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NEW BURNUP CALCULATION OF TRIGA IPR-R1 REACTOR

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ABSTRACT

The IPR-R1 TRIGA Mark I research reactor, located at the Nuclear Technology Development Center - CDTN, Belo Horizonte, Brazil, operates since 1960.The reactor is operating for more than fifty years and has a long history of operation. Determining the current composition of the fuel is very important to calculate various parameters. The reactor burnup calculation has been performed before, however, new techniques, methods, software and increase of the processing capacity of the new computers motivates new investigations to be performed. This work presents the evolution of effective multiplication constant and the results of burnup. This new model has a more detailed geometry with the introduction of the new devices, like the control rods and the samarium discs. This increase of materials in the simulation in burnup calculation was very important for results. For these series of simulations a more recently cross section library, ENDF/B-VII, was used. To perform the calculations two Monte Carlo particle transport code were used: Serpent and MCNPX. The results obtained from two codes are presented and compared with previous studies in the literature.

1. INTRODUCTION

Research reactors are used in a series of activities related to research and training, radioisotope production, analysis by neutron activation, reactor physics and thermal-hydraulics experiments. Furthermore, research reactors can provide valuable knowledge for testing and validation of nuclear codes. These reactors consist of a great deal of complexity into components with different dimensions and materials. The moderator material, the coolant, fuel elements, gaps, irradiation channels, control rods, graphite reflector, play an important role in modeling influence the sensitivity of the calculation. The reliability of the codes will depend on how the results are in agreement with the experiments and with other codes already well-established in the literature. Comparison between Serpent and MCNPX shows a good agreement, which is expected because the two codes can share the same cross section data and the same thermal scattering cross sections [1-2].

The most common test cases for testing and validation of nuclear codes are taken from the International Handbook of Evaluated Criticality Safety Benchmark Experiments [3]. In case of effective multiplication factors the values are generally close to unity, although there are some significant exceptions as well, such as the Sheba-II experiment, for which the criticality

constant is over-estimated by 3000 pcm [4-6]. For more complex structures such as research reactors the comparison between the codes have not yet been clearly shown, more research is needed. Another problem is the use of a number of simplifications to the model adopted. For example, most of the fuel element simulations consider a cylinder, assuming an equal mass of fuel distributed on all fuel elements.

In this study, the focus is modeling the IPR-R1 TRIGA Mark I (Fig. 1) using different radiation transport codes: MCNPX and Serpent Monte Carlo reactor physics calculation code. The IPR-R1 belongs to a class of small nuclear reactors designed and manufactured by General Atomics (USA), called TRIGA (Training, Research, Isotopes, General Atomics). This class of reactors, as indicated by the acronym, was developed mainly for staff training and research. All works performed using the reactor is dependent on the neutron flux data, as well as the temperature of the neutrons and other parameters. These data are extremely important, especially when you are applying the k0 standard method of neutron activation [8-9], routine method in CDTN in Neutron Activation Laboratory (LAN / CDTN). Hence the validation of the parameters is vitally important to confer reliability to the works which uses the reactor, also contributing as a security measure, indicating possible problems with the reactor.

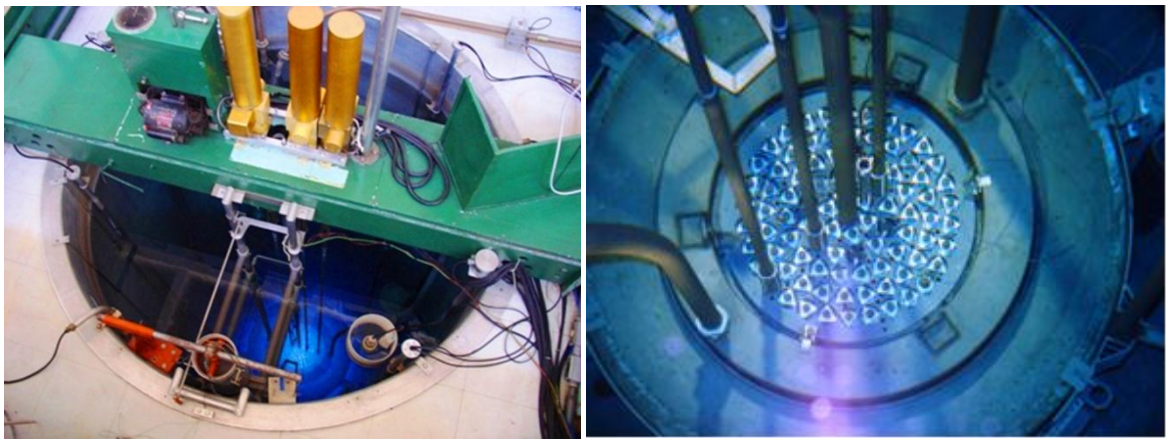

Figure 1: IPR-R1 TRIGA MARK I [7].

