24th ABCM International Congress of Mechanical Engineering
December 3-8, 2017, Curitiba, PR, Brazil

COBEM2017-2871
PARAMETRIC ANALYSIS OF THE PRIMARY CIRCUIT OF THE ANGRA 1 NUCLEAR POWER PLANT PWR TYPE AND ITS APPLICATIONS IN THE NAVAL NUCLEAR PROPULSION

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Abstract. *In this work we studied the primary circuit of the Angra 1 nuclear power plant, located in Rio de Janeiro, which is based on a Pressurized Water Reactor (PWR) type. A steady-state thermodynamics analysis was made to reach the operational parameters of primary circuit such as pressure, temperature, power generated among others. Previous studies available in literature of 2-loop Westinghouse Nuclear Power Plants, which is based on a PWR and similar to Angra 1, support this analysis in the sense of a correct procedure to deal with many complex processes to energy generation from a nuclear source. Temperature profiles in reactor and steam generator were studied with concepts of thermodynamics, heat transfer and fluid mechanics, showing the behavior into them. And finally, a hypothetical simulation for a nuclear submarine propulsion was made from analysis of all parameters studied from Angra 1, to fit the specifications in this kind of application based on actual designs. Using these concepts, the values of the total thermal power generated in reactor and the fuel rod average linear power presented deviations of 0.38% and 0.45%, respectively, compared with the available values of Angra 1. The maximum power and temperatures reached were lower than the safe operational limits. In the SG, the height of the tubes required to the secondary fluid reach the saturation was determined at 1.4 m for Angra 1 and 0.451 m for nuclear submarine. The results obtained in the naval propulsion simulation are similar with those found in literature.*

Keywords: *Nuclear Engineering, Energy, Thermodynamics, Reactor, Steam Generator.*

1. INTRODUCTION

In this work we carried out a steady-state analysis of the mass and energy balance of the primary circuit of a PWR type nuclear power plant, through thermodynamics, heat transfer and nuclear systems concepts. The study is based on literature involved in 2-loop Westinghouse plants and the use of data on properties such as pressure, temperature, flow, among other data available in Eletronuclear (2014) and Westinghouse (2005), aiming for reliable results to use in the primary circuit of a submarine.

Based on the thermodynamics laws, besides the use of the concepts of fluid mechanics and heat transfer, it was possible to obtain values related to the heat generation in the reactor and your transport to the secondary circuit inside the steam generator, and the profiles of temperature and heat transfer of these equipment.

An initial simulation was applied to a hypothetical nuclear submarine, based on a Scorpène-class (Poder Naval, 2008), using predefined operating parameters of Angra 1. The primary system of Angra 1 was reduced on a 1 : 3.5 scale to fit in a submarine. Thus, all thermodynamic parameters were recalculated in order to obtain the new thermal power system, in accordance with the space limitations of a submarine. Some data of INS Arihant, available in Naval-technology (2017), were used to compare with calculated parameters.

2. METHODOLOGY USED FOR THIS ANALYSIS

The simplified PWR power plant, according to Todreas and Kazimi (2011), is showing in Fig. 1.

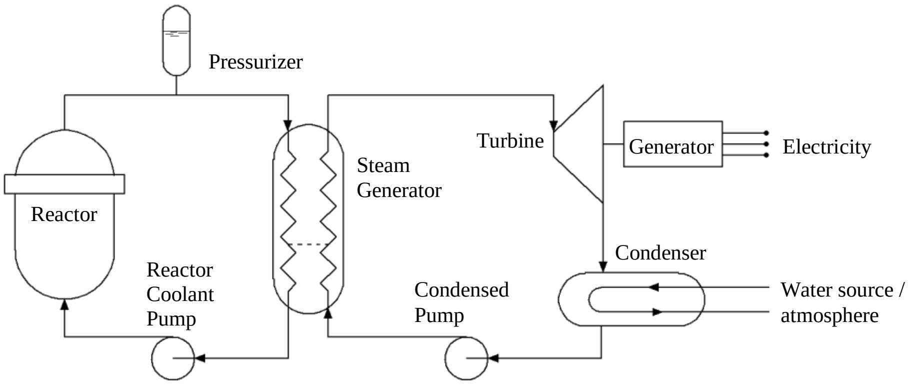

Figure 1. Simplified flowchart of typical PWR plant (Todreas and Kazimi, 2011).

To make this analysis, the primary circuit was divided in two control volumes, as showing in Fig. 2.

