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IMPLEMENTATION OF THE OPTIMIZATION FOR THE METHODOLOGY OF THE NEUTRONIC CALCULATION AND THERMO-HYDRAULIC IN IEA-R1 REACTOR

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ABSTRACT

This work objective was to create a manager program that would automate the programs and computer codes in use for neutronic calculation and thermo-hydraulic in IEA-R1 reactor thus making the process for calculation of safety parameters and for configuration change up to 98% faster than that used in the reactor today. This process was tested in combination with the reactor operators and is being implemented by the quality department. The main codes and programs involved in the calculations of configuration change are LEOPARD, HAMMER-TECHNION, TWODB, CITATION and COBRA . Calculations of delayed neutron and criticality coefficients given in the process of safety parameters calculation are given by the HAMMER-TECHNION and CITATION in a process that involves about eleven repetitions so that it meets all the necessary conditions (such different temperatures of the moderator and fuel). The results are entirely consistent with the expected and absolutely the same as those given by manual process. Thus the work shows its reliability as well the advantage of saving time, once a process that could take up to four hours was turned in one that takes around five minutes when done in a home computer. Much of this advantage is due to the fact that were created subprograms to treat the output of each program used and transform them into the input of the other programs, removing from it the intermediate essential data for this to occur, thus avoiding also a possible human error by handling the various data supplied.

1. INTRODUCTION

The IEA-R1 is a pool type reactor, moderate and cooled by light water. It uses graphite and beryllium as reflector and its fuel is plate type. It was designed and built by U.S. company Babcock & Wilcox in 1956 and its first criticality occurred on September 16, 1957 [1]. In the last ten years the reactor has operated 64 hours a week in powers ranging between 2 and 5 MW thermal, to meet various internal and external customers of IPEN.

There are many users of the IEA-R1 reactor, as well as different purposes for the reactor operation. In order the meet the operational demand, the core configuration is frequently changed replacing highly burned fuels by the low burn up ones.

This configuration change of fuel elements from reactor core is calculated based on a methodology developed by the division of reactor physics at the Center for Nuclear Engineering. The calculation methodology developed employs a sequential execution of various codes and computer's programs of reactors physics and thermo-hydraulics areas, so that in the end of calculations some criteria must be satisfied so that the new calculated configuration can be approved.

1.1 Neutronic Calculation Methodology

The neutronic calculation methodology from the Reactors Physics Division is based on the following programs: LEOPARD [2] and TECHNION-HAMMER [3] for cross sections generation, TWODB [4] to calculate the core and fuel combustion in two dimensions and CITATION [5] to calculate the core in three dimensions.

Figure 1 shows a calculation diagram [6]. The generation of the fuel cross sections is performed using the LEOPARD program, using the standard model of the cell (fuel, cladding and moderator) with extra region to homogenize the fuel element. The cell data are provided in the LEOPARD.DAT file and the LEOPARD.SAI and LEOPARD.OUT output files are generated. The cross sections are generated with a range of approximately 30 days of burning until get to 50% of U-235 burned and are recorded in the unformatted file named LEOPARD.BIN. The program called LINXS converts this file in a compatible format with the 2DB program and save the unformatted file LINXS.BIN. The number of groups of the cross section of the fuel elements is provided in the LINXS.DAT input file. The TWOLIB.DAT formatted file is generated by the CONVERD program. This step is necessary to merge files from several types of fuels. The conversion to the file TWODB.LIB unformatted is made by the program CONVERB. This is the library of the cross sections of the fuel elements to be used by TWODB.

For non-fuel cross sections, use the HAMMER-Technion program, since the LEOPARD program only allows cell pattern modeling with extra region. The macroscopic cross sections of these regions are provided through TWODB.DAT file with other data from the reactor to TWODB program.

The reactor operation is simulated by the TWODB program in two dimensions without control rods. The results of the program recorded in the TWODB.OUT output file are: the effective multiplication factor, the burning fuel as fissile material percentage, neutron fluxes, average power densities and macroscopic cross sections (they are also recorded in the TWODB30.BIN unformatted file).