With the advance of the operating time of the reactor undergoes the power fluctuations as a result of burnup of the fuel since the beginning of operation. Despite the emergence of fertile elements also occur a reduction of fissile material and the appearance of fission products responsible for poisoning the reactor. It is necessary then to add new fuel elements to compensate for these changes. Besides burns, the change-over in core configuration changes the distribution of neutron flux and, consequently, the reactor power. Since the beginning of its operation, CDTN's TRIGA reactor underwent six changes in the core configuration. Since the beginning of its operation, CDTN TRIGA reactor underwent six changes in the core configuration. its Initially maximum thermal power was 30 kW. Later were added more fuel in core to increases the power. Now the reactor is in process to license to operate at 100 kW, but in the past these operate in power higher than 250 kW.

With this change has become essential reassign various parameters of the reactor, especially the neutron flow and thermal power. Previous work focused to determine such parameters

after the last change in the core configuration [9-13]. The neutron fluxes as well as the ratio of the total flow of neutrons and epithermal and thermal flows in order to ensure accuracy in the application of neutron activation analysis. Experiments were conducted to update the parameters α (measures the departure from the Boltzmann curve) and f (is the ratio of the flux of thermal neutrons and epithermal) required for application of the parametric method k0 [8]. In order to determine the thermal power and the temperature distribution in the reactor analytical and experimental studies were performed [10-12]. A new methodology for calibration and monitoring of the power dissipated by the core was developed, thus deploying new power measurement channels thermal process. The thermal conductivity of the fuel element and the heat transfer coefficient of the refrigerant were evaluated experimentally. A formulation has been proposed for determining the value of the conductance at the interface between fuel and cladding (gap).

Besides the experimental work flow values for the sixth embodiment of the IPR-R1 core they were obtained through the use of three computational codes: MCNP4B (transport) ORIGEN2.1 (burns and radioactive decay) and MONTEBURNS (engagement transport / burn), the theoretical values of thermal and the total flux calculated for some positions axially distributed on the turntable and the central tube [13]. To characterize the spent fuel from the IPR - R1 were selected some nuclides of most interest and calculated their mass, decay heat, activity and radiotoxicities. This data calculated referring to the total fuel in 63 fuel elements present in the nucleus in November 2004. Based on these results was observed a small burn the fuel elements in the IPR past 44 years of reactor operation. Using data obtained in this work is possible see a reduction of 96 grams in mass U235 compared to the initial inventory of this nuclide with 63 elements. This value corresponding to an average burnup of U235, equal to 4.1%.

Another simulation work [14] used these data and included an aluminum structure in rotary specimen rack (or "lazy Susan") to investigate and determine the neutron flux in each sample positions of the rack. This work the geometry is refined to recalculate the core composition with the Monte Carlo codes.

Previous works have been considered simplified structures and presents uncertainties about the current composition of the fuel elements, due to the very long history of operations as the absence of information. Thereby new investigations become necessary. In this work we present the results of burnup using a more detailed geometry with the introduction of new devices, like the control rods and the samarium poison discs. This increase of materials in the simulation burnup calculation was very important for results. To perform the calculations two Monte Carlo particle transport code were used: Serpent and MCNPX. The results obtained from two codes are presented and compared.