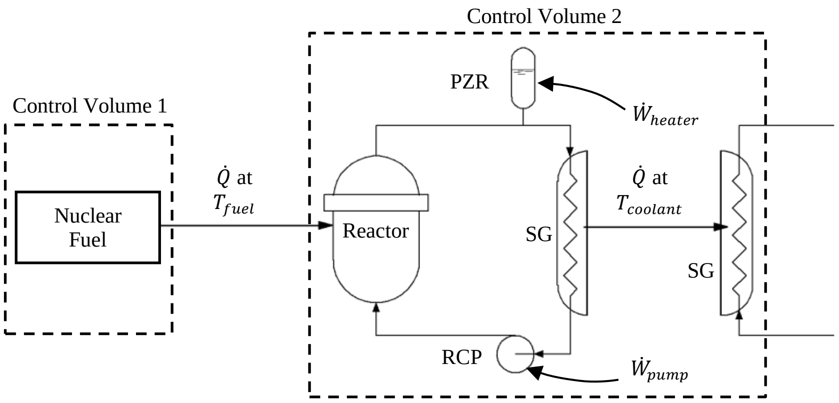

Figure 2. Simplified primary circuit with analyzed control volumes (Todreas and Kazimi, 2011).

Control Volume 1 contains the nuclear fuel. This analysis was made from Newton's and Fourier's laws (Incropera et al., 2008), with concepts of nuclear systems (Todreas and Kazimi, 2011). Nuclear fuel was analyzed in three steps, which are: fuel rods, reactor core and coolant. Volume Control 2 contains all the primary circuit equipment. In this work only the reactor and steam generator were analyzed. This analysis was made from the first law of thermodynamics (for reactor) and analysis of heat exchangers without phase change (for SG). All the following equations were developed to be possible for modeling the temperature profiles in *Microsoft Excel* software.

2.1 Analysis of Control Volume 1

2.1.1 Fuel rods analysis

The first step was determining the temperature profiles on the fuel rods, as shown in Fig. 3, using the following equations, developed according to Todreas and Kazimi (2011).

$$\Delta T_f = \frac{q'''_f}{4k_f} a^2 \tag{1}$$

$$\Delta T_g = \frac{q'''_f . (a+g)}{2h_g} \tag{2}$$

$$\Delta T_{cl} = \frac{q'''_f . a^2}{2} \frac{\ln\left(\frac{a+g+b}{a+g}\right)}{k_{cl}} \tag{3}$$

Where q'''_f is the power density of the nuclear fuel [W/m³], ΔT_f is the temperature variation in the fuel pellet [°C], ΔT_g is the temperature variation in the gap between fuel pellet and cladding [°C], ΔT_{cl} is the temperature variation in the cladding [°C], k_f is the thermal conductivity of the fuel pellet [W/m.°C], h_g is the thermal conductance of the gap with pressurized Helium [W/m².°C] and k_{cl} is the thermal conductivity of the cladding [W/m.°C].

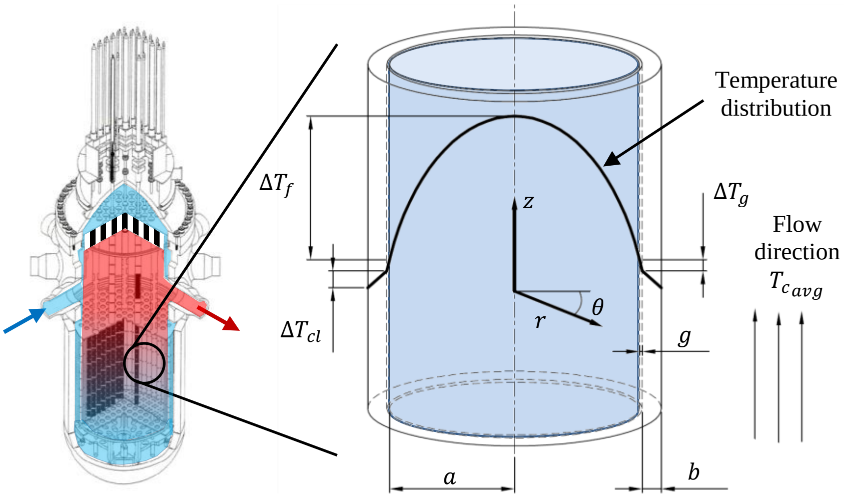

Figure 3. Temperature distribution on the fuel rod.

The maximum temperature on the fuel pellet centerline depends on the local coolant average temperature and is given by sum of the temperature variations in the fuel, gap, cladding and the coolant itself, as following:

$$T_{f\,max} = \frac{q'''_f}{2}\left[\frac{a^2}{2k_f} + \frac{(a+g)}{h_g} + \frac{a^2.\ln\left(\frac{a+g+b}{a+g}\right)}{k_{cl}} + \frac{(a+g+b)}{h_{c,cl}}\right] + T_{c_{avg}} \tag{4}$$

Where $T_{c_{avg}}$ is the coolant average temperature [°C] and $h_{c,cl}$ is the convective coefficient between coolant and cladding [W/m².°C].

