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EVALUATION OF THERMOMECHANICAL BEHAVIOR OF NUCLEAR FUEL USING THE CODES CNFR AND FRAPCON

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Abstract. *One of the tasks of the Brazilian Nuclear Energy Commission (CNEN) is to perform independent calculations related to the thermo-mechanical behavior of the fuel elements used in the nuclear power plants of Angra I and Angra II. For this purpose, well established computer codes are used to ensure that the safety analysis reports submitted by the operator does not violate any aspect related to plant safety. In this work, the eighth recharge cycle of Angra I Power Plant was simulated using the Brazilian Code of Reactor Physics (CNFR), neutronic simulation code developed in Brazil specifically for the nuclear power plants in operation in the country and made available for use of CNEN. This cycle was chosen because the loading of nuclear fuel used was made entirely with fresh fuel. The fuel burnup simulated with the CNFR code and the power history considered were coupled to the FRAPCON code in order to simulate the thermo-mechanical behavior of the fuel rod considered. The results obtained were consistent with those reported by the operator using the Alpha/Phoenix-P/Anc (APA) methodology.*

Keywords: *nuclear fuel, CNFR code, FRAPCON code*

1. INTRODUCTION

One of the tasks of the Brazilian Nuclear Energy Commission (CNEN) is to perform independent calculations related to the thermo-mechanical behavior of the fuel elements used in the nuclear power plants of Angra I and Angra II. CNEN recently acquired the right to use two computer codes that will enable a security analysis more effective. These codes are CNFR (*Código Nacional de Física de Reatores* - Brazilian Code of Reactor Physics) and FRAPCON 3.4 (Fuel Rod Analysis) (Silva and Silva, 2005) (Geelhood et al, 2011). As in both codes the fuel rods geometry and composition can be supplied by the user, the connection between the codes FRAPCON 3.4 and CNFR occur from the history of power on a specific fuel rod calculated by the code CNFR. The second is in charge of the whole analysis of the thermomechanical behavior of the fuel rod in question. In this study the codes are briefly described. Also, a study on the feasibility of automatic coupling is realized between both. Finally, a complete simulation of the thermomechanical behavior of the fuel rod under the most severe historical power is performed in order to demonstrate that safety limits are not violated.

2. THE CNFR AND FRAPCON CODES

This section briefly described both codes used to simulate a complete cycle of a PWR nuclear reactor in operation in Brazil.

2.1 The CNFR code description

The Brazilian Code of Reactor Physics (CNFR in Portuguese) is a code developed entirely in Brazil by a multi-institutional team. It is possible to obtain an accurate analysis of the reactor cores of pressurized light water nuclear power plants in operation in the country. This code is able to simulate the behavior of these reactors in steady state, solving neutronic models, thermo hydraulic models, and isotopic decay models. From the neutronic standpoint the main advantages of CNFR code is that it uses the nodal expansion method, which is a comparative advantage to the method of finite differences to calculate pin to pin. One advantage is that the CNFR code does all calculations assuming homogeneous fuel elements, but at the end of the processing did the reconstruction of the power density in each rod of the fuel elements. The CNFR code is under license by the Brazilian Commission of Nuclear Energy (CNEN), this code is an alternative to other international codes such as the codes SAV (Siemens-KWU - Germany), BB (Westinghouse - USA), CASCADE (Framatome - France), SIMULATE (Studsvik - Sweden).

The code structure is divided into two main systems, which are the Generation Data of Nuclear (GEDAN) system and Generation Data of Reactor (GEDAR) system (Fig. 1). The GEDAN has the purpose of automatically generating a library of nuclear data in the macro groups for use in the GEDAR structure system. The GEDAR numerically solve the neutron diffusion equation by a nodal method for obtaining neutron flux and hence the power density.

