

Characterization of Burned Fuel of the TRIGA IPR – R1 Research Reactor Using Monteburns Code

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ABSTRACT

Monteburns is a completely automated computational tool that links the Monte Carlo code MCNP with the burnup and decay code Origen2.1. This code system was used to simulate the core of the CDTN's TRIGA research reactor and nowadays is the current methodology used for the theoretical analysis of the main neutronic parameters of such reactor. Some results of the characterization of the burned fuel elements inside the core will be presented, including the mass of transuranic elements and of some fission products, the activity, heatload and radiotoxicities of the fuel. Furthermore, the core excess reactivity and control rods reactivity worth of the reactor after the power upgrading to 250 kW as well as the axial and spectral distribution of the neutron flux in the main irradiation devices were calculated.

1. INTRODUCTION

The utilization of Monte Carlo transport methods to simulate nuclear reactors is becoming a practical option due to the increasing processing power of the computers. Burnup calculations of nuclear fuel, in spite of still being very slow, can already be carried out. That was the main reason for CDTN (Center for Nuclear Technology Development), Belo Horizonte, Brazil, to abandon the previous neutronic calculation methodology of its research reactor (TRIGA IPR – R1), based on WIMS [1] and CITATION [2] codes and replace it with the Monte Carlo transport based calculations.

2. MODELING

The simulation considered evolution of the IPR – R1 core since the beginning of life until present. The assumed burnup and decay history was as follows:

- From 1960/11/06 to 1964/06/03: 56 fuel elements; burning 149 MWh at 30 kW reactor power, followed by 1152 days decaying.
- From 1964/06/03 to 1973/07/23: 57 fuel elements; burning 231 MWh at 100 kW reactor power, followed by 3181 days decaying.
- From 1973 to 1996: 58 fuel elements; burning 1224 MWh at 100 kW reactor power, followed by 7885 days decaying.
- From 1996 to 2002/May: 59 fuel elements; burning 196 MWh at 100 kW reactor power, followed by 2000 days decaying.
- 2002/May: 63 fuel elements (4 fresh fuel elements go in to core). 250 kW.

The assumption of constant and continuous burnup followed by a decay time after (also continuously) in each cycle is just an approximation. However, simulating power history step by step would be impractical because of the numerous startups, power level changes and shutdowns of the reactor. Such assumption as well as the uncertainty on the reactor thermal power calibration are the strongest sources of errors of the burned fuel characterization.

The effects of the nuclides over k_{eff} calculations for TRIGA reactors were verified [10] and they have shown that for small burnup on TRIGA fuel (less than 5%) it is enough to consider only the influence of Xe135, Sm149, Sm151, Pu239, Nd143, U236, Pm147, Rh103, Xe131, Cs133, Tc99, Nd145 and Pu240. Considering then that the average burnup of the fuel elements in IPR – R1 have less than 5% U235 burned only these nuclides were considered in the current simulation.

The excess reactivity, ρ , is calculated as follows

$$\rho = \frac{k_{\text{eff}} - 1}{k_{\text{eff}}}$$

and the control rods reactivity worth as

$$\Delta\rho = \frac{1}{(k_{\text{eff}1})} - \frac{1}{(k_{\text{eff}2})}$$

where, $k_{\text{eff}1}$ and $k_{\text{eff}2}$ are, effective neutron multiplication factor of the cores with the control rod fully inserted and fully withdrawn.

The shutdown margin is given by the difference between the reactivity worth of the considered control rods (the most worthy rod is assumed fully withdrawn) and the excess reactivity. The calculation model considers the shadowing effect of the Safety and Regulating rods.

The error of the MCNP geometry model is estimated to be within 500 pcm and is mostly due to uncertainties on material composition specifications and simplifications of geometry [12]. The errors attributed to the power history and to the reactor thermal power calibration were not evaluated.

In MCNP part of the calculation the ENDF/B-VI and ENDF/B-V cross section data was used, based on the availability. No correction of temperature was used to the cross sections data. The thermal burnup and decay data were used in the Origen part of the calculation. The *criticality* calculations were done using the KCODE option of MCNP with 5.000 histories per cycle and 950 active cycles, skipping the first 50 cycles. The temperature of the water in the reactor tank was assumed to be 23°C. This lead in the errors of about 40 pcm (0.04 %) in k_{eff} estimates. The *neutron flux* calculations were also performed with the KCODE option of MCNP with 50.000 histories per cycle and 900 active cycles, skipping the first 100 cycles.