2.1.2 Reactor core analysis

According to Todreas and Kazimi (2011), the power distribution in a homogeneous reactor core follows a Bessel and cosine function, as shown in Fig. 4:

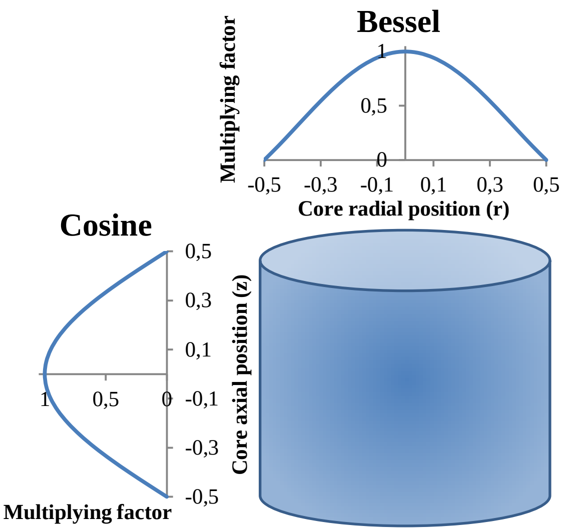

Figure 4. Power generation profiles according multiplying factors (Todreas and Kazimi, 2011).

Thus, the second step was to determine the power density profiles in the reactor core. The power density in a homogeneous cylindrical core, according to Todreas and Kazimi (2011), is given by:

$$q'''(r,z) = q'''_{max} J_0\left(\frac{2{,}405r}{R_e}\right)\cos\left(\frac{\pi z}{L_e}\right) \tag{5}$$

Where q'''_{max} is the maximum core power density [W/m³], J_0 is the zero order Bessel function, r is the core radial position, with r = 0 at center [m], $R_e$is the core extrapolated radius [m], z is the core axial position, with z = 0 at middle point [m] and $L_e$is the core extrapolated length [m]. In this work, the effects caused by neutron flux have been neglected, therefore the extrapolated dimensions were considered equal to physical dimensions.

Looking at Fig. 4 is possible to note that the maximum core power density is located in the middle of the axial and radial position. According to Westinghouse (2008), the average fuel power density (neglecting effects from neutron flux) is given by:

$$q'''_{avg} = \frac{\dot{Q}_{tot}.Fr_{ger}}{L_e.N_{fr}.\pi.a^2} \tag{6}$$

Where $\dot{Q}_{tot}$ is the total reactor thermal power [W], "a" is the radius of fuel pellet [m], N_{fr} is the number of fuel rods and Fr_{ger} is the Fraction of heat generated in fuel [%]. Note that is possible to determine the average linear power (q'_{avg}) multiplying the Equation 6 by $\pi.a^2$ and the total reactor thermal power $(\dot{Q}_{tot})$ can be obtained by thermodynamics first law.

2.1.3 Coolant analysis

The third step was determining the coolant temperature distribution along the core length, by an energy balance on coolant, using core power distribution cited above. Note that the coolant inlet temperature shall be known. This profile can be modeled by the following equation:

$$T_{c2_{avg}}(z) = T_{c1} + 0{,}61135\left(\frac{a^2 q'''_{max}.L_e}{\dot{m}_c c_p}\right)\left(sen\left(\frac{\pi z}{L_e}\right)+1\right) \tag{7}$$

Where T_{c1} is the coolant inlet temperature [°C], $T_{c2_{avg}}$ is the average coolant temperature at specified position [°C], $\dot{m}_c$ is the coolant mass flow in a single channel [kg/s] and c_p is the coolant specific heat [kJ/kg.°C].

2.2 Analysis of Control Volume 2

The total thermal power of the reactor can be obtained by thermodynamics first law, where $\dot{m}_{tot}$ is the coolant flow through reactor [kg/s] and h_{out} , h_{in} is the coolant enthalpy at reactor outlet and inlet [kJ/kg], respectively, as follows:

$$\dot{Q}_{tot} = \dot{m}_{tot}(h_{out} - h_{in})_{reactor} \tag{8}$$

Figure 5 shows the way chosen for steam generator analysis:

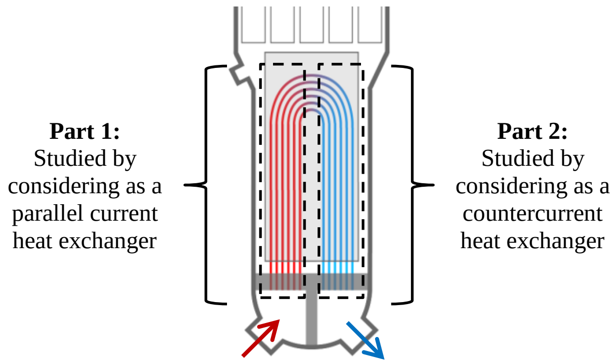

Figure 5. Selection of parts for steam generator analysis.