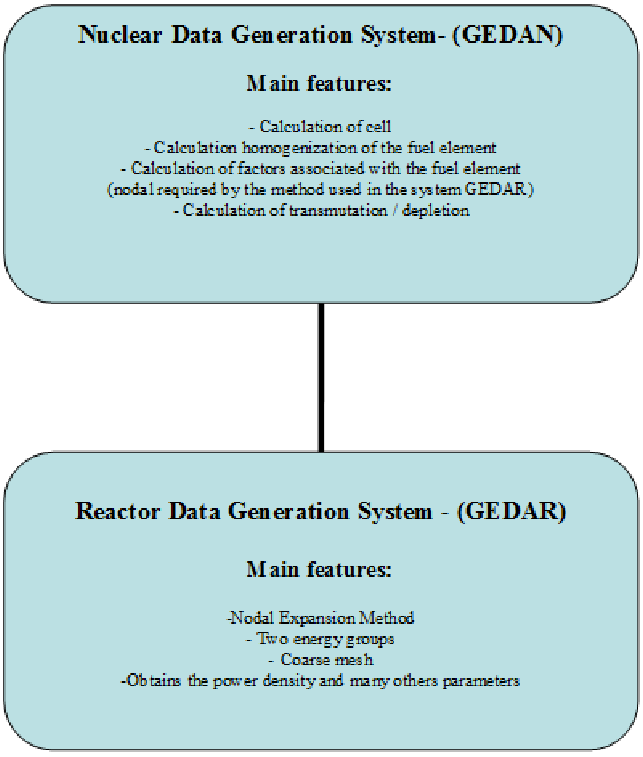

Figure 1. Simplified diagram of the CNFR code.

Another feature of the code is to provide a friendly graphical interface that allows the user to enter the data core design intuitively. Furthermore, it is possible to monitor the entire operation of the cycle in question (Fig. 2).

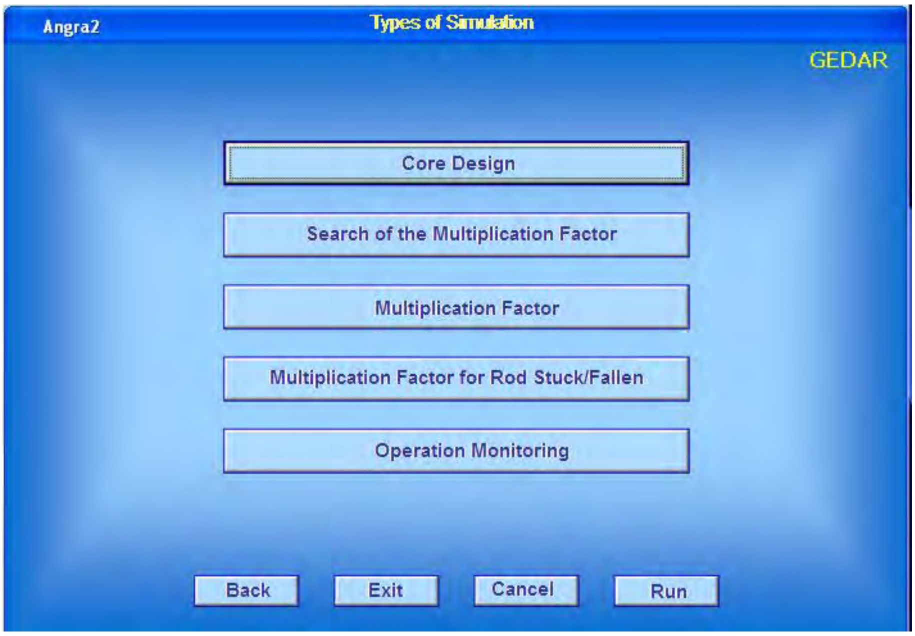

Figure 2. Options available in GEDAR module.

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2.2 The FRAPCON 3.4 code description

FRAPCON-3.4 is an analytical tool developed by Pacific Northwest National Laboratory (PNNL) that calculates LWR fuel rod behavior in "steady-state" (Fig. 3). This includes situations such as long periods at constant power and slow power ramps that are typical of normal power reactor operations. The code calculates the variation with time of all significant fuel rod variables, including fuel and cladding temperatures, cladding hoop strain, cladding oxidation, fuel irradiation swelling, fuel densification, fission gas release, and rod internal gas pressure. In addition, the code is designed to generate initial conditions for transient fuel rod analysis by FRAPTRAN, the companion transient fuel rod analysis code.

As the CNFR code the FRAPCON-3.4 has two major modules. FRAPCON-3.4 uses fuel, cladding, and gas material properties from MAPTRO that have been recently updated to include burnup-dependent properties and properties for advanced zirconium based cladding alloys. For the mechanical model, the user may select the FRACAS-I model (finite difference model) or the FEA (finite element analysis) model. The FRACAS-I model is recommended by PNNL and is the default selection.

