THERMAL AND NEUTRONIC SIMULATIONS OF THE ANGRA 2 NUCLEAR POWER PLANT

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Abstract. *In this work, reference data from the Angra 2 Final Safety Analysis Report (FSAR) have been used to perform the model and simulation of the Angra 2 Nuclear Power Plant (NPP). The system is a Pressurized Water Reactor (PWR) type with four coolant loops. The moderator is H2O which contains dissolved boric acid and which also acts as coolant through the core. The fuel assemblies are mechanically identical. They have a square cross section and are arranged in the reactor core in a pattern closely matching the circular shape of the surrounding cylindrical core barrel. Angra 2 was simulated using the thermal hydraulic RELAP5 code and the neutron kinetic PARCS code. The results have been compared with the available data and demonstrate that the developed model is capable to reproducing the behavior of the reactor in steady state operation.*

Keywords*: RELAP5,PARCS, WIMS, ANGRA 2*

1. INTRODUCTION

This work presents steps of the coupling methodology application between RELAP5 and PARCS codes using as a model the Angra 2 reactor. The coupled thermal hydraulic and neutron kinetics system codes has been extensively used especially to simulate transients cases involving feedback effects into the macroscopic cross sections caused by thermal hydraulic behaviour. As first step, only the core nodalization was developed and then coupled with a preliminary neutronic simulation (Reis et al., 2015). Now, the thermal hydraulic nodalization using RELAP5 model 3.3 (US NRC, 2001) simulates the whole plant including the core, the 4 recirculation loops and the pressurizer. The reactor core neutronic nodalization using PARCS code (US NRC, 2006), the corresponding mapping that connects both codes and the cross sections generations were obtained. The radial relative power distribution to "beginning of cycle 1- full power" were utilized in RELAP5 nodalization and taken from FSAR. Similarly, geometrical and materials data such as fuels, cladding, moderator, etc. were also taken from FSAR, beginning of cycle 1 (FSAR, 2013). Results of thermal hydraulic and neutronic simulations are presented.

2. ANGRA 2 DESCRIPTION

Angra 2 is a pressurized water reactor type (PWR) with four coolant loops, of German technology Siemens / KWU (now Areva NP), resulting from nuclear agreement between Brazil and Germany signed in 1975. The core is composed of 193 fuel assemblies having a square cross-section, arranged in the reactor core in a pattern closely matching the circular shape. The moderator is H_2O acting as coolant through the core.

Each fuel assembly has 256 positions being 236 occupied by fuel rods and the remaining 20 by guide tubes. Each fuel rod is composed of uranium dioxide pellets in a zircaloy cladding tube. The main characteristics of reactor are shown in Tab. 1.

Table 1. Angra 2 basic data (FSAR, 2013).

CURRENT STATUS	IN OPERATION
GEOGRAPHIC COORDINATES	-23o 00' 30 South -44o 28' 26" West
TYPE OF REACTOR	PWR
REACTOR SYSTEM SUPPLIER	SIEMENS/KWU
OPERATOR	ELETRONUCLEAR
REACTOR POWER	3.771 MWt / 1.350 MWe
REACTOR CORE CHARACTERISTICS	
FUEL MATERIAL	ENRICHED URANIUM -UO2
Nr. OF FUEL ASSEMBLIES	193
Nr. OF FUEL RODS PER ASSEMBLY	236
LOOPS NUMBER	4
GEOMETRY OF FUEL ASSEMBLY	16 x 16 - 20
CONTROL RODS NUMBER	61

3. THERMAL HYDRAULIC MODELLING

The overall objective of thermal hydraulic design of the reactor core is to provide adequate heat transfer compatible with the heat generation distribution in the core such that on heat removal by the reactor coolant system. The RELAP5 code was used to generate the ANGRA 2 thermal hydraulic nodalization represented in a general way in Fig. 1. The radial average power distribution relative to "beginning of cycle 1- full power" was taken from FSAR (see Fig. 2) and was used to simulate the power of the heat structures in RELAP5 nodalization. The point kinetic model was used in RELAP5 simulations. The axial power distribution was calculated considering a cosine profile starting from the average relative power distribution of the Fig. 2.