2. NUCLEAR CODES

2.1 MONTEBURNS

Several factors contributed to the software coupling. Development of simulation techniques in parallel and the use of "clusters" allowed reduce the simulation time and increase capacity of calculations. Factors that benefited the calculations burnup. MONTEBURNS is a fully automated software tool that engages the carriage Monte Carlo code, MCNP4B, with the

burn code and radioactive decay, ORIGEN2.1. This code is used in previous simulations and the data is compared whit new versions of MCNP and Serpent Monte Carlo code. [15]

2.2 MCNPX

MCNPX is an extended version of MCNP code with additional functions. MCNP is used to calculate only neutron particles. On the other hand, MCNPX can concern all particles and all energies. Particularly, MCNPX also included improvements in transmutation module and library tools through CINDER90 code. CINDER90 is a FORTRAN program or "code" with a data library used to calculate the inventory of nuclides in an irradiated material. This contains library file decay, fission yield, and 63-group cross section data not calculated by MCNPX. The library of nuclear data, constantly growing in breadth and quality with international cooperation, describes 3400 nuclides, 1325 fission products, and yield sets for over 30 actinides. [16]

2.3 Serpent

Serpent is a three-dimensional continuous-energy Monte Carlo reactor physics burnup calculation code, developed at VTT Technical Research Centre of Finland since 2004. The publicly available Serpent 1 has been distributed by the OECD/NEA Data Bank and RSICC since 2009, and next version of the code, Serpent 2, is currently in a beta-testing phase and available to registered users. [1]

The burnup capability in Serpent is entirely based on built-in calculation routines, without any external coupling. The number of depletion zones is not restricted, although memory usage becomes a limiting factor for Serpent when the number of burnable materials is large.
The code treats depleted materials in fuel pins different from materials in ordinary cells. Each pin type is treated separately and further divided into annular depletion zones of equal volume. This division is important for calculated accounting for the effects caused by spatial self-shielding.

The irradiation history is defined in units of time or energy. Reaction rates are normalized to total power, specific power density, flux or fission rate and the normalization can be changed by dividing the irradiation cycle into a number of separate depletion intervals. Volumes and masses needed for the normalization are automatically for simple geometries, such as 2D fuel pin lattices. The values can also be obtained from a Monte Carlo based volume calculation routine or entered manually. Serpent 2 offers various higher-order methods and sub-step solutions for burnup calculation [17-19].

3. THE MODEL SIMULATED

3.1. Geometry and Composition of Core

The core starts with 53 cylindrical fuel elements and ends with 63 fuel elements in final configuration. In the final configuration there is an additional complication that is the

addition of 4 stainless steel-clad fuel elements in the core that are very different to the other 59 aluminum fuel elements.

In the proposed model the control rods were inserted in order to reproduce the same behavior observed in experiments (Fig. 2). According to the General Atomics Manual [20] the control rods have a cylindrical shape and are sealed in an aluminum container, the lower end have a conical shape to reduce the resistance imposed by water during a shutdown. For a preliminary test the conical tip was not reproduced. The up and down position of the control rods were defined as: Up position coincide with a plane located slightly above the active region of the fuel; Down position is considered when the middle point of fuel and the control rods is in the same height.

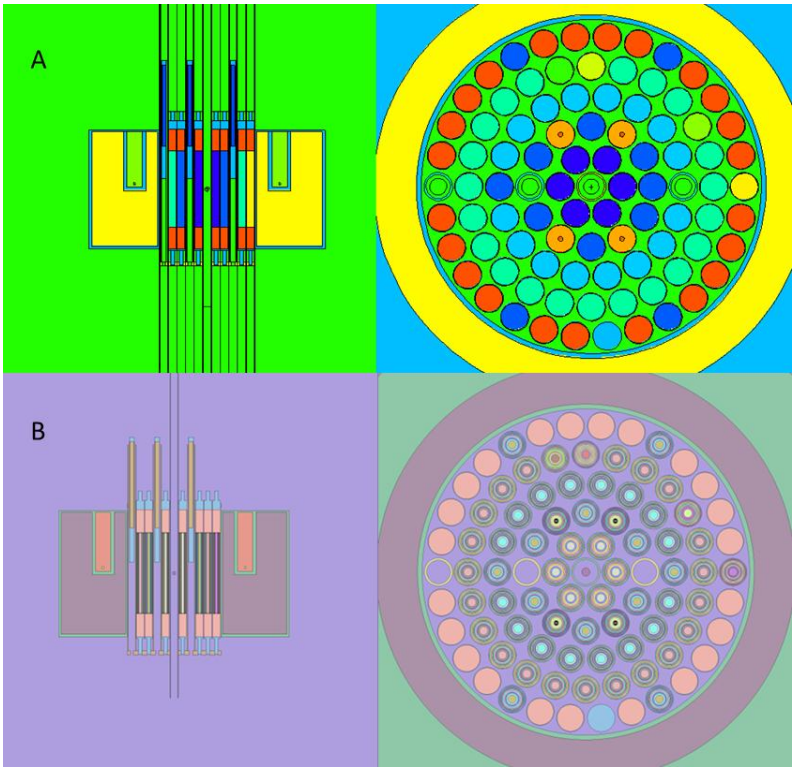

Figure 2: Geometry of core with 63 fuel elements in MCNPX(A) and Serpent(B)