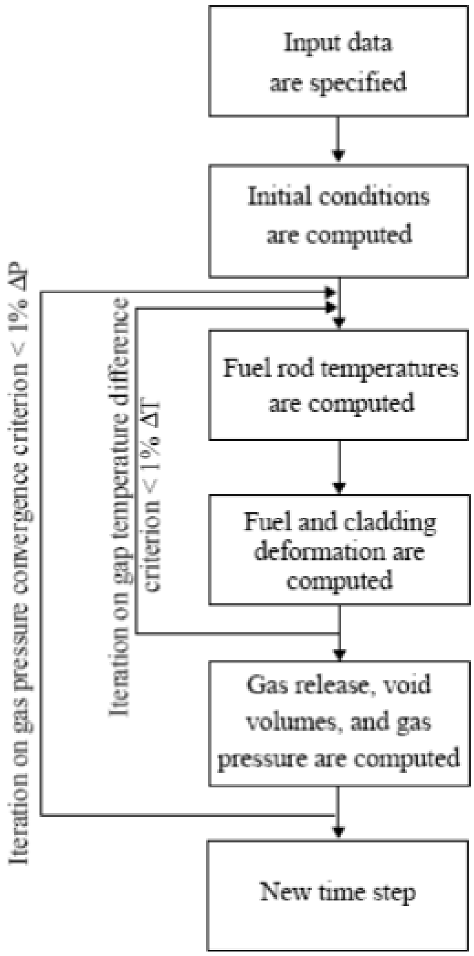

Figure 3. Simplified FRAPCON-3.4 flowchart (Geelhood et. al., 2011).

The FRAPCON code does not have a graphical interface for data entry as CNFR code. The user must enter the data in a file input for code that is divided into three distinct sections:

- FRPCN: Composed of integer variables, this section of the input file indicates how simulation is divided regarding the number of burning steps, radial edges where the calculations are made, amounts of axial regions in which the fuel rod is divided among others.
- FRaPCON: It consists of real and integer variables, and covers all engineering parameters and operation such as insert diameter, rod diameter, thickness and material of the coating among others. It is the largest section of the input file.
- EMFPCN: Indicates calculation models that will be considered.

3. THE REACTOR SIMULATION

In order to assess the accuracy of the code CNFR, the results obtained from the simulation of a cycle of operation of a PWR reactor using the code CNFR will be compared with the reference results obtained from the Alpha/Phoenix-P/Anc (APA) code. This code was chosen to proceed with a preliminary verification code CNFR because it is licensed by CNEN and is used by the operator of the Brazilian nuclear plants, with results consistent with the experimental measurements carried out during the operation.

For simplicity, it was chosen to simulate a cycle of 363 days of operation at full rated thermal input of 1876 MW fully charged with fresh fuel elements. There are three distinct regions of enrichment of the isotope ^{235}U. All elements simulated burnable poison are also fresh. The code is able to indicate the different regions of enrichment, location of control rods banks and burnable poison. An example of displaying a map of the core is shown in Figure 4, which was generated from the CNFR code.