In configurations changes, the gain of reactivity is calculated with the TWODB program in two dimensions with control rods and the macroscopic cross-sections recorded by the TWODB program (in the file TWODB30.BIN), converted to the TDBCIT.DAT formatted file by the TDBCIT1 program. These cross sections are provided to the CITATION program by the CITATION.DAT file along with reactor's core data. In three dimensions, the effective multiplication factors are calculated without control rods for determination of reactivity

excess (FORT.51 output file). With the rods in the critical position are calculated the distributions of neutron fluxes and power densities, which are stored in the FORT.36 and FORT.37 unformatted files respectively. The program reads the DENS FORT.37 and allowed the power density distribution in a suitable format for a thermo-hydraulic analysis of reactor's core.

Finally, to obtain the thermo-hydraulic datas of reactor's channels we use the COBRA [7] computer program. But first, you must use the DENS_BE program, which reads the output of CITATION program and generates an axial power profile that will be used as input of the COBRA program.

For the cross sections generation with the LEOPARD program, we considered the average temperatures on 2 MW power operations, as follows: 315 K for fuel, 314 K to cladding and 305 K for the coolant. For determination of reactivity excess to cold (293 K) we added the potency defect in the value of reactivity excess calculated at the operating temperatures.

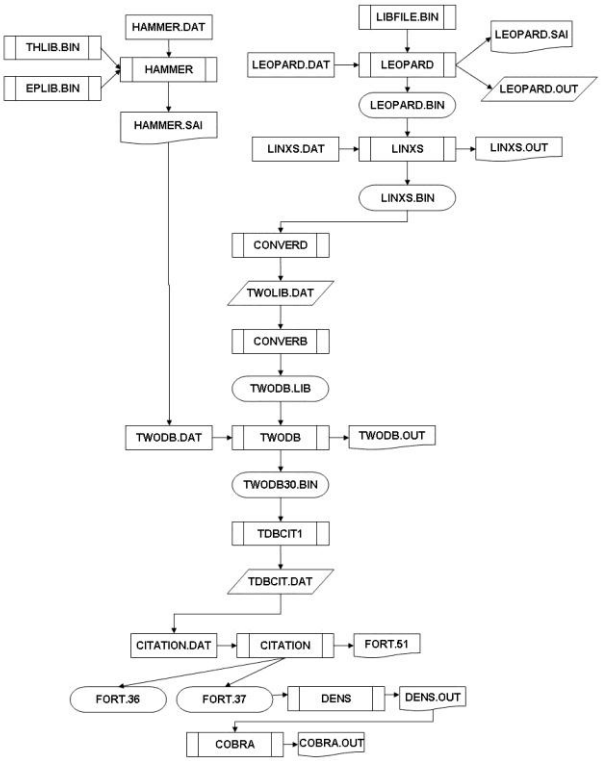

Figure 1. Diagram of Neutronic Calculation

1.2 Effective Kinetic Parameters - Theoretical foundations

1.2.1 Emission of neutron in fission [8]

The nuclear fission phenomenon is characterized not only by the fission products formation but also by the simultaneous emission of some gamma rays and neutrons.

The neutrons emitted as a result of the fission process can be classified into two types: prompt neutron and delayed neutron. Prompt neutron, which constitute about 99% of all fission neutron are emitted within a very short time interval elapsed after the fission, the order of 10^{-14} seconds.

It is known that the thermal neutrons are more effective in nuclear reactions in the most used elements. Although the production of thermal neutrons is less than 1% of the total number of emitted neutrons, these thermal neutrons have a strong influence on the time behavior of the system, playing an important role in controlling the system.

1.2.2 Methodology for calculating the reactor temperature coefficient

Temperature variations on a core of a reactor affect the value of criticality reactor (K) by the change in reactivity of the core components, which in turn alter the microscopic cross-sections (Dopller Effect).