The neutron flux distribution for IPR – R1 was simulated in the central thimble and rotary groove air. In the central thimble the neutron flux was calculated in 51 positions located at radius zero and axially varying from $z = -25$ cm to $z = +25$ cm, being $z = 0$ the center of the active core. Regarding the rotary groove air were 26 positions, axially, from $z = 1$ cm to $z = 26$ cm and radially located in the center of the groove. Figure 2 shows an axial view of the MCNP geometric model in which one can see such simulated positions as the series of small black traces in the central thimble and in the rotary groove air. The neutron flux spectra were calculated in the central thimble at the coordinates (0, 0, 0) and at the center of the rotary groove in $z = 1$ cm.

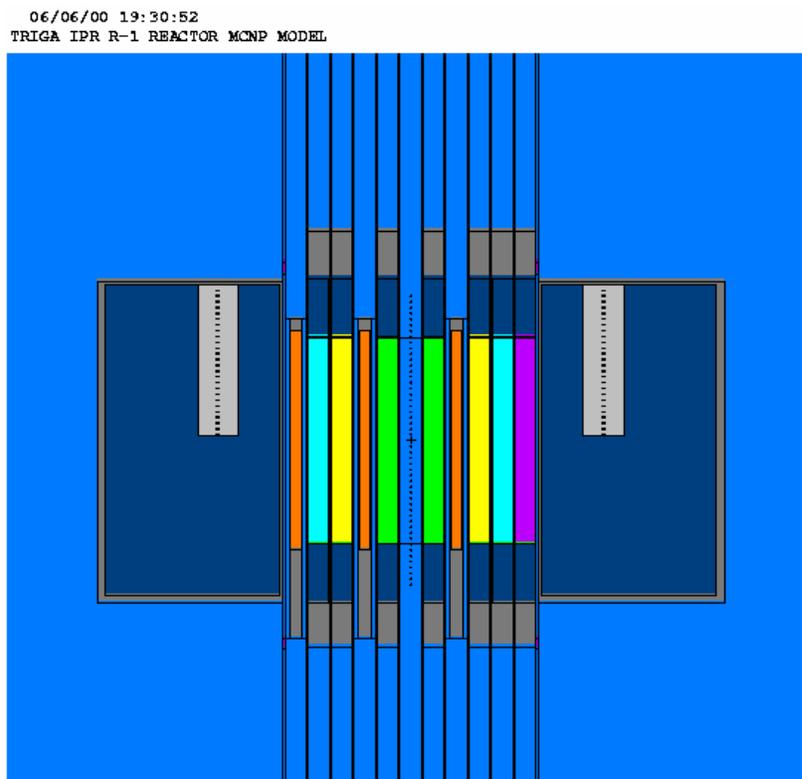


Figure 2. Axial view of the IPR – R1 MCNP model.

3. RESULTS

3.1 Criticality

Table 1 shows the calculated and measured [11] values of excess reactivity, control rods worth (IPR – R1 has 3 rods: Control, Safety and Regulating) and shutdown margin for IPR – R1 at 250 kW core configuration. It can be noticed that the differences between calculations and measurements are below the 500 pcm. The standard deviation for k_{eff} in the Monte Carlo simulations was around 40 pcm. Such value is not negligible to the Regulating rod and this can explain the high relative difference found in this case.

Table 1 – Measured and Calculated Criticality Results.

	Calculated ρ (pcm)	Measured ρ (pcm)	Difference (pcm)
Excess reactivity	1558	1822	264
Control rod reactivity worth	2335	2412	77
Safety rod reactivity worth	2221	2212	-9
Regulating rod reactivity worth	274	408	134
Shutdown margin – Control rod out	839	798	41

3.2 Axial And Spectral Distribution Of Neutron Flux In The Central Thimble And Rotary Groove Air.

Neutron flux distributions are plotted in figures 3 and 4. The thermal neutrons are those with energy below 0.4 eV, which is the cutoff energy for cadmium, material mostly used to cover gold foil detectors. Flux measurements under similar conditions to the simulations are being planned in order to check the results.