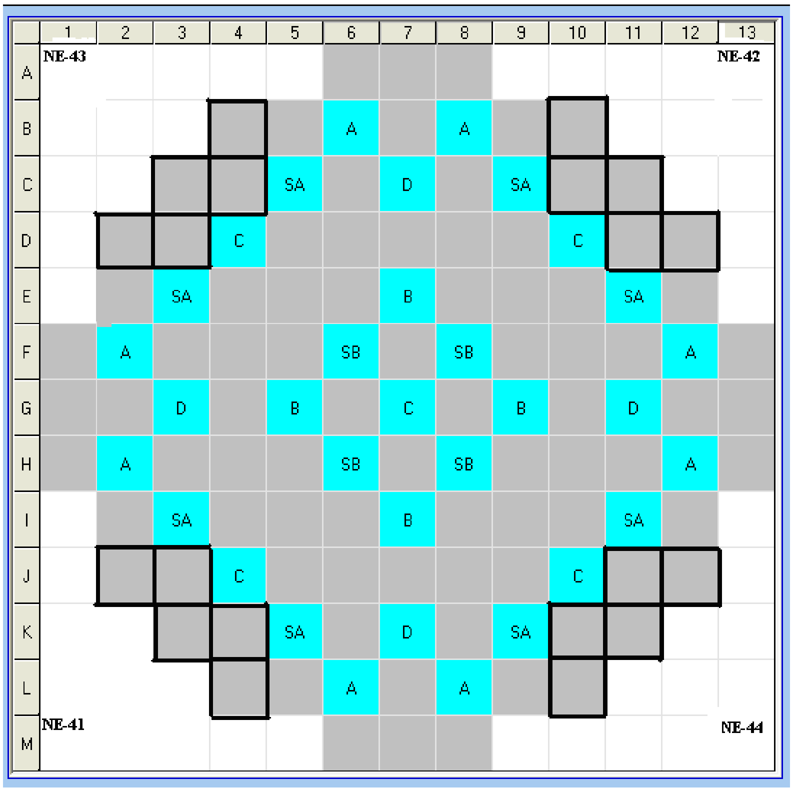

Figure 4. Visualization of the reactor core in the CNFR code.

Table 1 shows the data of fuel rod design under consideration.

Fuel	UO_2
Cladding	Zircaloy – 4
Pellet diameter	8.192 mm
Inner diameter of cladding	8.357 mm
Fuel density	95% of theoretical density
Active fuel length	365.8 cm
Internal pressure (He)	3.103 MPa
Coolant inlet temperature	549.5 ^{0}C
Average linear heat generation rate	17.55 W/m

Table 1. Geometry and composition for the fuel rod simulated.

From the specifications shown on Table 1 simulations were made for the behaviour of the central temperatures in the fuel rod, of the external temperature of the UO_2 pellet, and the mean temperature, also in the region of the fuel pellet during the burnup process at a linear power rated provided from the CNFR code for 363 days of non-stop operation.

ISSN 2176-5480

The fuel rod was axially sub-divided into 7 equally-spaced intervals and where the central interval analysis was considered, that is, of the fourth interval, contained between the quotas [1.829m, 2.351m].

4. PRELIMINARY RESULTS

The first step before discussing the coupling between the codes CNFR and FRAPCON is to check if the new code provides good results in neutron reactor operation against the APA code.

4.1 Verification of the CNFR code

From the description of the cycle presented in the previous section, we compared the results obtained by the CNFR codes with the reference method (APA). In Figures 5 to 7 are the percentage deviations relative to the reference method (APA) of the average power in each element considering symmetry of ¼ of core at different instants of burning in a scheme to full power (FPD – Full Power Days).

	Step 1 - Burnup - 0 MWD/MTU - 0 FPD								Step 2 - Burnup - 113 MWD/MTU - 3 FPD						
	7	8	9	10	11	12	13		7	8	9	10	11	12	13
G	0.3	0.2	0.6	0.7	0.7	2.3	0.6	G	1.8	0.0	0.7	0.1	0.3	0.7	1.3
H	0.2	0.3	0.9	0.4	0.4	1.9	1.4	H	1.8	1.2	0.5	0.2	0.0	0.4	2.7
I	0.6	0.9	0.6	0.3	0.9	0.1		I	0.7	0.5	0.4	0.1	0.3	0.4	
J	0.7	0.4	0.3	0.5	0.9	3.0		J	0.1	0.2	0.1	0.1	0.3	3.2	
K	0.7	0.5	1.0	1.0	2.3			K	0.3	0.1	0.3	0.4	2.4		
L	2.3	1.9	0.3	3.0				L	0.6	0.4	0.2	3.2			
M	0.4	1.7			Average % =	0.9		M	1.4	2.8			Average % =	0.8	
	Step 3 - Burnup - 226 MWD/MTU - 6 FPD								Step 4 - Burnup - 755 MWD/MTU - 20 FPD						
	7	8	9	10	11	12	13		7	8	9	10	11	12	13
G	0.3	0.4	0.4	0.6	0.1	1.1	0.1	G	0.6	0.6	0.2	0.4	0.1	0.9	0.1
H	0.4	0.2	0.6	0.5	0.1	1.0	1.2	H	0.6	0.2	0.3	0.4	0.1	0.7	1.2
I	0.4	0.6	0.5	0.2	0.4	0.3		I	0.2	0.3	0.3	0.2	0.3	0.2	
J	0.6	0.5	0.2	0.2	1.0	1.8		J	0.4	0.4	0.2	0.2	0.9	1.6	
K	0.1	0.1	0.5	1.0	1.2			K	0.0	0.1	0.3	0.9	0.9		
L	1.1	1.0	0.6	1.8				L	0.8	0.7	0.4	1.8			
M	0.0	1.3			Average % =	0.6		M	0.2	1.5			Average % =	0.5	