Usually it is considered for simplicity only study these changes in water and nuclear fuel. Thus the effects of temperature changes are the dominant resonance absorption (Dopller effect) due to changes in fuel and energy spectrum of neutrons caused by changes in the density of the moderator. Thus this coefficient called temperature coefficient α_T depends on the variation of the reactor criticality (k) and temperature (T) variation.

$$\alpha_T = \frac{1}{K}\frac{dK}{dT} \tag{1}$$

The methodology for calculating the reactor temperature coefficient is based on the TECHNION-HAMMER program for generation of cross sections, the TWODB program to calculate the core and fuel combustion in two dimensions homogeneous model and the CITATION program, which in this case works in two dimensions interesting in this case only the cross-sections, no matter the depth control bars. This is calculated for different temperatures of the moderator and fuel. And it says a security parameter for the Alpha must always be less than zero to ensure the safety of reactor. Figure 2 shows the diagram of this calculation.

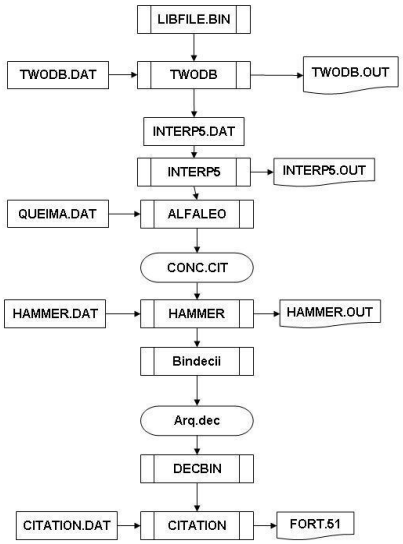

Figure 2. Calculation of the reactor criticality (multiplication factor)

1.2.3 Methodology for delayed neutrons calculation:

The methodology of delayed neutrons calculation is based on the Technion-HAMMER program to generate cross-sections, the TWODB program to calculate the behavior of the core and burning of the fuel element in two dimensions for the homogeneous model and the CITATION program in this case works in two dimensions, because for this model only interests the cross sections and not the depth of the control bars, similar to the calculation of the reactor criticality coefficient but here the thermal neutrons are part of interest. Figure 3 shows the diagram of this calculation.

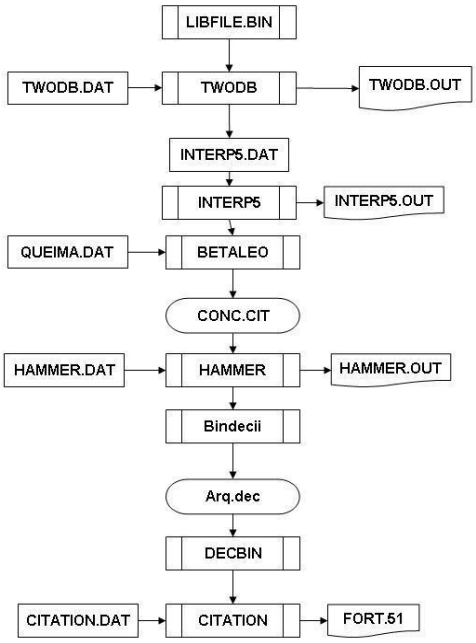

Figure 3. Calculation of delayed neutrons

2. THE OPTIMIZATION OF THE METHODOLOGY

To accomplish the proposed automation project was created a main program that manages the others. With the main program creation, all others become subroutines and several modifications were made to them to fit it. Some programs were developed for the treatment of input and output data.

For the creation of input used by CITATION program that calculates burning in three dimensions, it had to be created a new program to generate its input file from the results obtained by TWODB program, for this new on was given the name of CRINP2BCIT.

Other programs were also designed to vary the input and output files of the HAMMER program, and they're respectively called ELEMENHAM and ELEMENHAMSAI. To make the connection between HAMMER and TWODB was also necessary to create a program to modify TWODB input and it was named HAMIPUT2DB.