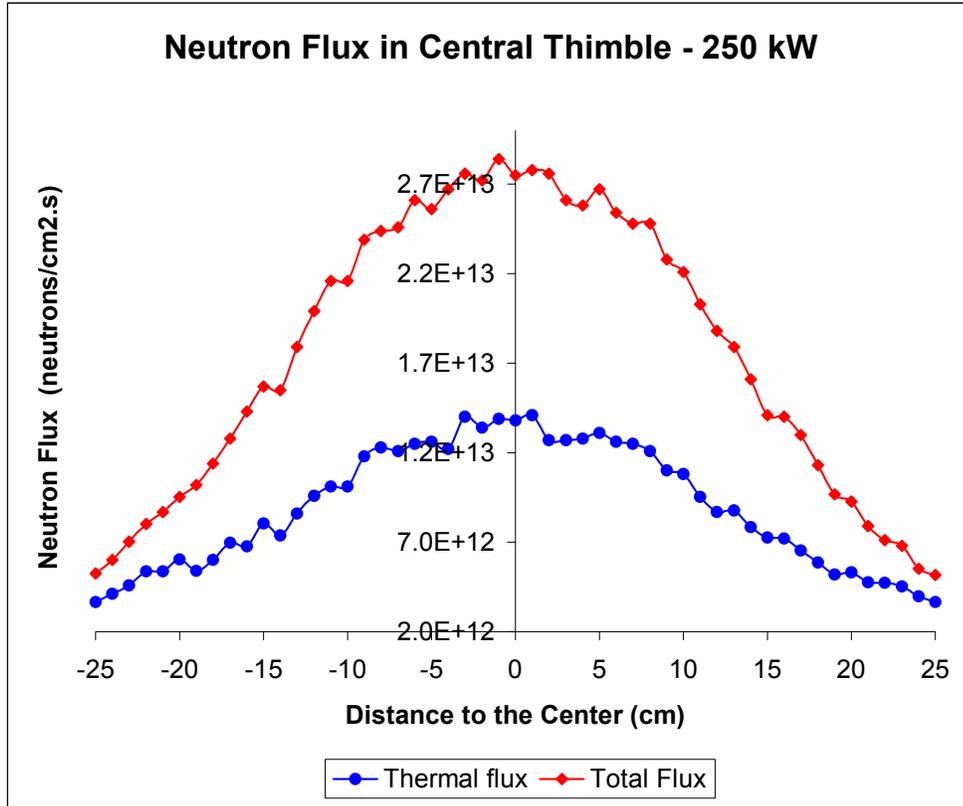


Figure 3. Axial Distribution of Neutron Flux in Central Thimble.