Data for BOL (Begin Of Life, 1960), positions and composition of each fuel element are well known. The model does not consider the individual composition of each fuel element and consider only an average per ring. The estimate composition for the disc of samarium present in fuel elements was also maintained on the basis of assumptions made in previous simulations [13].

3.2. Data Libraries

In the tests, ENDF/B-VII library was used. Some nuclides do not have the cross section defined by the ENDF/B-VII, in this case Serpent fills the gaps with data from libraries: BROND 2.2, JENDL 3.2, JEF 2.2, JEFF 3.1.1, JENDL 3.3, or ENDF/B VI.8 [1,16]. In MCNPX the gaps are filled manually in the input file.

In the case that neutrons transported have sufficiently low energies (typically less than 4eV) to the point that the effects of molecular bonds become important (molecules formed by hydrogen bonds like water, and some crystalline solids, among other materials) a specific thermal scattering cross sections - S(α,β) must to be used. These libraries are used to replace the low-energy free-gas elastic scattering reactions for some important bound moderator nuclides, such as hydrogen in water or carbon in graphite. Thermal systems cannot be modeled using free-atom cross sections without introducing significant errors.

In Serpent 2.1.21, data for zirconium hydride are not available. In the simulation data was imported from the library file ENDF 70/sab available to MCNPX and used for both codes. Thermal scattering data for light water, hydrogen in zirconium hydride, zirconium on the zirconium hydride and graphite was used.

Radioactive decay and fission yield data used in the calculation is read from standard ENDF format data libraries. The decay libraries contain data for almost 4000 nuclides and meta-stable states. The total number of different nuclides produced from fission, transmutation and decay reactions is generally lower, in the order of 1500. The concentrations of all included nuclides with decay data are tracked in the burnup calculation. Energy-dependent fission yields are available for all main actinides (31 nuclides in ENDF/B-VII data).

3.3. History of operation until 2004

The burnup calculation performed is based on a simplified burn history proposed by Dalle [9]. This historic data considers only two steps, one continuous burn period and one decay period, between the changes in the configuration of core. The simulation considered the evolution of the IPR – R1 core since the beginning of life, 1960, until 2004. The assumed burnup and decay history was as follows:

- From 1960 to 1964: 56 fuel elements; burning 149 MWh at 30 kW reactor power, followed by 1090 days decaying.
- From 1964 to 1973: 57 fuel elements; burning 231 MWh at 100 kW reactor power, followed by 2120 days decaying.
- From 1973 to 1996: 58 fuel elements; burning 1224 MWh at 100 kW reactor power, followed by 7891 days decaying.
- From 1996 to 2002: 59 fuel elements; burning 196 MWh at 100 kW reactor power, followed by 2048 days decaying.
- From 2002 to 2004: 63 fuel elements (4 fresh stainless steel fuel elements go in to core) burning 91 MWh at 250 kW, followed by 898 days decaying.

4. RESULTS

Simulations of the burnup IPR-R1 TRIGA reactor were performed with Serpent 2.1.21 and MCNPX. The simulations were performed with 10,000 neutrons and were sampled 75 steps after initial 25 steps in each simulation. These tests were run on Linux operating system.

Both software results offer a large list of nuclides in output. The output also gives information of total mass, total activities and total decay heat for the entire core. To simplify the analysis, only some nuclides of interest results are presented. The mass in grams for these nuclides are shown in Fig. 3. The values have good agreement between themselves, showing minor variations.