The 193 assembly positions were grouped into 37 thermal hydraulic channels, as shown in the Fig. 3. This core was previously developed (Reis et al., 2015b). In the model, the flow area of coolant through the core was proporcionally divided into 37 regions, representing 37 thermal hydraulic channels (identified as pipes numbered from 600 up to 636 in the Fig. 1). Each one of these channels was associated with a correponding heat structure representing the fuel. These 37 hydrodynamic channels were divided into 34 axial volumes. The core nodalization was originarily constructed with 10 thermal hydraulic channels (Mantecón et al., 2015).

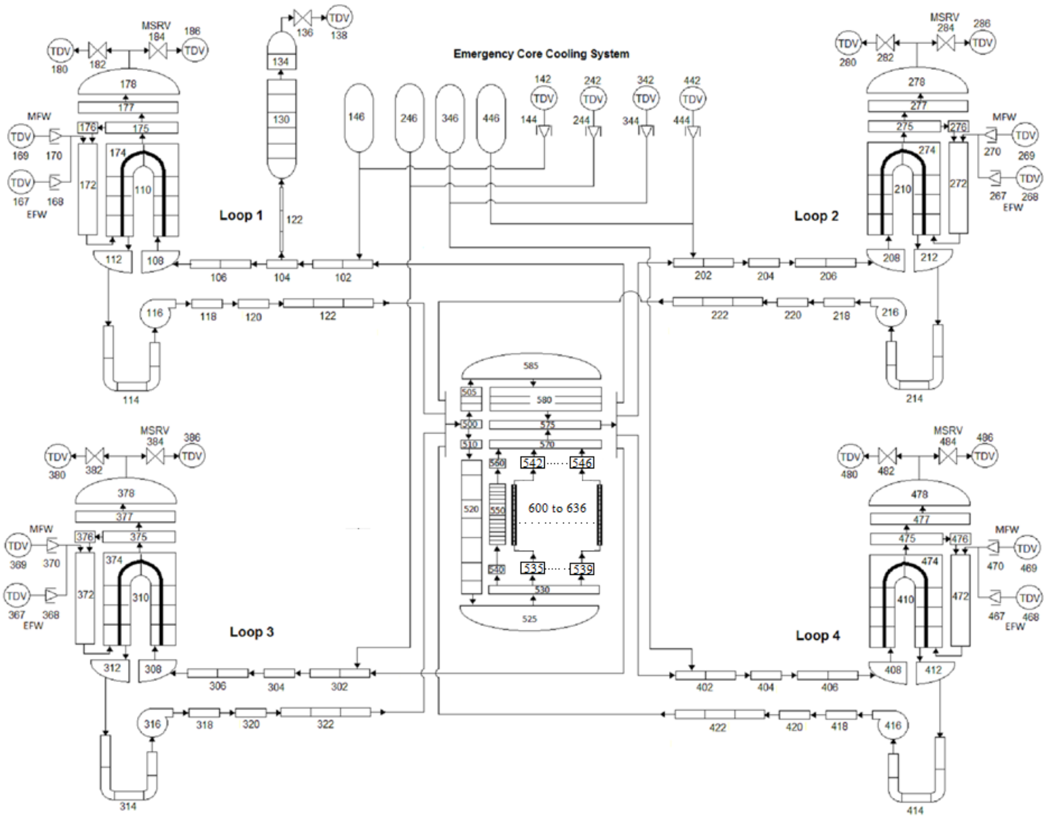

Figure 1. General RELAP5 nodalization for the Angra 2 PWR.

1.191	1.198	1.303	1.267	1.132	0.998	0.999	0.845
1.198	1.205	1.212	1.273	1.121	1.040	0.988	0.822
1.303	1.212	1.196	1.174	1.171	0.962	0.900	0.735
1.267	1.273	1.174	1.215	1.089	0.978	0.773	0.518
1.132	1.121	1.171	1.089	1.132	0.982	0.828	
1.002	1.040	0.962	0.978	0.982	1.101	0.617	
0.999	0.957	0.900	0.773	0.828	0.617		
0.845	0.840	0.743	0.518				

Figure 2. Radial average power distribution - 1/4 core symmetry (FSAR, 2013).