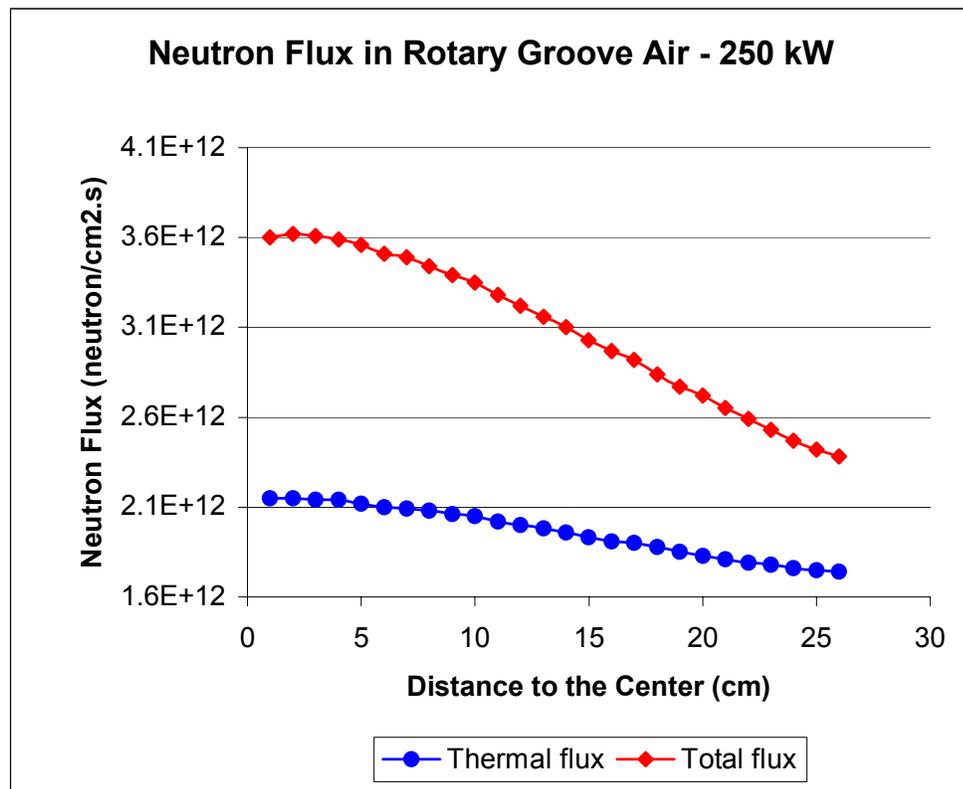


Figure 4. Axial Distribution of Neutron Flux in Rotary Groove Air.

The results of neutron spectrum are presented in table 2 and figures 5 and 6. One can notice the high proportion of neutrons in the energy range between 0.01 eV and 0.1 eV in which can be found the neutrons thermalized at water temperature (around 23 °C, which means around 0.026 eV). The neutron flux below 0.4 eV represents 50% and 60% of the total in central thimble and rotary groove air, respectively.

Table 2 – Spectral Distribution of Neutron Flux in Central Thimble and Rotary Groove Air

Energy range (eV)	Neutron Flux in Central Thimble (neutrons/cm².s)	Neutron Flux in Rotary Groove Air (neutrons/cm².s)
0 to 0.01	7.24E+11	1.02E+11
0.01 to 0.1	1.06E+13	1.68E+12
0.1 to 0.4	2.45E+12	3.69E+11
0.4 to 1	6.21E+11	7.22E+10
1 to 10	1.29E+12	1.60E+11
10 to 10²	1.16E+12	1.55E+11
10² to 10³	1.33E+12	1.67E+11
10³ to 10⁴	1.32E+12	1.82E+11
10⁴ to 10⁵	1.62E+12	2.20E+11
10⁵ to 10⁶	3.03E+12	2.98E+11
10⁶ to 14E10⁶	3.28E+12	1.98E+11
14E10⁶ to 20E10⁶	0.00E+00	3.63E+07
Total	2.75E+13	3.60E+12

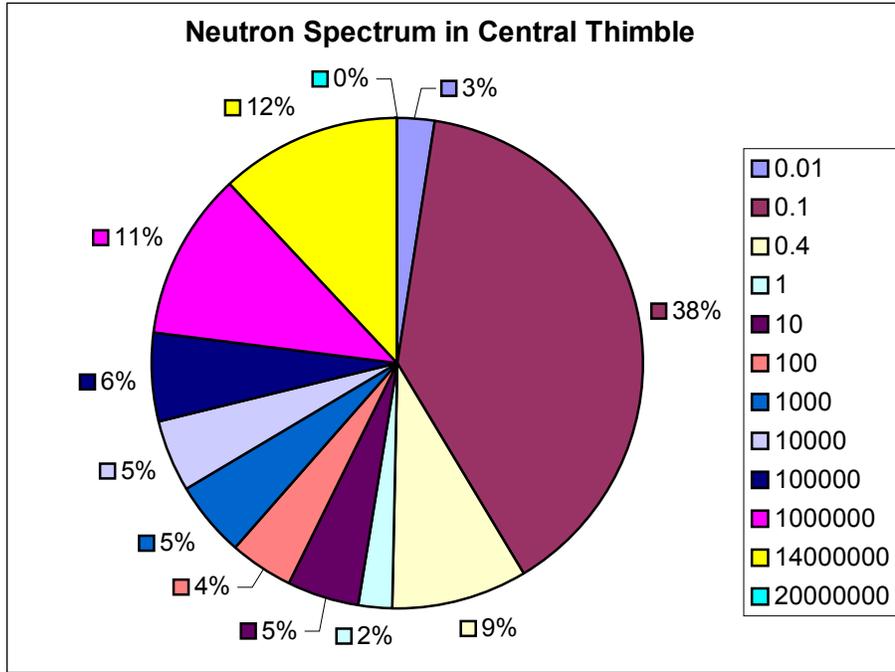


Figure 5. Percent

Distribution of Neutron Flux Spectrum in Central Thimble – 250 kW.