Figure 5: Deviation between the results obtained for the average power calculated with the CNFR code and the Alpha/Phoenix-P/Anc methodology in the beginning of cycle.

	Step 5 - Burnup - 1509 MWD/MTU - 40 FPD								Step 6 - Burnup - 3018 MWD/MTU - 80 FPD						
	7	8	9	10	11	12	13		7	8	9	10	11	12	13
G	0.1	0.0	0.4	0.4	0.0	0.9	0.0	G	0.4	0.3	0.4	0.2	0.0	0.5	0.3
H	0.0	0.3	0.4	0.4	0.1	0.7	1.1	H	0.3	0.4	0.3	0.2	0.1	0.5	1.0
I	0.4	0.4	0.4	0.0	0.5	7.2		I	0.4	0.3	0.2	4.5	0.4	0.2	
J	0.4	0.4	0.0	0.4	1.1	1.3		J	0.2	0.2	4.4	0.4	1.1	1.0	
K	0.1	0.2	0.5	1.2	0.6			K	0.0	0.1	0.5	1.1	0.3		
L	0.9	0.7	0.5	1.3				L	0.5	0.6	0.4	1.0			
M	0.1	1.4			Average % =	0.7		M	0.3	1.3			Average % =	0.7	
	Step 7 - Burnup - 4527MWD/MTU - 120 FPD								Step 8 - Burnup - 6036 MWD/MTU - 160 FPD						
	7	8	9	10	11	12	13		7	8	9	10	11	12	13
G	0.1	0.1	0.1	0.0	0.0	0.2	0.7	G	0.4	0.4	0.1	0.1	0.0	0.1	0.7
H	0.1	0.1	0.1	0.1	0.1	0.1	1.3	H	0.4	0.3	0.1	0.1	0.1	0.1	1.2
I	0.1	0.1	0.1	0.1	0.3	0.0		I	0.1	0.1	0.0	0.1	0.3	0.1	
J	0.0	0.1	0.1	0.4	0.8	1.2		J	0.1	0.1	0.1	0.5	0.8	0.5	
K	0.0	0.1	0.3	0.8	0.3			K	0.0	0.2	0.4	0.8	0.0		
L	0.1	0.1	0.1	1.2				L	0.1	0.1	0.2	0.8			
M	0.7	1.7			Average % =	0.3		M	0.7	1.3			Average % =	0.3	

Figure 6: Deviation between the results obtained for the average power calculated with the CNFR code and the Alpha/Phoenix-P/Anc methodology in the middle of cycle.

ISSN 2176-5480

Step 10 - Burnup - 9054 MWD/MTU - 240 FPD								Step 11 - Burnup - 10562 MWD/MTU - 280 FPD							
	7	8	9	10	11	12	13		7	8	9	10	11	12	13
G	0.6	0.8	0.3	0.6	0.2	0.0	0.3	G	0.4	0.6	0.2	0.8	0.3	0.2	0.5
H	0.8	0.4	0.7	0.2	0.4	0.3	0.2	H	0.6	0.3	0.7	0.2	0.6	0.2	0.7
I	0.3	0.7	0.1	0.4	0.3	0.6		I	0.2	0.7	0.1	0.7	0.1	0.8	
J	0.6	0.2	0.4	0.4	0.9	0.8		J	0.8	0.2	0.7	0.2	0.7	1.4	
K	0.2	0.4	0.3	0.9	1.4			K	0.3	0.5	0.1	0.7	1.8		
L	0.0	0.3	0.8	0.6				L	0.2	0.2	0.9	1.4			
M	0.3	0.0			Average % =	0.4		M	0.5	0.7			Average % =	0.5	
Step 12 - Burnup - 12071 MWD/MTU - 120 FPD								**Step 8 - Burnup - 13700 MWD/MTU - 363 FPD**							
	7	8	9	10	11	12	13		7	8	9	10	11	12	13
G	0.4	0.6	0.3	1.0	0.4	0.3	0.9	G	0.6	1.0	0.4	1.3	0.6	0.4	1.6
H	0.6	0.3	0.8	0.2	0.8	0.1	1.5	H	1.0	0.4	1.1	0.4	1.0	0.0	2.6
I	0.3	0.8	0.2	0.9	0.0	1.0		I	0.4	1.1	0.3	1.2	0.1	1.3	
J	1.0	0.2	0.9	0.1	0.6	2.0		J	1.3	0.4	1.2	0.0	0.5	3.1	
K	0.4	0.8	0.1	0.6	2.4			K	0.6	1.0	0.1	0.5	3.2		
L	0.3	0.1	1.1	2.0				L	0.4	0.1	1.5	2.9			
M	0.9	1.3			Average % =	0.7		M	1.6	2.4			Average % =	1.0	