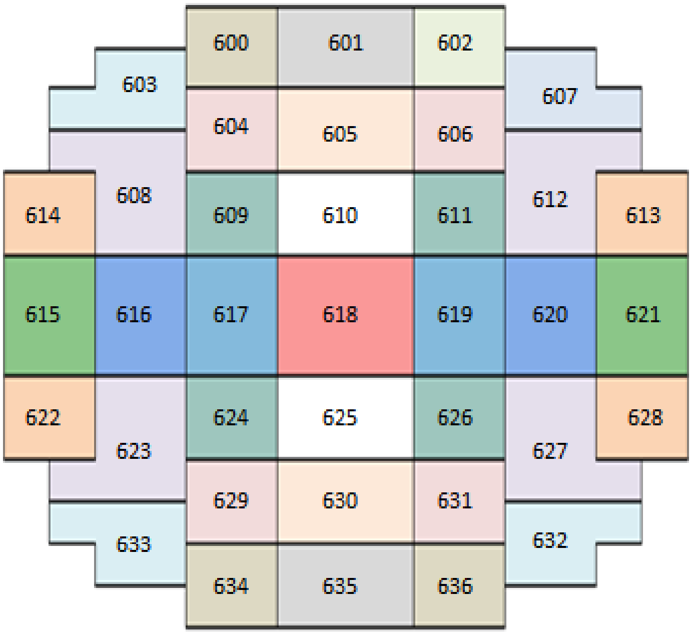

Figure 3. Representation of the 37 thermal hydraulic channels considered in the Angra 2 core nodalization using the RELAP5.

4. NEUTRONIC NODALIZATION – ANGRA 2 FIRST CORE – BOL

The reactor core contains the fuel assemblies, the control assemblies and the moderator/coolant. The Angra 2 fuel assemblies are mechanically identical. They have square cross sections and are arranged to match closely the shape of the surrounding cylindrical core barrel.

The core that was simulated in this work was the first core. It consists of fuel assemblies with three different enrichment levels of 1.90, 2.50 and 3.20 w/o ^{235}U.

The reactor power is controlled by the control assemblies and the boric acid dissolved in the coolant. Under BOL conditions, 84 fuel assemblies contain fuel rods with integrated gadolinium absorber (Gd_2O_3) as it can be seen in Fig. 4. In this way, were defined seven (6) compositions corresponding to the six possible fuel combinations.

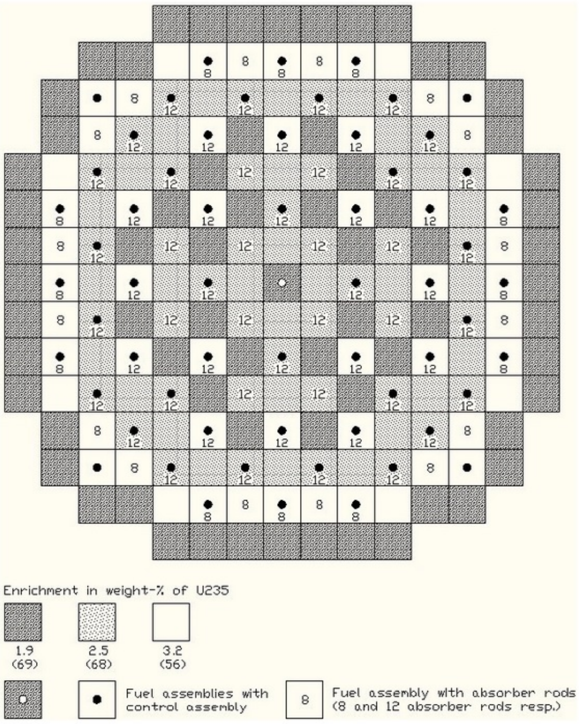

Figure 4. Cross section of the reactor core with 193 fuel assemblies (FSAR, 2013).

5. MACROSCOPIC CROSS SECTION GENERATION

The cross section libraries were generated by WIMSD-5B code (Winfrith Improved Multigroup Scheme) (Den et al., 2004) which is a general lattice cell program that uses transport theory to calculate flux as a function of energy and position in the cell.