The INPUTCOBRA program was also and it builds the COBRA input file by using data of DENS output. By averaging the calculation of the potential for normalized axial element that is given by the output of DENS and putting it in COBRA input file along with other factors such as operating potency.

Thus the new flowchart for reactor simulation becomes as shown below (the new programs are seen in red), in Figure 4:

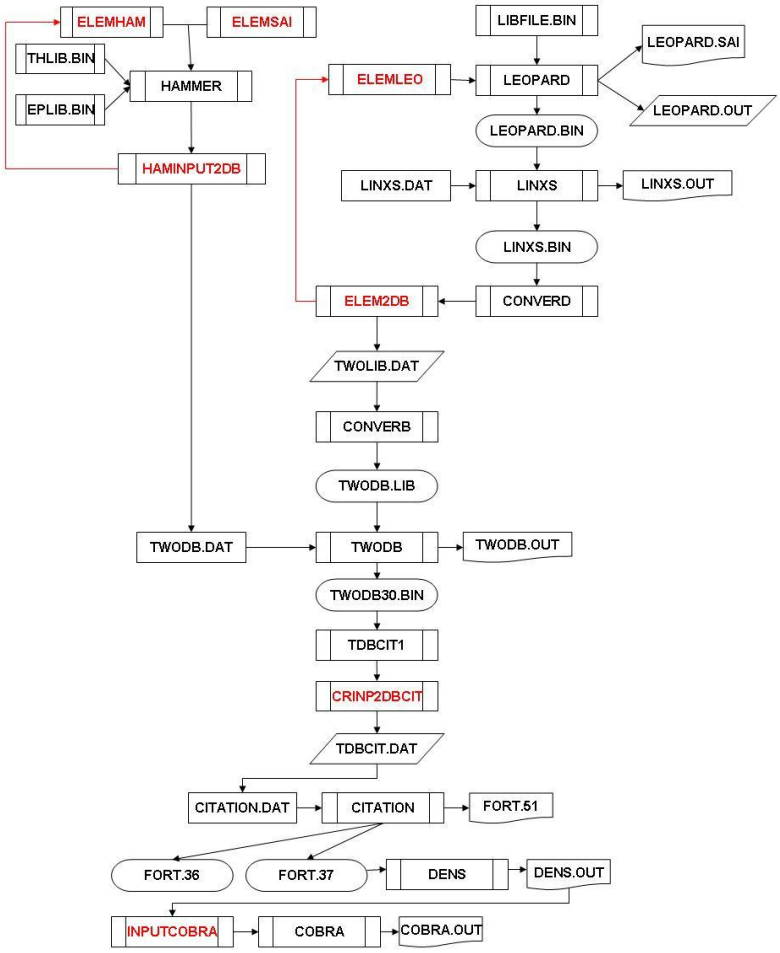

Figure 4. New flowchart for neutronic calculation

For the delayed neutron calculations was made a program that creates the input files for INTERP5, CITATION AND BETALEO and they're respectively called: Inputinterp5-2, INPUTQUEIMA and INPUTCITBETA. It was also done, a sequence in BAT of the program,

rather than a manager program, because of the short time till the date of the reactor configuration's next change, in which we're interested in testing the overall operation of the program.

The same was done for the calculation of reactor reactivity coefficients (ALFALEO), which in turn was run in eleven different configurations to do all security parameters be considered. And for these parameters, were also made programs to create input files for CITATION and ALFALEO being respectively named as INPUTQUEIMA and INPUTCITBETA. To calculate the multiplication factor and delayed neutrons, the flowchart became as shown below (the new programs are seen in red) in Figure 5:

