thermal hydraulics codes like COBRA [2], and with the advance in the computers and numerical methods with CFD software's such as ANSYS-CFX, or Multiphysics codes, which couples neutronics with thermal hydraulics calculations.

In the UFABC, a project of a nuclear reactor whose fuel is composed of mixed oxides of Uranium and Thorium (U-Th)O_2 based on the reactor AP-1000, titled APTh-1000 [3] was developed. For this project there was a need to ensure that the thermal limits and safety parameters of the reactor in question would be duly respected. However, most of the conventional codes did not use (U-Th)O_2 fuels, besides being too expensive.

Thus, a simplified thermal hydraulic code was developed. The code is conservative, using the basic laws of conservation of mass, momentum and energy. Its name is STC-MOX-Th: "Simplified Thermal-hydraulics Code-Mixed Oxide Thorium" [4] and was developed in MATLAB environment, providing axial and radial temperature distributions as well as the distribution of DNBR and other hydraulic calculations such as pressure drop, considering only the hottest channel in the reactor core. For these calculations, adequate correlations were used, considering the type of fuel used.

This work describes the development of the code, as well as the results obtained for the APTh-1000 reactor, in order to show that the design developed in [3] is within the parameters of safety in the hydraulic term of view.

2. CODE DESCRIPTION

2.1 Thermal-hydraulic Model

The thermal hydraulics analysis is made in a single channel of a typical PWR, as illustrated in Figure 1, in which the basic conservation of mass, momentum and energy were solved.

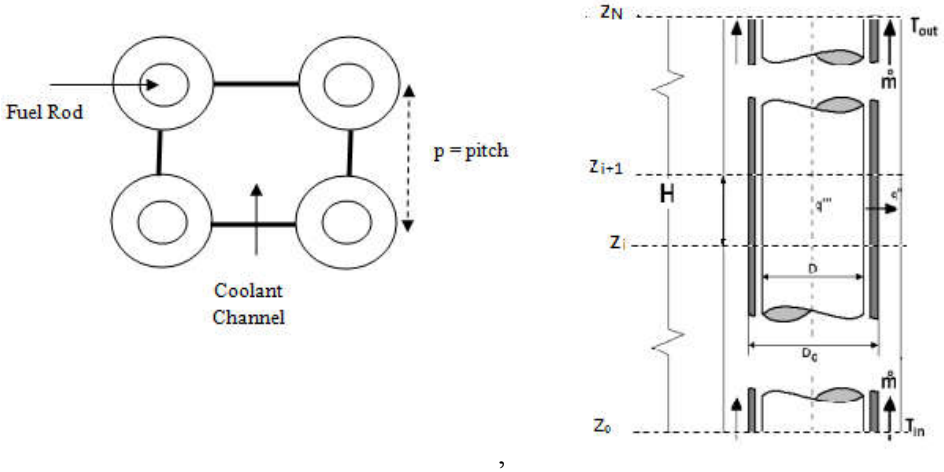

Figure 1: Simplified diagram of a single PWR channel with axial nodes.

The code subdivide the channel in axial nodes (i), i=0, 1,..., n, where the conservation laws are applied. The mass and energy conservation neglecting cross flow between channels yields:

$$\dot{m}_c dh_z = q'(z)dz = q''(z)p_\alpha dz = q'''(z)\pi(D_H/2)^2 dz\,, \tag{1-a}$$

or, by discretize the energy balance equation for the axial control volume, z_i:

$$h_{i+1} = h_i + \int_{z_i}^{z_{i+1}} q'(z)dz \cong h_i + \frac{\Delta z_i}{2\dot{m}_c}\left(\frac{q_i + q_{i+1}}{2}\right), \tag{1-b}$$

where h is the coolant enthalpy , m_C is the mass flow, here calculated conservatively by dividing the coolant mass flow by the number of channels in the core, taking in account that 2-3% flows at the edges of the core [5]. The q, q', q" and q''' are power, linear power, heat flux and power density axial distribution respectively, and coming from the neutronics, and here conservatively assumed the maximum generated in the core (hot spot) and constant in the radial direction, p_a is the wet perimeter, $D_H = 4p_a/A$; $A= p_a^2-\pi(d_F/2)^2$, the hydraulic diameter. The power distributions varies with fuel burn up, but at Beginning of Cycle (BOC), could be given by a cosine distribution [6]:

$$q''' = q'''_{\max}\cos\left(\frac{\pi z^*}{\tilde{H}}\right), \tag{2}$$

where $\tilde{H}$ is the extrapolated length, ($\tilde{H} = H + d$), H the physical length, d the extrapolated distance, and z^*=z-H/2.