In this work, geometry, position, composition and keeping the V_m/V_f (moderator volume/fuel volume) ratio were considered to define the cells. As output, WIMSD-5B code provides the diffusion coefficient and the macroscopic cross sections that are necessary for the neutronic code to perform the power calculation considering the feedback effects. The cross section sets generated by WIMSD-5B code were included in the PARCS model. A program developed in the Nuclear Engineering Department of UFMG (*Universidade Federal de Minas Gerais*) was used to adapt the cross sections calculated by the cell calculation code in the format required by the neutron analysis code PARCS.

To prepare the cross sections sets, 7 compositions were considered in the model (Tab. 2), being six fuel compositions and one reflector. The compositions were obtained from (FSAR, 2013).

Table 2. First core assemblies enrichment

Assemblies	Enrichment
1	Reflector
2	1.9%
3	2.5%
4	2.5% 224/12 *
5	3.2%
6	3.2% 228/8 **
7	3.2% 224/12 ***

*224 fuel rods with 2.5% and 12 fuel rods with burnable poison integrated in fuel
**228 fuel rods with 3.2% and 8 fuel rods with burnable poison integrated in fuel
***224 fuel rods with 3.2% and 12 fuel rods with burnable poison integrated in fuel

6. ANGRA 2 THERMAL HYDRAULIC RESULTS

The RELAP5 steady state calculations have been performed for the model at 3795.00 MW of power operation. The calculated temperature at the inlet and outlet of thermal hydraulic channels were compared with available FSAR data. The reactor core was firstly developed in a previous work developed by (Reis et al, 2015) to the RELAP5 MOD 3.3 and it was then adapted to the complete reactor plant nodalization developed by (Mantecón et al., 2015) to RELAP5-3D. In this way, in this new nodalization, the whole reactor is simulated with a discretized core with 37 channels to the RELAP5 MOD 3.3, as it was shown in Fig. 1. In the Table 2 are presented some calculation results compared with the reference data (FSAR, 2013).

Table 2. Comparison between steady state RELAP5 results and reference data.

	Reference	**RELAP5**	
		Channel 601	Channel 619
Coolant inlet Temperature (K)	564.30	562.82	562.82
Coolant outlet Temperature (K)	599.25	596.01	602.87
Pressure (MPa)			
Core inlet	15.93	15.75	
Core outlet	15.80	15.69	

The outlet coolant temperatures of central (618), intermediary (605) and peripheral (600) channels are presented in Fig. 5. The difference observed at the channels outlet is due to the different power distribution of each corresponding heat structure. In the Fig. 6, is shown the fuel and cladding temperatures of the heat structure of the thermal hydraulic channel 619. The values correspond with those expected.

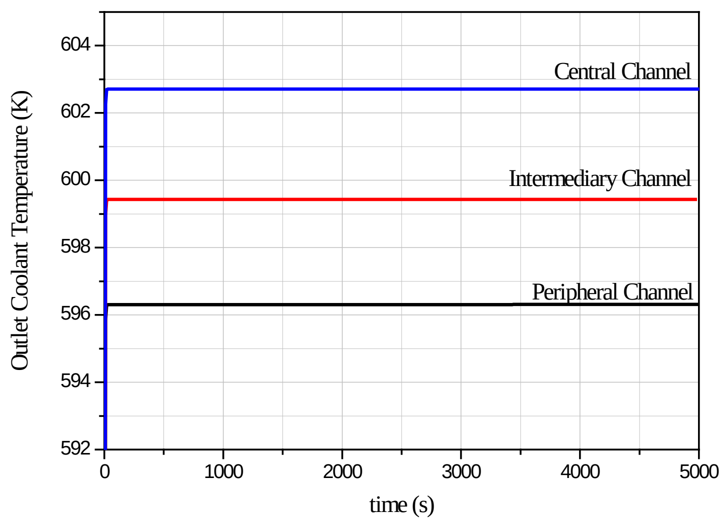

Figure 5. Coolant outlet temperature.