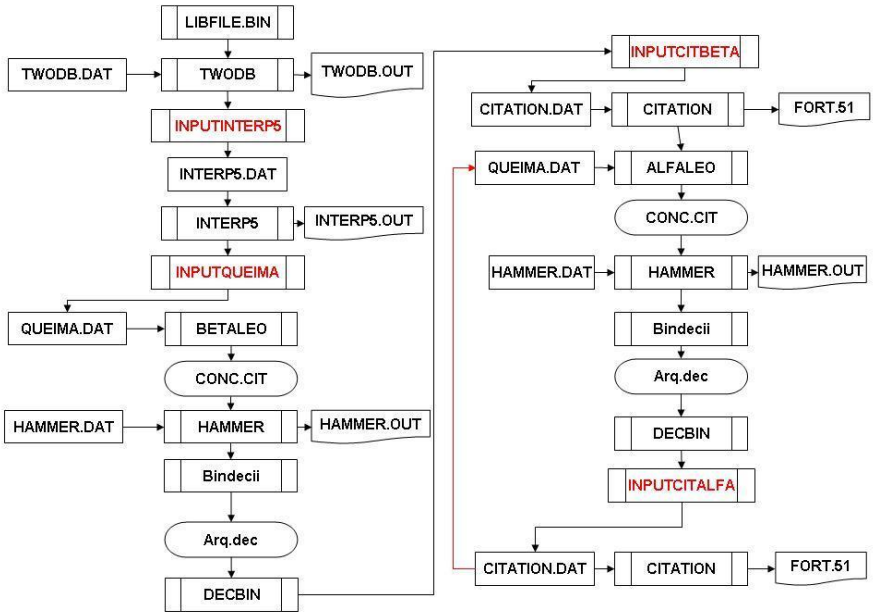

Figure 5. New Flowchart for calculation of the multiplication factor for the different parameters and delayed neutrons

2.1. Optimization results

The program was tested comparing the results of manual calculations for two configurations of the nuclear reactor IEA-R1, 243A and 245 (see appendix), with the results of this optimization program developed.

To ensure that the program works as expected it was compared to the output of Citation (FORT.51) and analyzed the multiplication factor. This factor (K) is responsible for defining if a reactor is subcritical, when k <1, critical, when k = 1 or supercritical, when k> 1. The reactor's criticality is related to several parameters such as temperature, which directly affects the absorption of neutrons by nuclear fuel or even the nuclear fuel burn up that generates fissile products that absorb neutrons.

By solving the diffusion equation (see appendix) the program Citation obtains a multiplication factor for each interaction of criticality. Is expected that these values are identical to the calculated using the conventional method till the last decimal place, since the algorithm of the program was unchanged for the resolution.

Below is shown a graph of the multiplication factor by number of interaction in the old methodology (manual) and in the new one (optimized):

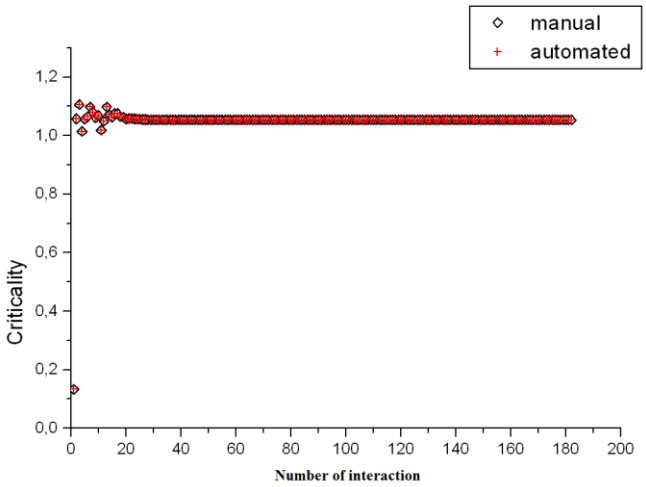

Figure 6. The multiplication factor by number of interaction for the configuration 243A of the IEA-R1 Reactor – May 2010.