In order to optimize the analysis of the steam generator, the heat transfer from primary to secondary circuit was studied assuming that the inlet tubes (Part 1) was considered as a parallel current heat exchanger and the outlet tubes (Part 2) was considered as a countercurrent heat exchanger, as shown in Fig. 5.

For the Part 1, a typical profile of the parallel current heat exchanger was analyzed to obtain a general equation to model it in *Microsoft Excel* software, as following:

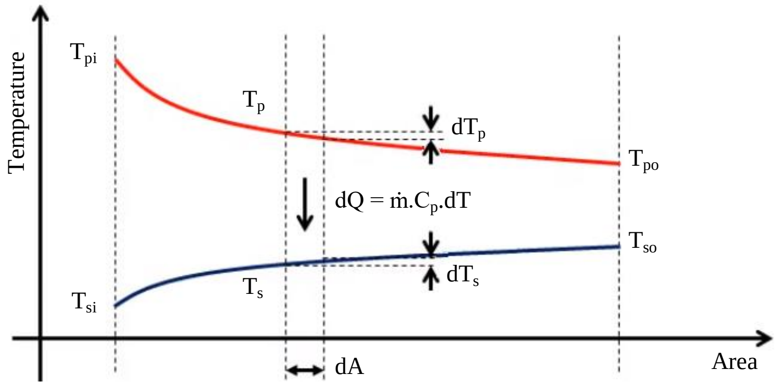

Figure 6. Analysis of a typical profile of the parallel current heat exchanger.

Through this analysis, it was possible to develop the general equation of the parallel current heat exchanger (Seleghim, 2014). The following equation can be modeled in *Microsoft Excel* software:

$$\frac{dT_p}{dA} + U.\boldsymbol{X}'.T_p = -\frac{\dot{m}_p.c_{p_p}}{\dot{m}_s.c_{p_s}}\,\frac{dT_p}{dA} + U.\boldsymbol{X}'.\left(T_{si} + \frac{\dot{m}_p.c_{p_p}}{\dot{m}_s.c_{p_s}}.(T_{pi} - T_p)\right) \tag{9}$$

With $\boldsymbol{X}' = \left(\frac{1}{\dot{m}_p.c_{p_p}} + \frac{1}{\dot{m}_s.c_{p_s}}\right)$

Where dA is the infinitesimal area of heat transfer [m²], U is the overall heat transfer coefficient [W/m².°C], subscripts $_{p}$, $_{s}$ refers to primary and secondary fluid respectively, subscripts $_{pi}$, $_{po}$ refers to primary fluid inlet and outlet respectively, and subscripts $_{si}$, $_{so}$ refers to secondary fluid inlet and outlet respectively.

For the Part 2, a typical profile of the countercurrent heat exchanger was analyzed to obtain a general equation to model it in *Microsoft Excel* software, as following:

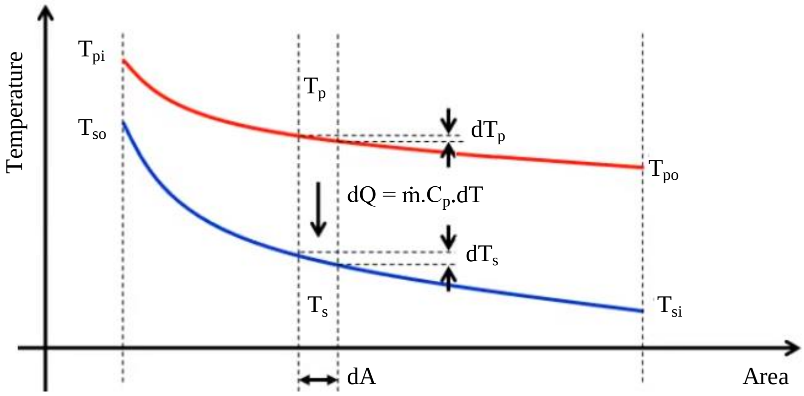

Figure 7. Analysis of a typical profile of the countercurrent heat exchanger.