Figure 7: Deviation between the results obtained for the average power calculated with the CNFR code and the Alpha/Phoenix-P/Anc methodology in the end of cycle.

From the percentage deviations shown in Figures 5 to 7 it can be concluded that the code presents results consistent with the EPA, which in turn agrees with the experimental data obtained directly from Power Plant.

Another important aspect is the position of the maximum value and the power density in the core, which can be seen in Table 2:

Table 2. Position and value of maximum and minimum power density at 0 FPD (0MWD/MTU) .

Code	Position of maximum power density	Value of maximum power density	Position of minimum power density	Value of minimum power density
APA	G12	1.261	J12	0.655
CNFR	G12	1.291	J12	0.635

The value obtained for the maximum power density will be multiplied by the Average linear heat generation rate to determine the maximum linear heat generation rate that will be simulated in FRAPCON code. In Figure 8 can be seen the deviation between the boron concentrations simulated throughout all operation cycle.

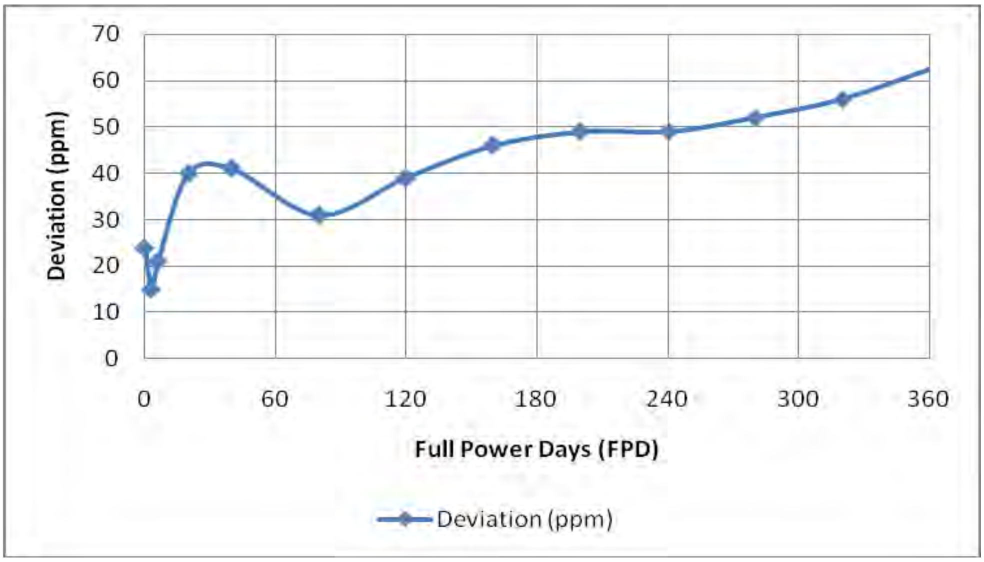

Figure 8: Deviation between the results obtained for the concentration of boro calculated with the CNFR code and the Alpha/Phoenix-P/Anc methodology during all cycle.

ISSN 2176-5480

From Figure 8 it can be concluded that the difference between the simulations exceeds 50 ppm after 240 days of operation. The next step of the study would be to compare directly with the data measured and stored on your computer's process and complete Power plant which codes provides the best prediction.