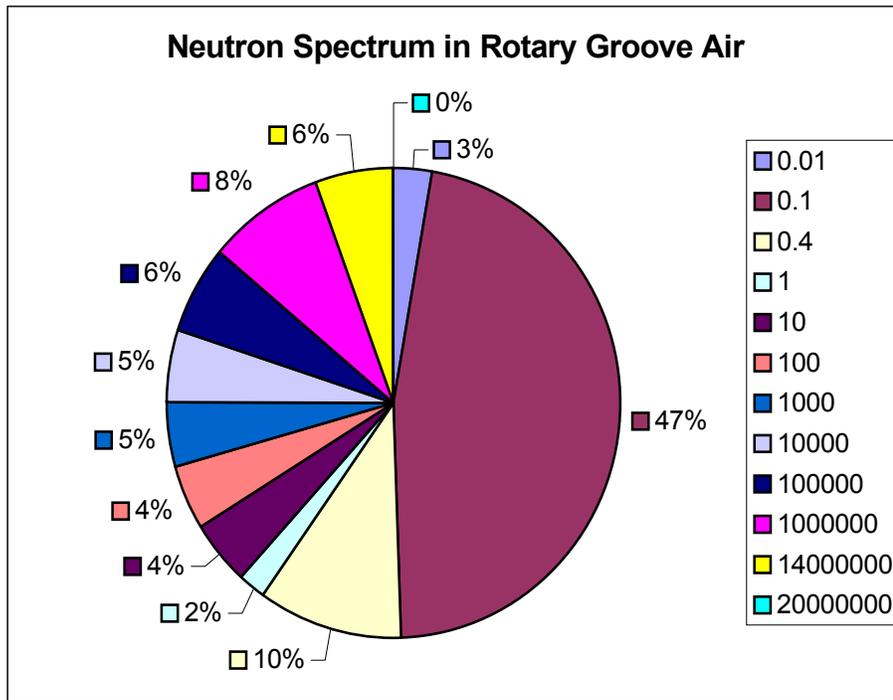


Figure 6. Percent Distribution of Neutron Flux Spectrum in Rotary Groove Air – 250 kW

3.3 Fuel Characterization

Monteburns system was utilized to characterize the burned fuel of the IPR – R1. Selected chemical elements, yields of nuclear reactions in the fuel, were analyzed. These nuclides had their masses, heatload, activity, and radiotoxicities calculated. The values presented in table 3 refers to the total of fuel inside the reactor core (63 fuel elements) which followed the irradiation history as described on section 2 of this paper.

Table 3. Mass, Activity, Heatload and Radiotoxicities of some nuclides in the Fuel.