Through this analysis, it was possible to develop the general equation of the countercurrent heat exchanger (Seleghim, 2014). The following equation can be modeled in *Microsoft Excel* software:

$$\frac{dT_p}{dA} + U.\boldsymbol{X}''.T_p = \frac{\dot{m}_p.c_{p_p}}{\dot{m}_s.c_{p_s}}\ \frac{dT_p}{dA} + U.\boldsymbol{X}''.\left(T_{si} + \frac{\dot{m}_p.c_{p_p}}{\dot{m}_s.c_{p_s}}.(T_p - T_{po})\right) \tag{10}$$

With $\boldsymbol{X}'' = \left(\frac{1}{\dot{m}_p.c_{p_p}} - \frac{1}{\dot{m}_s.c_{p_s}}\right)$

The overall heat transfer coefficient (U) depends on the convective coefficient of the primary circuit and the secondary circuit, which are determined by the number of Reynolds, Nusselt, Prandtl and friction factor of the flow tubes, internally and externally. In this study, the Nusselt number was obtained by Gnielinski correlation (Incropera et al., 2008) where f is the tube friction factor (obtained by Moody chart or correlations of Colebrook, Churchill, etc.):

$$Nu = \frac{\left(\frac{f}{8}\right)(Re - 1000)Pr}{1 + 12{,}7\left(\frac{f}{8}\right)^{0{,}5}\left(Pr^{\frac{2}{3}} - 1\right)} \tag{11}$$

It is important to note that the Eq. (9) and Eq. (10) must be applied only in the subcooled region of the secondary fluid. In other words, it should be used only where the secondary fluid presents single phase.

3. RESULTS AND DISCUSSION

3.1 Angra 1

Table 1 contains the main data (Eletronuclear, 2014; Todreas and Kazimi, 2011; Westinghouse, 2005; NUREG-1754, 2001) required for the analysis of the reactor to determining the temperatures profiles.

Table 1. Parameters of Angra 1 reactor core used for this analysis.

Operational Data of the Angra 1 Reactor Core	
Fraction of heat generation in fuel [%]	97.4
Nominal pressure [kgf/cm²]	158
Total number of fuel rods	28435
Number of fuel rods per assembly	235
Total coolant flow through reactor [kg/s]	8966
Coolant temperature at the reactor inlet [°C]	287.5
Coolant temperature at the reactor outlet [°C]	324.3
PWR 16 X 16 Fuel Assembly	
Pellet diameter at BOC / MOC [mm]	8.23 / 8.38
Gap at BOC / MOC [mm]	0.086 / 0.010
Cladding thickness [mm]	0.535
Convective coefficient water-cladding [W/m².K]	34000
UO_2 Thermal Conductivity [W/m.K]	2.163
Cladding thermal conductivity [W/m.K]	13.85
Thermal conductivity in gap [W/m.K]	0.4902

Using Eq. (8) and modified Eq. (6), it was possible to find 1889.11 MWth of total thermal power generated in reactor and 17.68 kW/m of fuel rod average linear power. These values represent errors of 0.38% and 0.45%, respectively, compared with data in Eletronuclear (2014). The average core power density was defined as 332.34 MW/m³ at reactor startup (BOC) and 320.55 MW/m³ during normal operation (MOC) using Eq. (6). Average core power density values are not provided by Eletronuclear (2014).

By integrating the Bessel and cosine functions, it was possible to obtain the mean values of multiplication factors, defined as 0.611351 for the Bessel function and 0.63662 for the cosine function. Therefore, the value of the maximum core power density was defined as 852,91 MW/m³ at BOC and 823.62MW/m³ at MOC.

Equations (1), (2), (3), (4) and (7) provide us the following behaviors at BOC and MOC conditions, as shown in Fig.8. Note that the thermal expansion of fuel pellet at MOC decreases the power density and the maximum temperature on fuel rod centerline due to the increase of pellet volume.