The momentum equation allows calculating the pressure drop, along the axial direction, which are due to friction, gravitational and acceleration or form factor, in the space grids of the fuel assembly, given by:

$$P_{j+1} = P_j + f_j\frac{\rho_j (u_j)^2}{2D_H}\Delta z_j + K_j\frac{\rho_j u_j^2}{2}\Delta z_j + \rho_{nj} g\Delta z_j\,, \tag{3}$$

where, $\Delta P_n = P_{n+1} - P_n$ is the pressure drop, f is the friction fator, D_H is the hydraulic diameter, $\bar{u}$ is the coolant speed, ρ is the coolant specific mass, and g is the gravity acceleration. The friction factor [7] and the grid form factor [8], are given in function of the Reynolds number (Re), by:

$$f_i = 0{,}341\,\mathrm{Re}_i^{-0{,}25}, \tag{4}$$

and

$$K_i = 11{,}63\,\mathrm{Re}_i^{-0{,}25}\,. \tag{5}$$

Starting from the botton of fuel pin(z_0), where the input pressure, and enthalpy of the coolant are know, we may go to thermodynamic tables to find the values of all thermodynamics quantities, and so use equations (1-b), and (3) to calculate the values at z_{i+1} for all these quantities, as instance the fluid temperature, T_f. Therefore, the cladding, gap and fuel temperatures distribution can be reconstructed, as given by equations (6) to (9) as shown in[1].

$$T_s = T_f + \frac{q''}{h_s} = T_f + \frac{q'}{2\pi(r_F + t_c)h_s}, \tag{6}$$

$$T_c = T_s + \frac{q'}{2\pi r_F}\frac{t_c}{k_c}, \tag{7}$$

$$T_F = T_c + \frac{q'}{2\pi r_F h_G}, \tag{8}$$

and the Fuel Center Line Temperature, given by:

$$T_{CL} = T_F + \frac{q'}{4\pi k_{MF}}, \tag{9}$$

where T_S is the outer surface cladding temperature, h_S is the convective heat coefficient, and calculated from the Dittus-Boelter correlation [1], T_C is the inner surface cladding temperature, T_F is the inner fuel rod temperature, r_f, t_c, k_c are the radius of the fuel rod, cladding thickness and thermal conductivity respectively. The thermal conductivity of fuel varies with the temperature, and its mean value k_{MF} can be calculated by [1]:

$$k_{MF,j} = \frac{1}{T_j - T_{j-1}} \int_{T_{j-1}}^{T_j} k_F(T)dT, \tag{10}$$

For the temperature dependence of k_F, two correlations have been used, one of them for UO_2 [1], and other for mixed uranium thorium oxide [9]:

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$$k_{(UO_2)} = \frac{38{,}24}{402{,}4 + T} + 6{,}1256x10^{-13}(T + 273)^3 \ , \tag{11}$$

$$k_{(ThO_2)} = \frac{1}{A + B(T + 273)} \ , \tag{12}$$

where A = 0,0213 and B = 1,59x10 $^{-4}$.

Besides the temperature distributions, the code also calculates the DNBR (Departure from Nucleate Boiling Ratio) in function of the axial direction, given by [10]:

$$DNBR = \frac{q''_{chf}}{q''_l} \tag{13}$$

where q_{chf} it's the critical heat flux and q'' it's local heat flux. For the safety conditions, nuclear reactors usually is designed for DNBR ≥ 1, 3[1, 10]. For calculation of the critical heat flux, the W-3 [11] correlation was used.

2.2 Computational Model

The computational method solves the mass, energy and momentum balance equations in a simplified manner, treating as conservative system. Some quantities are calculated with a MATLAB thermodynamic table [12]. Figure 2 shows a simplified diagram of the code.