In Figure 9 you can display the percentage deviation between the simulated burning by both codes at the end of the operating cycle.

Deviation (%) - 13700 MWD/MTU - 363 FPD							
	7	8	9	10	11	12	13
G	0.5	0.6	0.4	0.7	0.3	2.6	0.1
H	0.6	0.4	0.6	0.4	0.4	0.1	0.5
I	0.5	0.7	0.3	0.5	0.0	0.3	
J	0.7	0.4	0.5	0.1	0.6	0.1	
K	0.3	0.4	0.0	0.6	0.5		
L	0.1	0.0	0.5	0.1			
M	0.2	0.7			Average % =	0.4	

Figure 9: Deviation between the results obtained for the average burnup calculated with the CNFR code and the Alpha/Phoenix-P/Anc methodology in the end of cycle.

From the percentage deviations shown in Figure 9 we conclude that both have models depletion codes agree with each other.

4.2 The simulation using the FRAPCON code

This section will present the results obtained with the FRAPCON code to the rod exposed to the most severe historical power calculated with the CNFR code. This rod is in the G12 element and the maximum linear heat generation rate is considered $q_{Max}' = 1.291 \times 17.55$ W/m = 22.66 W/m. For comparison, we also consider the rod located at the position J12, which presents the minimum linear heat generation rate of $q_{Min}' = 0.635 \times 17.55$ W/m = 11.15 W/m.

In Figure 10 may visualize the temporal evolution of centreline temperature of both fuel rods simulated using the FRAPCON code.

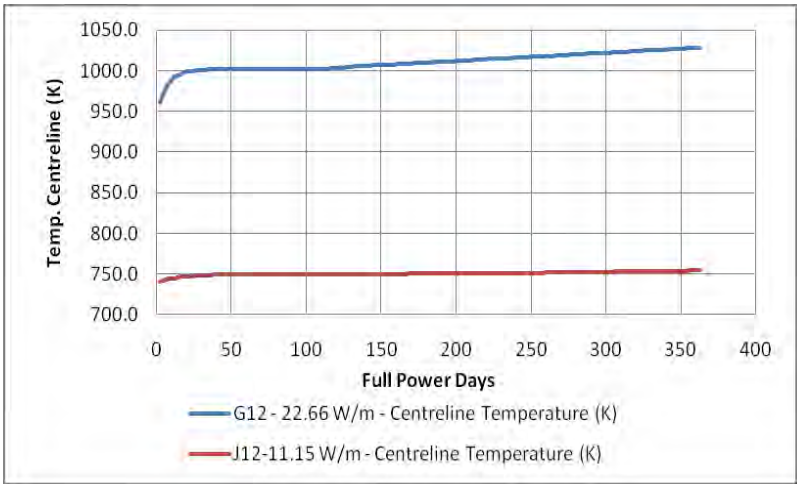

Figure 10: Evolução temporal da temperatura na linha central das varetas combustíveis simuladas com o código FRAPCON.

In the graph of Figure 10 is possible to visualize the influence of the power generated in the centreline temperature for identical rods in fuel elements located relatively close (G12 and J12). It is possible also see a slight increase in the core temperature of the rod G12 after 150 FPD, which does not occur in the rod subjected to a history of lower power.

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In Figures 11, 12 and 13 are graphs of temporal evolution, respectively, the thickness of the cladding layer of ZrO_2, in the H_2 concentration in the cladding and cladding axial strain in both simulated fuel rods using the FRAPCON code.

Figure 11: Temporal evolution of the ZrO_2 thickness in the cladding of fuel rods simulated with the FRAPCON code

Figure 12: Temporal evolution of the thickness of H_2 concentration in the cladding of the fuel rods simulated with the FRAPCON code