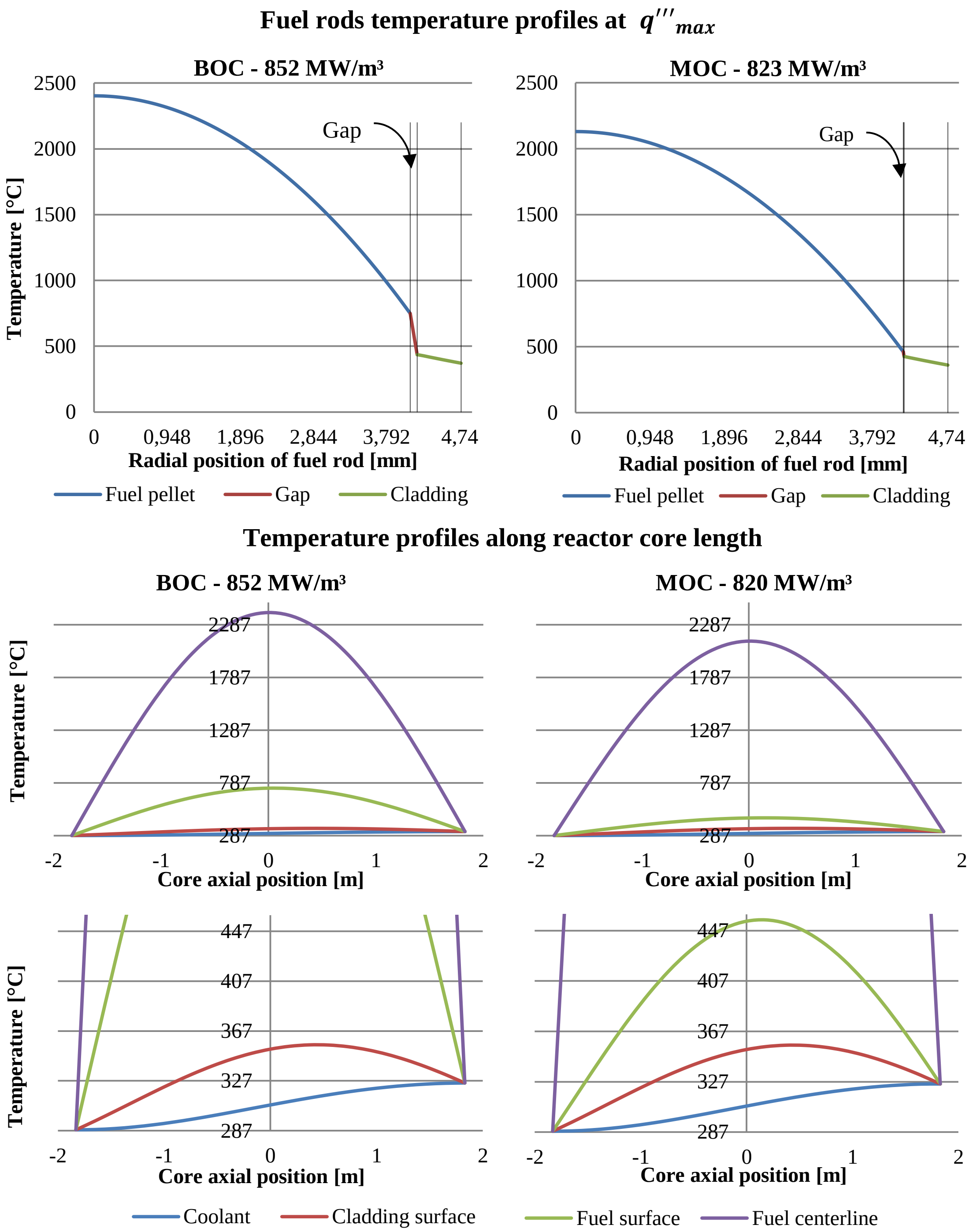

Figure 8. Temperature profiles on fuel rod and reactor core at BOC and MOC.

Table 2 contains the required main data (Eletronuclear, 2014; Westinghouse, 2005) to Steam Generator analysis to determine the temperature profiles.

Table 2. Angra 1 power plant steam generator data.

Steam Generators operational data	
Primary side pressure [kgf/cm²]	158
Secondary side pressure [kgf/cm²]	64.7
Primary side coolant flow [kg/s]	4479
Primary side inlet temperature [°C]	324.3
Primary side outlet temperature [°C]	287.5

Feedwater inlet temperature [°C]	221.1
Steam outlet temperature [°C]	279.2
Steam outlet quality	0.999
Mass flow of feedwater or steam outlet [kg/s]	515
Number of U-tubes	5428
Average U-tubes length [m]	22.1
Tubes external diameter [mm]	19.5
Tube wall thickness [mm]	1.05
Inconel tubes thermal conductivity [W/m.°C]	12.1

Using the Eq. (9), Eq. (10) and Eq. (11), it was possible to model the profiles in *Microsoft Excel* software. The temperature profiles of the tubes subcooled regions are shown in the Fig. 9. The length of the tubes required for the secondary fluid for beginning phase change was determined at 1.4 m as shown in Figure 9:

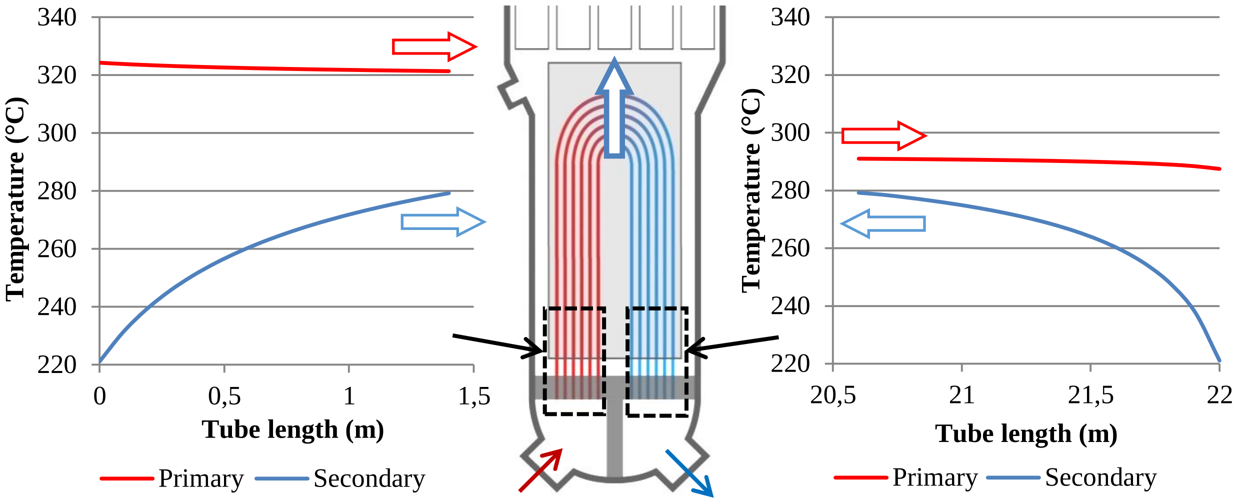

Figure 9. Temperature profiles in the subcooled region of Parts 1 and 2.

The overall temperature profile in the SG tubes is shown in Fig. 10. Note that in the phase change region of the secondary fluid, the effects caused by the boiling have been neglected. This causes the primary fluid temperature decrease following a linear behavior, while the secondary fluid maintains the temperature during the phase change.

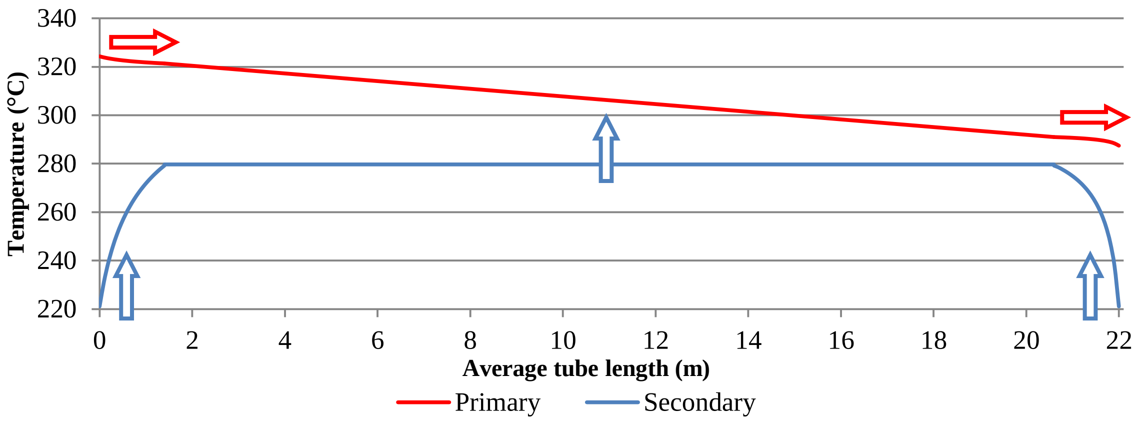

Figure 10. Overall temperature profiles in SG tubes.

3.2 Naval propulsion

To fit in a Scorpène Class submarine, with 11.7 m diameter (Deagel, 2012), the Angra 1 primary circuit was reduced on a 1 : 3.5 scale. Therefore, the geometric parameters were recalculated. Thus the core active length has changed from

3.66 m to 1.049 m, the internal diameter of primary circuit tubes has changed from 0.737 m to 0.215 m and the average SG tubes length has changed from 22.1 m to 6.3 m.

According to Eletronuclear (2014), when the coolant inside the tubes reaches a velocity around 15 m/s, the tube corrosion begins to be severe. Thus, the coolant mass flow in a single loop was determined assuming a velocity of 7.5 m/s and was defined as 203.5 kg/s. Therefore, the total flow through reactor was defined as 407 kg/s.