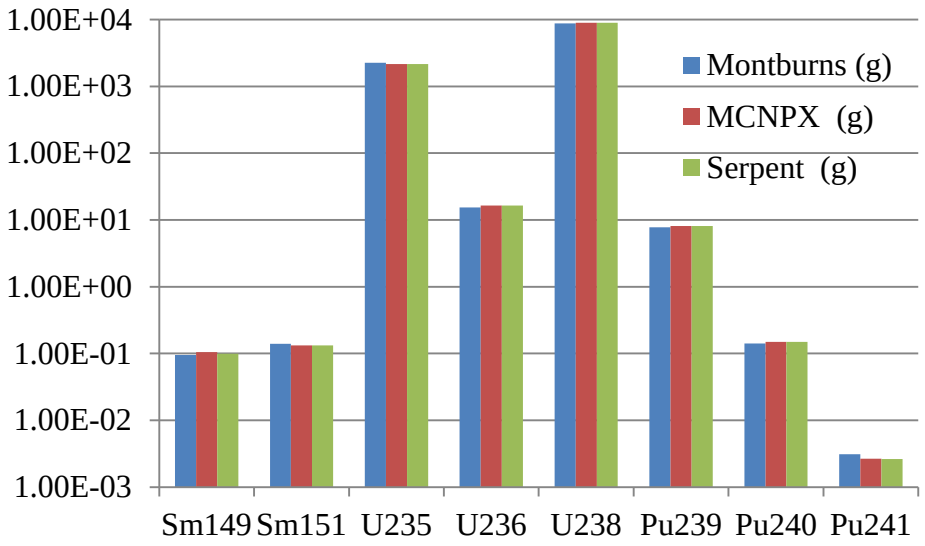

Figure 3: Mass in grams for the final configuration of IPR-R1

The variations become more evident when the values estimated by Monteburns are used as reference (Fig. 4). Compared with the previous calculation the mass of each nuclide has a small variation, where the average variation is less than 7%. The mean of variation observed in MCNPX is 6.88%, and 6.15% for Serpent. All the nuclides follow the same trend.

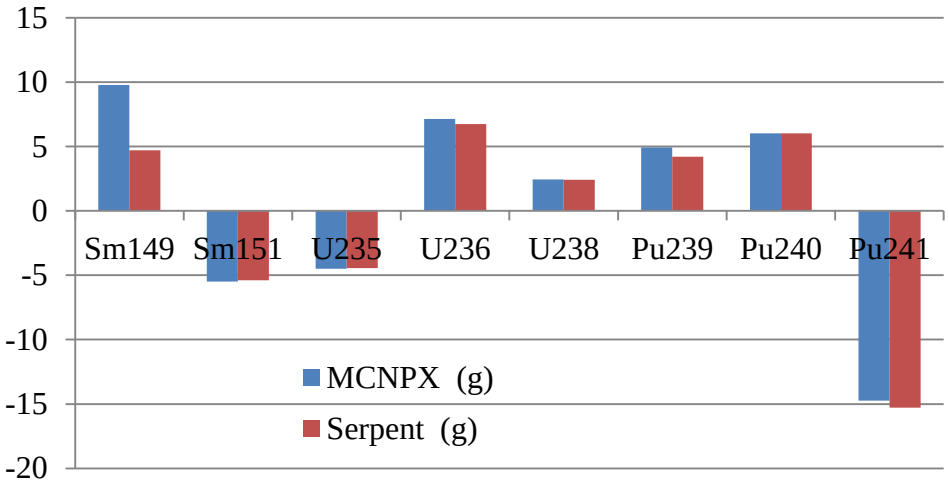

Figure 4: Relative composition of core based on Monteburns

The final mass of uranium-235 present in the nucleus is smaller in the new model, highlighting a 0,81% higher burnup in the Serpent than expected by MCNPX. The quantity of U235, U236 and U238 combined calculated in Serpent is less than 1% lower whit calculated in MCNPX.

The three different codes exhibit a wide variation only for results of Sm149. The MCNPX presents a result of lower value over Monteburns. The total mass of uranium is also lower in Serpent than MCNP, especially for Pu249 and Pu241.

Other neutronic parameter important to confirm the quality of the models and calculation methodologies is the value of effective multiplication constant. The evolution of Keff from BOL (beginning of life) until 2004 core is very important to understand the behavior of different software.

Figure 5 shows the observed Keff. Serpent and MCNPX predict similar values for all steps with the exception to the final configuration were stainless steel fuels were inserted in the core. This fuel has a very particular geometry and composition, and reasons for this difference must be further studied.