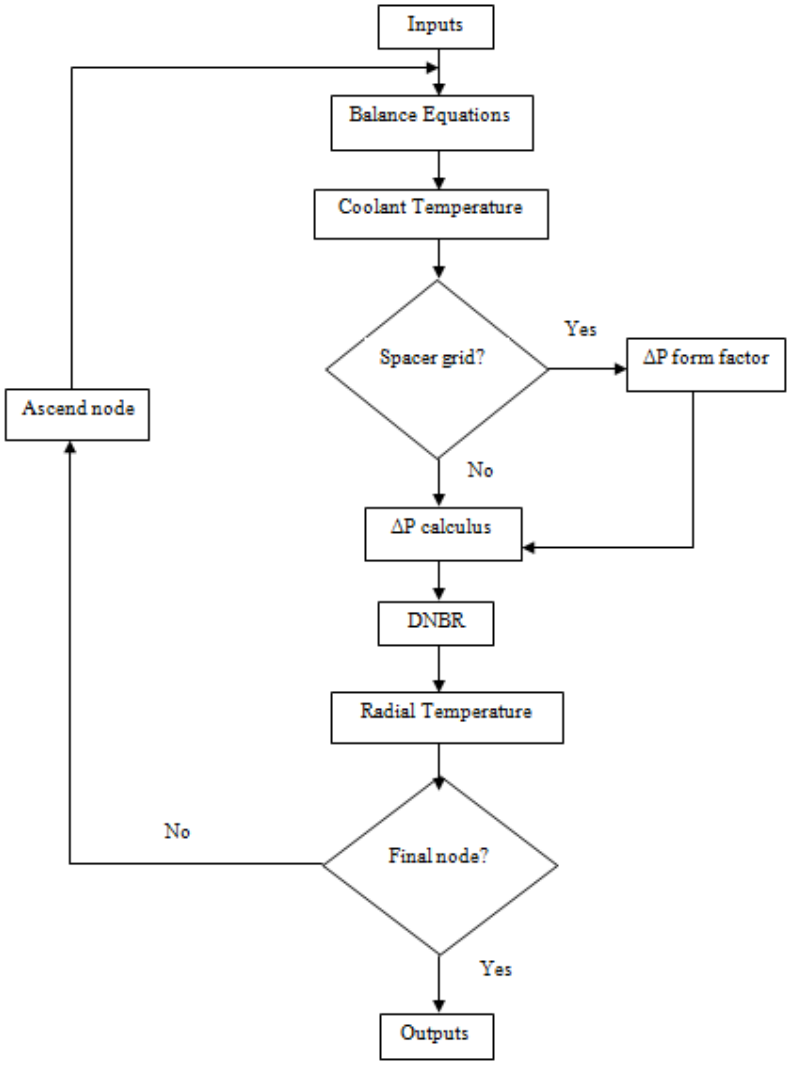

Figure 2: Simplified diagram STC-MOX-Th code.

A peculiarity of this code is to treat the thermal conductivity coefficient as a function of temperature, that is, k = k(T). This condition results in the difficulty that, the fuel along the heat conduction equation becomes non-linear equation in cylindrical coordinates [13]:

$$\frac{1}{r}\frac{d}{dr}k\frac{dT}{dr}+q'''=0\,. \tag{16}$$

To solve this problem, the finite element method [14] was used. The process of discretization of equation (16) is shown below, assuming that q "' is constant:

$$\int\left(\frac{d}{dr}Kr\frac{dT}{dr}\right)dr=-\int rq'''dr\,. \tag{17}$$

$$K\frac{dT}{dr}=-\frac{q'''r}{2}+\frac{(b-a)}{r}\,. \tag{18}$$

Where a and b are the integration constants. Integrating again into T and r, find at:

$$\int_{T_0}^{T}KdT=-\frac{q'''}{4}\left(r^2-r_o{}^2\right)+(b-a)\ln\left(\frac{r}{r_0}\right). \tag{19}$$

In order to perform a scan along the entire fuel pellet, it is assumed that the space is divided into very small intervals, which leads to the simplification of $r/r_0 \sim 1$, canceling the dependent Ln term in equation (19). Thus, we find at a temperature-dependent function, F (T), since r and r_0 are known values.

$$F(T)=\int_{T_0}^{T}KdT+\frac{q'''}{4}\left(r^2-r_o{}^2\right). \tag{20}$$

To find the solution to the problem, the F (T) zeros were determined by the secant method [15].