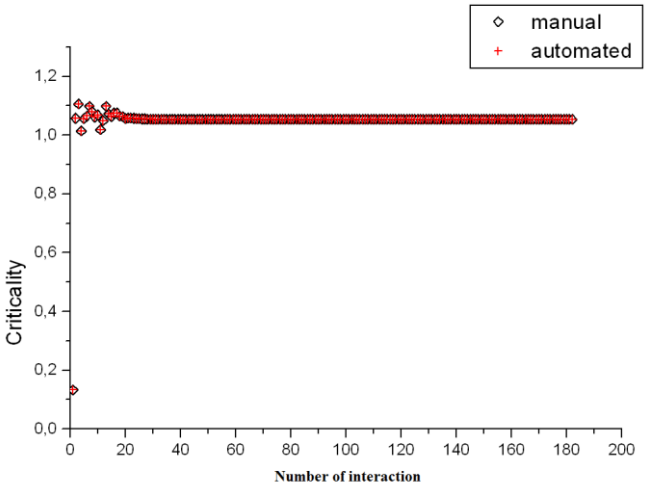

Figure 7. The multiplication factor by number of interaction for the configuration 245 of the IEA-R1 Reactor - January 2011.

Another parameters used to compare the simulation's functionality of reactor's configuration were the obtained data by COBRA program related to cladding temperature. This is one of

the most important safety parameters of a reactor, in order it can never exceed 368.15 K or 203 °F or 95ºC. Above this temperature the aluminum of rod's cladding starts to corrode faster and the reactor can be compromised. Below in Figures 8 and 9, a graph of the temperature variation along the fuel element's cladding.

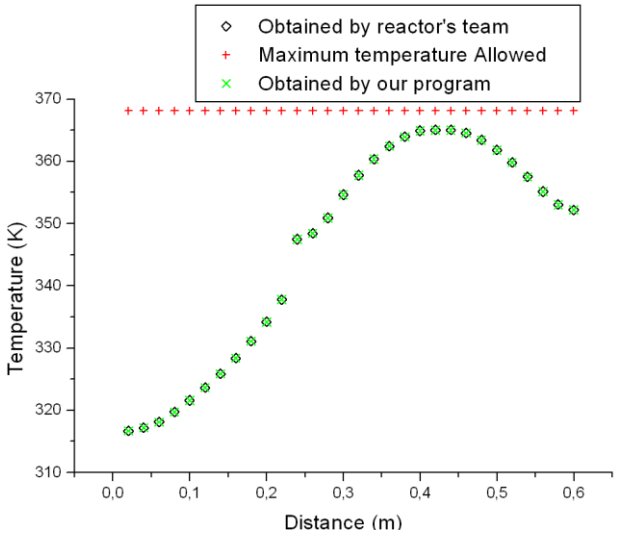

Figure 8. The temperature variation along the fuel element's cladding for the configuration 243A of the IEA-R1 Reactor – May 2010.

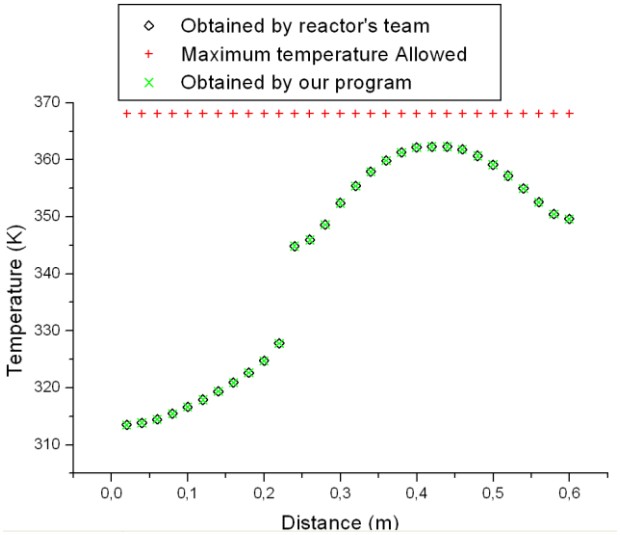

Figure 9. The temperature variation along the fuel element's cladding for the configuration 245 of the IEA-R1 Reactor - January 2011

In calculations made for the second part of the project, referring to the delayed neutrons and reactor reactivity's coefficients the results were very satisfactory as seen in the tables 1 and 2.