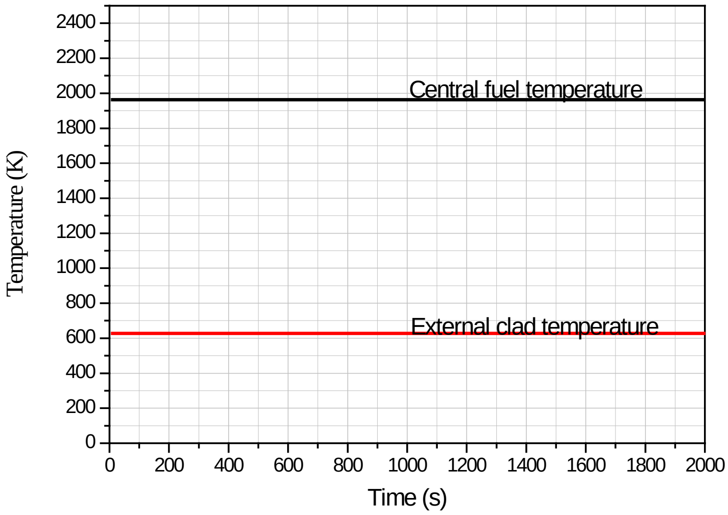

Figure 6. Fuel and clad temperature at half height of the heat structure related to the thermal hydraulic channel 619.

7. ANGRA 2 NEUTRONIC RESULTS

To obtain a configuration of planar power distribution that could reproduce approximately that from the Fig. 2, the control rods were partially inserted and after several tests, the average calculated assembly power distribution in steady state operation obtained is shown in Fig. 7. The k_{eff} obtained for this configuration was 1.011196. Figure 8 presents the axial power distribution, both considering as composition the first core, beginning of life, at left, without control rods and, at right, with control rods (partially inserted at steady state operation).

2	3	4	5	6	7	8	9	10	11	12	13	14	15	16
				0.6472	0.7933	0.8028	0.8130	0.8293	0.8291	0.6730				
		0.7505	0.9159	1.2144	1.1538	1.0537	1.1319	1.1134	1.2226	1.2623	0.9327	0.7552		
	0.7552	1.6218	1.1599	0.5734	0.9904	0.5583	0.8114	0.6415	1.0598	0.5945	1.1716	1.6218	0.7505	
	0.9327	1.1716	0.8847	0.8704	1.3274	0.9403	0.8018	0.9830	1.3852	0.8909	0.8847	1.1599	0.9159	
0.6730	1.2623	0.5945	0.8909	0.3488	1.0123	1.2136	1.2242	1.2323	1.0306	0.3488	0.8704	0.5734	1.2144	0.6472
0.8291	1.2226	1.0598	1.3852	1.0306	1.3401	1.2499	1.0437	1.2546	1.3401	1.0123	1.3274	0.9904	1.1538	0.7933
0.8293	1.1134	0.6415	0.9830	1.2323	1.2546	1.4111	1.4953	1.4111	1.2499	1.2136	0.9403	0.5583	1.0537	0.8028
0.8130	1.1319	0.8114	0.8018	1.2242	1.0437	1.4953	0.9319	1.4953	1.0437	1.2242	0.8018	0.8114	1.1319	0.8130
0.8028	1.0537	0.5583	0.9403	1.2136	1.2499	1.4111	1.4953	1.4111	1.2546	1.2323	0.9830	0.6415	1.1134	0.8293
0.7933	1.1538	0.9904	1.3274	1.0123	1.3401	1.2546	1.0437	1.2499	1.3401	1.0306	1.3852	1.0598	1.2226	0.8291
0.6472	1.2144	0.5734	0.8704	0.3488	1.0306	1.2323	1.2242	1.2136	1.0123	0.3488	0.8909	0.5945	1.2623	0.6730
	0.9159	1.1599	0.8847	0.8909	1.3852	0.9830	0.8018	0.9403	1.3274	0.8704	0.8847	1.1716	0.9327	
	0.7505	1.6218	1.1716	0.5945	1.0598	0.6415	0.8114	0.5583	0.9904	0.5734	1.1599	1.6218	0.7552	
		0.7552	0.9327	1.2623	1.2226	1.1134	1.1319	1.0537	1.1538	1.2144	0.9159	0.7505		
				0.6730	0.8291	0.8293	0.8130	0.8028	0.7933	0.6472				

Figure 7. Planar assembly power distribution – PARCS calculation.