3. RESULTS AND DISCUSSIONS

In this work, data from the AP-1000 reactor was used, which served as the base for the APTh-1000 reactor [3]. Here, with the aid of the STC-MOX-Th code, the temperatures of the refrigerant, the internal and external parts of the cladding, and the center line temperature of the fuel were calculated axially and radially, as well the DNB ratio. Table 1 shows the input data of the STC-MOX-Th code that were used in this work.

Table 1: Inlet thermal hydraulics quantities in AP-1000.

Quantity(unit)	
Fuel diameter(mm)	9,1
Inlet temperature(°C)	292
Cladding thickness (mm)	0,7
Gap thickness (mm)	0,02
Mass flow (kg/s)	13456
Number of rods	289
Number os spacer grids	8
Rod length – active – (cm)	366
Inlet pressure (bar)	155
Pitch(cm)	1,26
Heat flux (W/cm)	539

Initially the temperature distribution along the axial axis of the rod is calculated (with the exception of the central fuel temperature).

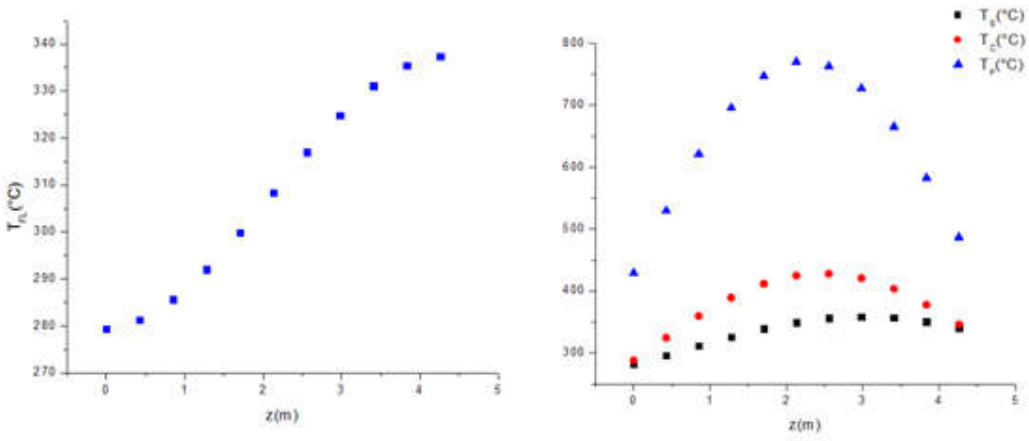

Figure 3: Axial temperature distribuction: (a) Coolant temperature; (b) Cladding outlet temperature (Ts), Cladding inlet temperature (Tc) and Fuel outlet temperature (Tf)

The coolant temperature behaved within the expected, since it is expected that it varies less than 60°C [16], and the obtained value for the maximum fluid temperature was 332°C. Likewise, all other temperatures behaved in the expected profile, mainly the calculation T_F, which in the hottest sub channel reached temperatures near 786°C [16].

In particular, fuel burn was considered and the temperature of the fuel as well as the DNB ratio was analyzed under three conditions: At the beginning, middle and at the end of the cycle (BOC, MOC and EOC respectively). The fuel used was (U-Th)O_2 with a proportion of 36% for UO_2.

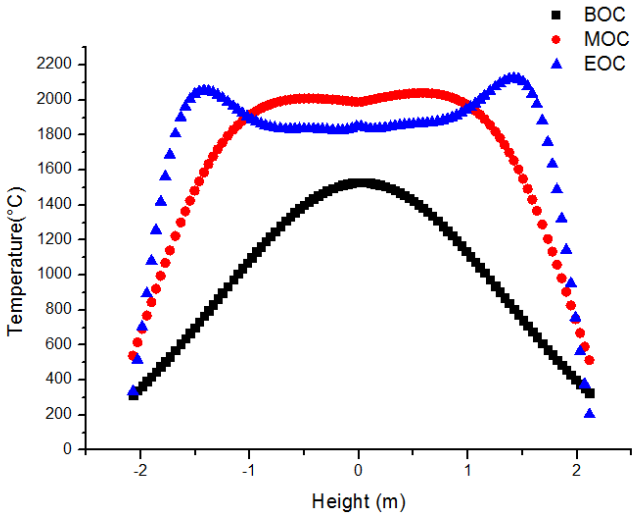

Figure 4: Center line temperature distribution for BOC, MOC and EOC

In the BOC regime, the maximum temperature reached is considerably lower than the others. (1529.1 ° C), while the maximum temperatures in MOC and EOC reached above 2000 ° C. However, such temperatures are below the melting point of the UO_2 (~ 2800 ° C)[17], which is lower than the melting point of the MOX, since the melting point of ThO_2 is greater than 3200 ° C [17]. The figure 5 shows the DNBR distribution.