The reduction scale used is illustrated in Fig. 11

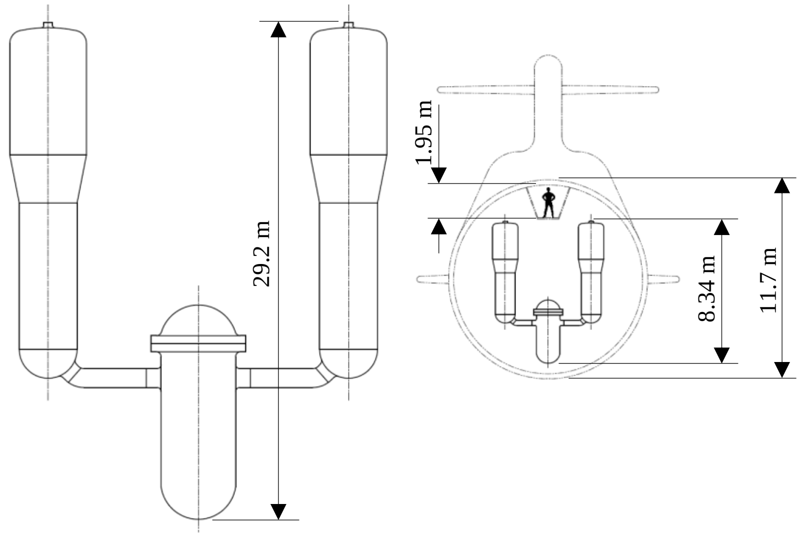

Figure 11. Illustration of the reduction scale of the Angra 1 primary circuit.

The same analysis realized to Angra 1, was made for submarine equipment. Using the thermodynamics tables with the Eq. (8), the total thermal power generated in the reactor was calculated and defined as a value of 85.7 MWth.

By the Eq. (6) adapted to use the value of fuel rods average linear power of Angra 1, it was possible to determine the quantity of fuel assemblies following the concept of a cylindrical core. The number of fuel assemblies was defined as 21. Therefore, the number of fuel rods was defined as 4935.

Through the Eq. (6), it was possible to determine the new value of the fuel rods average linear power, defined as 16.12 kW/m, and the average fuel power density, defined as 303.1 MW/m³ at BOC and 292.3 MW/m³ at MOC. The value of the maximum core power density was defined as 779 MW/m³ at BOC and 751MW/m³ at MOC.

The maximum temperature on fuel centerline for the encountered conditions was 2152.79°C.

Using the Eq. (9), Eq. (10) and Eq. (11), it was possible to model the profiles in *Microsoft Excel* software by iterations on values of the heat exchange area to determine the number of tubes that are required to reach the operational parameters of Angra 1. Thus, about 990 tubes of 19.05 mm external diameter were required to these conditions. The length of the tubes required for the secondary fluid reaching saturation was determined at 0.451 m.

With the results of the analysis, it was possible to develop the Tab. 3 with comparative data between Angra 1 (Eletronuclear, 2014; Westinghouse, 2005), the studied submarine and the INS Arihant (Naval-technology, 2017) to evaluate the encountered results:

Table 3. Angra 1 power plant steam generator data.

Reactor	Angra 1	Submarine	INS Arihant*
Power generated in the reactor [MWth]	1889.1	85.7	83
Submarine’s hull width [m]	---	11.7	11
Number of fuel elements	121	21	13
Fuel rods in each element	235	235	348
Number of fuel rods	28435	4935	4524
Piping average diameter [mm]	737	215	---
Average coolant flow [kg/s]	8966	407	---
Fuel assemblies active length [m]	3.66	1.049	---
Steam Generator	**Angra 1**	**Submarine**	**INS Arihant***
Rejected heat in each SG [MWth]	943.14	42.85	---
Secondary coolant average flow [kg/s]	515	23.41	---

Approximated number of tubes	5428	990	---
Tubes average length [m]	22.1	6.3	---
Tubes external diameter [mm]	19.05	19.05	---

* INS Arihant is India's first nuclear-powered submarine

4. CONCLUSIONS

The methodology used to calculate the thermodynamic parameters of the Angra 1 primary circuit was considered adequate due to short errors encountered. The extension of those concepts to a submarine was considered satisfactory, since the energy generated in the reactor is comparable to the reactors used for naval propulsion (Naval-technology, 2017; Ragheb, 2010) and the maximum values of power and temperature found are lower than the limits set for safe operation according to Westinghouse (2008).

The height of the SG tubes where the secondary fluid reaches saturation, in Angra 1 and submarine, represents about 15% of the total height. Only for this value evaluation, assuming a constant variation of the secondary fluid enthalpy, the saturation point is also about 15% of the total. Thus, the value found was considered satisfactory.

Studies related to technology involving nuclear energy contribute to the Brazilian economic and technological development. According to Padilha (2012), a technology transfer agreement signed between Brazil and France in 2009 provides for the construction of a Scorpène class nuclear submarine. It will be the first Brazilian nuclear submarine and its construction is an entry card for the select group of countries that dominate nuclear technology.

5. ACKNOWLEDGEMENTS

I would like to show my gratitude to my English teacher, Pauline D. Nogueira, for helping me with the grammar of this article.

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