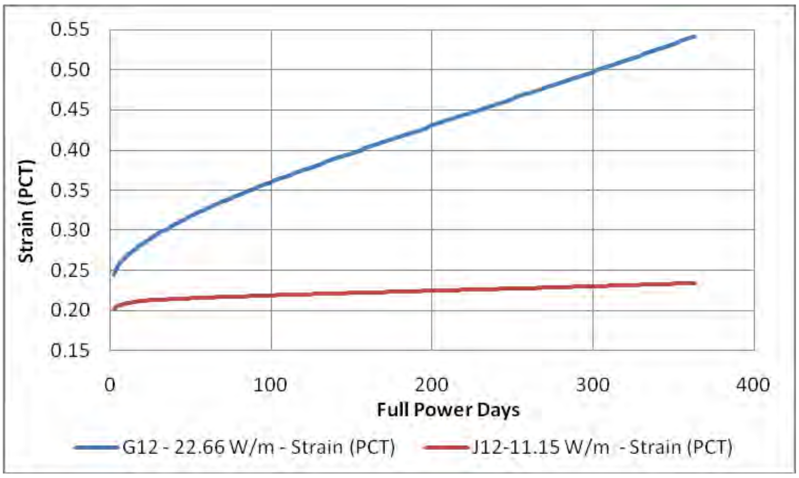

Figure 13: Temporal evolution of the cladding axial strain thickness of the fuel rods simulated with FRAPCON code

Figure 14 has a summary of some important safety parameters that are provided directly by the output file FRAPCON code.

```
                 G12 - 22.66 W/m                                                  J12 - 11.15 W/m
xxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxx xxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxx
x                Rod Internal Pressure                       x x             Rod Internal Pressure                      x
x                                                            x x                                                        x
x   Initial Cold Fuel Rod Plenum Volume =    1.22606 in^3    x x  Initial Cold Fuel Rod Plenum Volume =    1.22606 in^3    x
x                                                            x x                                                        x
x   Maximum Fuel Rod Internal Pressure  =    2925.28 psi     x x  Maximum Fuel Rod Internal Pressure  =    2634.01 psi     x
x            Peak nodal burnup          =       0.09 GWd/MTU x x           Peak nodal burnup          =       0.05 GWd/MTU x
x            Rod average burnup         =       0.06 GWd/MTU x x           Rod average burnup         =       0.03 GWd/MTU x
x            Fuel rod void volume       =    1.04528 in^3    x x           Fuel rod void volume       =    1.07094 in^3    x
x            Fission gas release        =       0.00 %       x x           Fission gas release        =       0.00 %       x
x            Time                       =      2.000 days    x x           Time                       =      2.000 days    x
xxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxx xxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxx
x                 Centerline Temperature                     x x              Centerline Temperature                    x
x                                                            x x                                                        x
x   Maximum Fuel Centerline Temperature =    2044.94 deg.F   x x  Maximum Fuel Centerline Temperature =    1156.43 deg.F   x
x            Axial node                 =          6         x x           Axial node                 =          6         x
x            Nodal burnup               =      16.66 GWd/MTU x x           Nodal burnup               =       8.20 GWd/MTU x
x            Rod average burnup         =      11.10 GWd/MTU x x           Rod average burnup         =       5.46 GWd/MTU x
x            Time                       =    363.000 days    x x           Time                       =    363.000 days    x
xxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxx xxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxx
x                 Strain Increment                           x x              Strain Increment                          x
x                                                            x x                                                        x
x   Maximum Strain Increment(elas+plas) =  0.009863 %        x x  Maximum Strain Increment(elas+plas) =  0.003469 %        x
x            Axial node                 =          7         x x           Axial node                 =          6         x
x            Nodal burnup               =       0.18 GWd/MTU x x           Nodal burnup               =       0.09 GWd/MTU x
x            Rod average burnup         =       0.12 GWd/MTU x x           Rod average burnup         =       0.06 GWd/MTU x
x            Time                       =      4.000 days    x x           Time                       =      4.000 days    x
xxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxx xxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxxx
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Figure 14: Regulatory Summary of important safety parameters.

5. CONCLUSION

The results presented in this paper show that the code CNFR is able to simulate the behavior of a thermohydraulic and neutronic operational cycle of a typical PWR reactor loaded with fresh fuel. It is a code that is promising and can be a national alternative compared to other options developed and sold by other countries. From the distribution of nuclear power obtained from the CNFR code, was inserted manually linear rated power at each burning step in the corresponding FRAPCON 3.4 code in order to simulate the fuel rods thermomechanical behavior. The results for key safety parameters were not violated. These preliminary results are important from the standpoint of permitting and automatic coupling between the codes shown a challenge which saves man-hours and automatically simulate allow more fuel rods simultaneously.

6. ACKNOWLEDGEMENTS

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