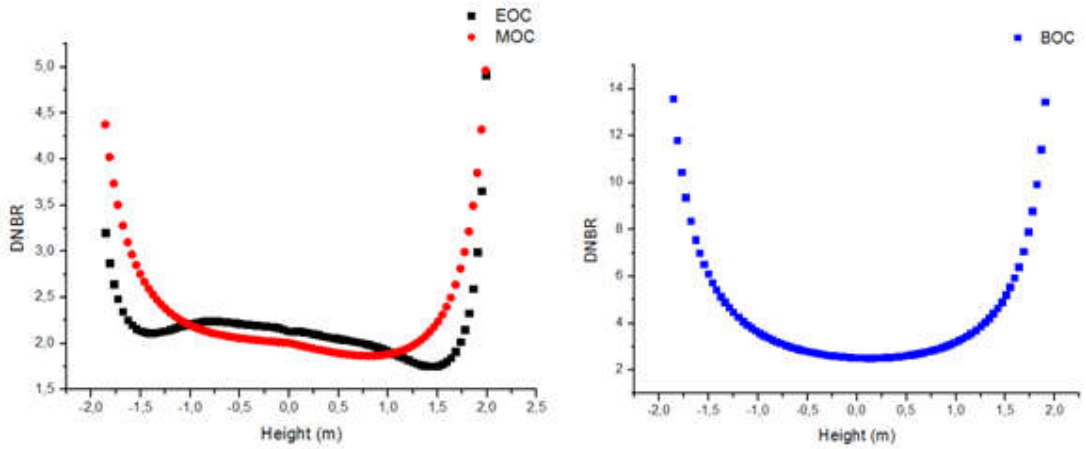

Figure 5: DNBR distribution.

Throughout the fuel cycle, the minimum value of the DNBR was always higher than 1.3. Standard value to ensure a safety margin in a nuclear reactor project [1]. The lowest value for DNBR was 1.75 in the EOC regime.

4. CONCLUSIONS

The obtained values to fuel central temperature ($T_{CL/BOC}$ = 1529.1°C; $T_{CL/MOC}$ = 2041.2°C; $T_{CL/EOC}$ = 2125.8°C) are lower than the melting point of UO_2 (2865°C) and ThO_2 (3390°C). Such a fact leads to the finding that the obtained temperatures do not reach the melting point of $(U,Th)O_2$.Moreover, the DNBR minimum values were abore 1.3, which is the minimum value considering the safety margin to the DNBR of a reactor of this size, in accordance with [1].

Such data show that the neutronic analysis of the APTh-1000 reactor project is feasible from the thermos hydraulic point of view, and it is within the recommended safety parameters.

The STC-MOX-Th code has proven to be efficient, considering what it was supposed to be calculated. The code did not cause problems, even under burning conditions analysis, that is exactly when the heat flux distribution is more irregular. The temperature distribution in MOC and EOC presented the expected standard, following the heat flux given as input parameter from neutronic calculus.

Although the program's data and performance has proven to be efficient, it still functions only as a preliminary analysis to the thermos hydraulic study of PWRs. Being a conservative program, it provides safety and trustworthiness to the obtained data. However, it still requires the account of more complex situations, such as crossflow and the insertion of more detailed correlations to $(U,Th)O_2$ thermal conductivity coefficient. It should be noted that this last situation is an area that needs more advances.

Nevertheless, STC-MOX-Th has proven to be an important tool in order to carry out an initial thermo hydraulic analysis, without the need of expensive or complex softwares, presenting the differential of using $UO_2$and mixed oxides of $(U,Th)O_2$.

ACKNOWLEDGMENTS

The authors are grateful to Capes, CNPq and Eletronuclear, as well as to their respective academic institutions for supporting the project.

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