Table 1. Fraction of delayed neutrons per family for the configuration 243A of the IEA-R1 Reactor – May 2010

Fraction of delayed neutrons per family	
Obtained by reactor's team	Obtained by our program
2.87638×10^{-04}	2.87638×10^{-04}
1.56547×10^{-03}	1.56547×10^{-03}
1.42993×10^{-03}	1.42993×10^{-03}
3.08518×10^{-03}	3.08518×10^{-03}
9.79227×10^{-04}	9.79227×10^{-04}
2.02836×10^{-04}	2.02836×10^{-04}

Table 2. Fraction of delayed neutrons per family for the configuration 245 of the IEA-R1 Reactor - January 2011

Fraction of delayed neutrons per family	
Obtained by reactor's team	Obtained by our program
2.87880×10^{-04}	2.87880×10^{-04}
1.56618×10^{-03}	1.56618×10^{-03}
1.43050×10^{-03}	1.43050×10^{-03}
3.08670×10^{-03}	3.08670×10^{-03}
9.79382×10^{-04}	9.79382×10^{-04}
2.027451×10^{-04}	2.027451×10^{-04}

Table 3. Reactor multiplication factor for the different parameters of security in the configuration 243A of the IEA-R1 Reactor – May 2010

Criticality for calculating the reactivity coefficients			
Obtained by reactor's team	Obtained by our program	Obtained by reactor's team	Obtained by our program
0.9771521	0.9771521	0.9751310	0.9751310
0.9738870	0.9738870	0.9765801	0.9765801
0.9703091	0.9703091	0.9720200	0.9720200
0.9742540	0.9742540	0.9755995	0.9755995
0.9765500	0.9765500	0.9726790	0.9726790
0.9745845	0.9745845		

Table 4. Reactor multiplication factor for the different parameters of security in the configuration 245 of the IEA-R1 Reactor - January 2011

Criticality for calculating the reactivity coefficients			
Obtained by reactor's team	Obtained by our program	Obtained by reactor's team	Obtained by our program
0.9986469	0.9986469	0.9966908	0.9966908
0.9953282	0.9953282	0.9980443	0.9980443
0.9919758	0.9919758	0.9932460	0.9932460
0.9955956	0.9955956	0.9970679	0.9970679
0.9980344	0.9980344	0.9942942	0.9942942
0.9960696	0.9960696		

3. CONCLUSION

Given that the program's algorithm remained unchanged and being it a transformation into sub-sections of a main program, it was expected that up to the last decimal place values were coincide, which was saw in the graphs and tables, and therefore it proves to be very reliable, having already been started its implementation phase with the staff of the IEA-R1.

In the last months with the aim of making the best possible program for use by operators of the reactor were made a lot of changes. This change provides to enter any number of fuel cross sections and no fuel cross sections in programs and remove any limitation into changes in input data supplied by operators, allowing them to change this data any way they wish.

In the current stage of the project is that it is being evaluated by the quality department of the IEA-R1 through tests and reports in order to become the new computational code to be used for the calculations of configuration.

Currently, it takes about three hours to do all the calculations and assembly manuals inputs. Using the manager program this time was reduced to just five minutes of calculations, optimizing the time of the operators so they can give priority to other tasks.

ACKNOWLEDGMENTS

Thanks for the configuration information from IEA-R1 and input data of the programs provided by Dr. Valdemir Gutierrez Rodriguez. For the scientific initiation scholarship provided by FAPESP and later, after the expiry of this period, CNPq. For help with understanding of the codes by Mr. Mitsuo Yamaguchi.