Nuclide		Mass (grams)	Activity (Ci)	Heatload (W)	Inhalation radiotoxicity (m3 of air)	Ingestion radiotoxicity (m3 of water)
Name	MCNP ID number					
Hydrogen	1001.60c	146	0	0	0	0
Zirconium-93	40093.50c	3,6	8,9e-3	1,0e-6	2,2e6	11,1
Technetium-99	43099.60c	1.82	3,1e-2	1,6e-5	4,4e5	103
Rhodium-103	45103.50c	0,96	0	0	0	0
Iodine-127	53127.60c	5.1e-02	0	0	0	0
Iodine-129	53129.60c	0,25	4.5e-5	2,1e-8	2,2e6	745
Xenon-131	54131.50c	1,1	ND	ND	ND	ND
Xenon-135	54135.50c	0	0	0	0	0
Cesium-133	55133.60c	2,7	0	0	0	0
Cesium-135	55135.60c	2,3	2,7e-3	8,9e-7	ND	ND
Cesium-137	55137.60c	1,9	1,7e2	0,18	8,2e10	8,2e6
Neodymium-143	60143.50c	2,5	0	0	0	0
Neodymium-145	60145.50c	1,7	0	0	0	0
Promethium-147	61147.50c	4,9e-2	4,5e1	1,6e-2	ND	ND
Samarium-149	62149.50c	9,6e-2	2,3e-14	ND	ND	ND
Samarium-151	62151.50c	0,13	3.4	3,9e-4	ND	ND
Uranium-235	92235.60c	2170	4,7e-3	1,2e-4	2,4e8	156
Uranium-236	92236.60c	13,1	8,5e-4	2,3e-5	4,2e7	28,2
Uranium-238	92238.60c	9010	3,0e-3	7,7e-5	1,0e9	75,6
Neptunium-237	93237.60c	3,0e-2	2,1e-5	6,6e-7	2,1e8	7,14
Neptunium-239	93239.60c	8,6e-14	2,0e-8	4,8e-11	ND	ND
Plutonium-239	94239.60c	6,7	0,4	1,3e-2	6,9e12	8,3e4
Plutonium-240	94240.60c	0,1	2,3e-2	7,2e-4	3,9e11	4,6e3
Plutonium-241	94241.60c	1,9e-3	0,19	5,9e-6	6,4e10	967
Plutonium-242	94242.60c	2,1e-5	8,2e-8	2,4e-9	1,4e6	1,6e-2
Plutonium-244	94244.60c	4,4e-13	7,7e-18	2,2e-19	ND	ND
Americium-241	95241.60c	1,2e-3	4,1e-3	1,4e-4	2,0e10	1,0e3
Americium-243	95243.60c	1,0e-7	2,0e-8	6,4e-10	1,0e5	5,0e-3
Curium-242	96242.60c	5,3e-10	1,8e-6	6,5e-8	4,4e5	8,8e-2
Curium-243	96243.60c	2,4e-10	1,2e-8	4,6e-10	6,2e4	2,5e-3
Curium-244	96244.60c	6,7e-10	5,4e-8	1,9e-9	1,8e5	7,8e-3
Curium-245	96245.60c	9,8e-13	1,7e-13	5,6e-15	ND	ND
Curium-246	96246.60c	4,3e-15	1,3e-15	4,3e-17	ND	ND
Curium-247	96247.60c	1,3e-15	1,2e-19	3,9e-21	ND	ND
Actinides	--	11200	1310	1,3	4,6e13	5,1e7

ND

– Data not available in the Origen2.1 and/or MCNP4B libraries.

It was already mentioned that the assumption of constant and continuous burnup followed by a decay time after for each cycle is just an approximation. This assumption as well as the uncertainty on reactor thermal power calibration are the strongest sources of errors of the burned fuel characterization. The magnitude of these errors remains to be estimated.

The simulation has shown that the mass of U235 reaches 2170 grams and just 6.8 grams for all plutonium isotopes, reflecting the very low burnup of the IPR –R1 fuel. Regarding the activity of fission products cesium137 and promethium147 show the highest values. The strong activity of the sum

of all actinides can not be explained only by the activity of the actinides listed in table 3, but also other actinides like californium, berkelium, etc, have important contribution, not only in the activity, but also over the heatload and radiotoxicities.

4. CONCLUSIONS

Monteburns system is currently being used for research reactor neutronic analyses. The code system has also been used in simulations of the IPR – R1 TRIGA reactor at CDTN, Belo Horizonte, Brazil. Some experimental data are available for checking the calculation results, but further experiments are necessary in order to validate other calculated parameters.

Criticality calculations are well within the expected accuracy of the calculation methodology and MCNP model. The neutron flux spectrum simulation show the high proportion of neutrons in the energy range between 0.01 eV and 0.1 eV in which can be found the neutrons thermalized at water temperature (around 23 Celsius, which means around 0.026 eV). The neutron flux below 0.4 eV represents 50% and 60% of the total in central thimble and rotary groove air, respectively.

Some chemical elements, yields of nuclear reactions in the fuel, have been analyzed and such nuclides had their masses, heatload, activity, and radiotoxicities theoretically evaluated. The activities currently being done on the characterization of the burned fuel of IPR – R1 research reactor are part of the efforts to reach an appropriated strategy (transport and interim storage) to deal with spent fuel.

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