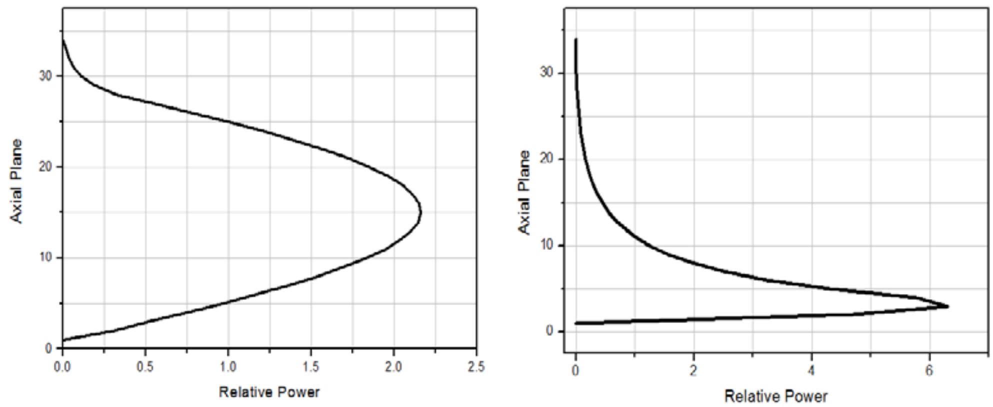

Figure 8. Core axial relative power distribution at BOL, without control rods (left) and with control rods (right) – PARCS calculation.

8. ANGRA 2 - COUPLING RESULTS

The coupling between PARCS code and RELAP5 code was done in steady state. A mapping interconnecting both codes were developed and applied to achieve the coupled situation. Preliminary results were obtained. In the Fig. 9, the fuel and cladding temperatures at half height of the heat structure of the thermal hydraulic channel 619 are shown. Fig. 10 shows the inlet and outlet temperatures of the thermal hydraulic channel 619.

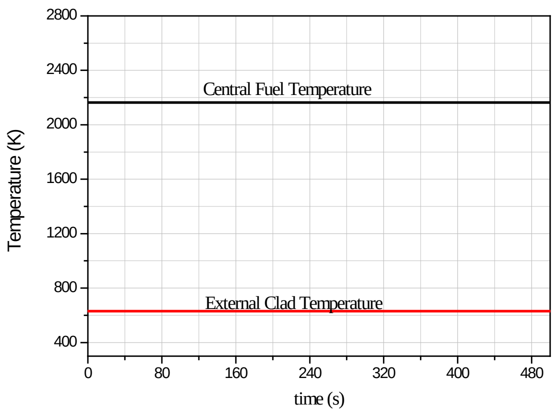

Figure 9. Fuel and clad temperature at half height of the heat structure related to the thermal hydraulic channel 619.

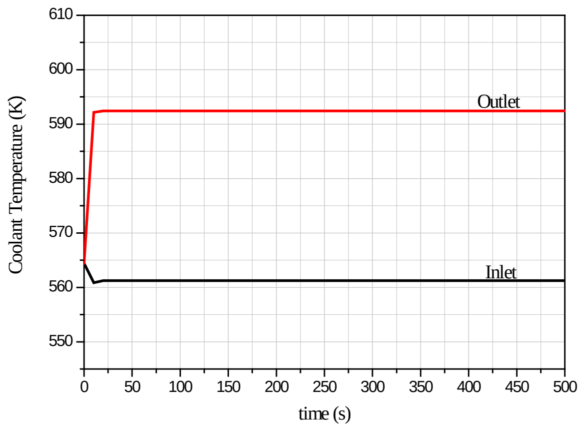

Figure 10. Inlet and outlet coolant temperatures of the thermal hydraulic channel 619.

8. CONCLUSIONS

In this work, nodalizations of the Angra 2 nuclear reactor were developed using the thermal hydraulic RELAP5 code and the neutron kinetics PARCS code to perform the models. The steady state results were presented. The macroscopic cross sections, generated by WIMSD-5B code, were adapted to obtain a compatible format for the PARCS code. To perform this adaptation, a software developed in the Nuclear Engineering Department of UFMG was used. The mapping that interconnects thermal hydraulic volumes with kinetics nodes is also ready to perform the coupling. The work presented good results, but the coupling steel need improves. After this step of coupled model verification, transient calculations will be performed.

9. ACKNOWLEDGEMENTS

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