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APPENDIX

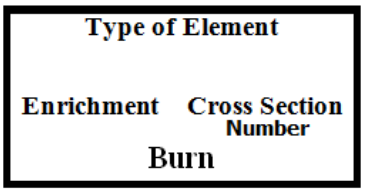

Figure 10. Layout of Fuel Elements of Figures 10 and 11

	3	4	5	6	7
4	EC-172 3,0 - 16/16 40,23	195 – PLR 2,3 – 13/22 39,94	EC-201 3,0 – 16/16 4,04	EC 177 3,0 – 16/16 27,52	196-PLR 2,3 – 13/22 32,92
5	197 - PLR 2,3 – 13/22 22,79	ECC 211 13 15,71	EC-178 3,0-16/16 27,81	ECC 198 13 42,35	200 - PLR 2,3 – 13/22 22,92
6	203 - MR 3,0 – 16/28 16,75	EC208-In 3,0 – 16/16 1,46	EIBE	205 - MR 3,0 – 16/28 8,07	EC 184 3,0 – 16/16 19,45
7	EC 209 3,0 – 16/16 1,95	ECC 199 13 43,05	202 - MR 3,0 – 16/28 19,50	ECC 212 13 15,54	194 - PLR 2,3 – 13/22 39,91
8	204 - MR 3,0 – 16/28 8,79	191 – PLR 2.3 – 13/22 40,55	EC 174 3,0 – 16/16 26,49	193-PLR 2,3 – 13/22 38,10	EC 176 3,0 – 16/16 26,97

Figure 11. Core configuration 243A – 3.5 MW – May -2010 (IEA-R1)

	3	4	5	6	7
4	EC-172 3,0 - 16/16 40,478/43,34	195 – PLR 2,3 – 13/22 41,566/44,36	EC-178 3,0 – 16/16 8.841/9,69	EC 177 3,0 – 16/16 29,734/32,07	196-PLR 2,3 – 13/22 34,624/37,11
5	197 - PLR 2,3 – 13/22 26,466/28,52	ECC 211 13 21,679/23,43	EC-201 3,0-16/16 8.894/9,75	ECC 220 16 4,474/4,92	200 - PLR 2,3 – 13/22 26,621,28,68
6	EC 208-In 3,0 – 16/16 7,831/8,59	203-MR 3,0 – 16/28 22,046//23,91	EIBE	EC 184 3,0 – 16/16 23,732/25,71	205 - MR 3,0 – 16/28 13,376/14,61
7	EC 209 3,0 – 16/16 7,428/8,15	ECC 221 16 4,729/5,20	202 - MR 3,0 – 16/28 24,095/26,10	ECC 212 13 21,339/23,07	EC 206 3,0 – 16/16 0,0
8	204 - MR 3,0 – 16/28 12,258/13,40	EC 207 3,0 – 16/16 0,0	EC 174 3,0 – 16/16 28,723/31,00	193-PLR 2,3 – 13/22 39,604/42,31	EC 176 3,0 – 16/16 28,172/30,42

Figure 12. Core configuration 245 – 4.0 MW – January -2011 (IEA-R1)

ECC - Control fuel element
EC - Fuel Element
EIBE - Beryllium Irradiation Cladding
MR - Fuel Element with side plates with half mass in U-235
PLR - Fuel Element with side plates with low enrichment

Diffusion Equation:

$$-D_g^r(X)''+\sum_{Rg}^r \phi_g^r(X)=\frac{1}{K_{eff}} X_g \sum_{g'=1}^{G} \nu \sum_{fg}^r \phi_{g'}^r(X)+\sum_{g'=1;g'\neq g}^{G} \sum_{gg'}^r \phi_{g'}^r(X) \quad (2)$$

Where $0 \leq x \leq L$, in which:

L is the limit of the field;

r is the region (r = 1: V), V is the total number of regions;

g is the energy group (g = 1: G), where G is the total number of energy groups;

ϕ_g^r is the scalar flux of neutrons for the energy group g in region r;

D_g^r(X) is the diffusion coefficient for neutron energy group g in region r;

$\sum_{Rg}^r (X)$ is the removal cross section of the macroscopic energy group g in region r;

$\sum_{gg'}^r (X)$ is the scattering cross section of the macroscopic group g to group g' energy in the region r;

$\sum_{fg'}^r (X)$ is the fission cross section of the macroscopic energy group g in region r;

K_{eff} is the effective multiplication factor;

ν shall be the average number of neutrons released by fission;

X_g is the integrated spectrum of the fission energy group g.