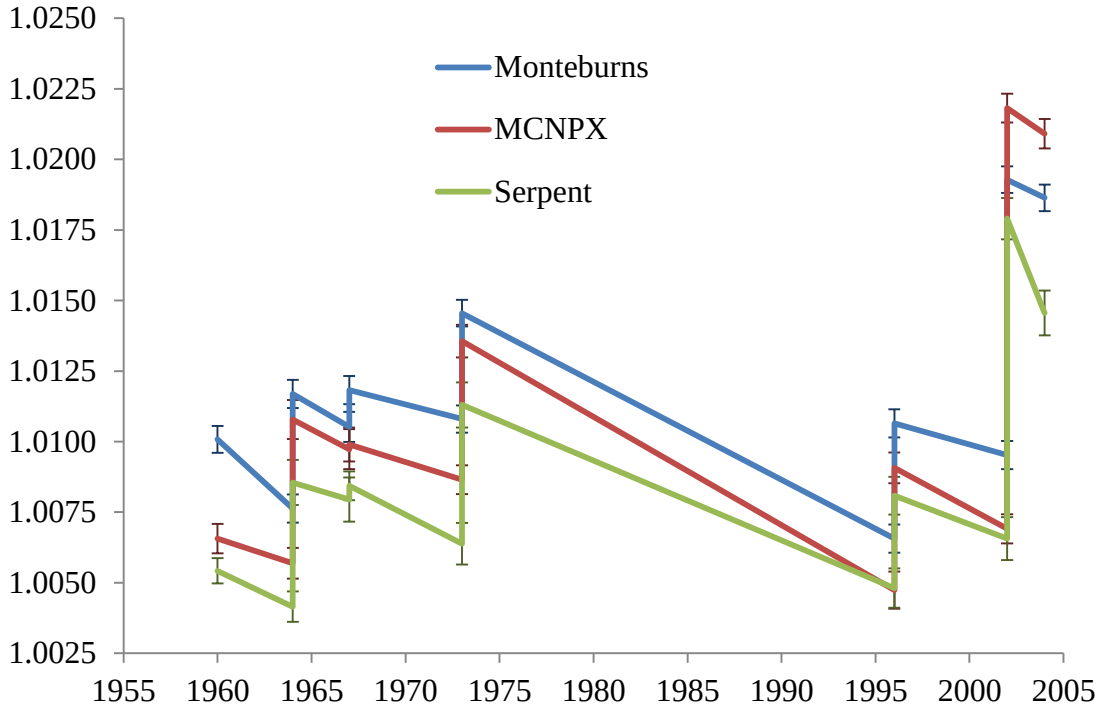

Figure 5: Evolution of K_{eff}

5. FINAL REMARKS – CONCLUSIONS

The current model proposed in this paper provides a more detailed geometry and considers nuclide changes in the fuel and also the evolution of other important materials in the core. Because of these factors it was expected a great difference of composition from that found

with previous simplified Monteburns calculation, but for the vast majority of nuclides the variations are small.

It is suggested for future work that a more detailed burn history be used because burnup with large time steps that encounter large flux-shape changes during the time step will lead to inaccurate calculations. Time steps small enough to capture the flux-shape change accurately over time are necessary. Another fact that needs to be discussed is the actual reactor operating power. There is an operating power uncertainty documented and questioned for the IPR-R1 [21-23], this data about the power is very important to calculate the burnup.

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REFERENCES

1. LEPPÄNEN, J. Serpent – a Continuous-energy Monte Carlo Reactor Physics Burnup Calculation Code. User's Manual, 2013.
2. LEPPÄNEN, J. and VIITANEN, T. Cross-section libraries for Serpent 1.1.7. Documentation, October 22, 2013.
3. OSTERLUND, M. et al., "Selecting benchmarks for reactor calculations", PHYSOR 2014, Kyoto, Japan, September 28 - October 3, 2014.
4. LEPPÄNEN, J., et al., "Calculation of effective point kinetics parameters in the Serpent 2 Monte Carlo code", Annals of Nuclear Energy, 65, (2014) 272-279.
5. BROW, F., Kiedrowski, B., Bull, J., 2010. MCNP-1.60 Release Notes, Los Alamos National Laboratory, LA-UR-10-06235, (2010).
6. KIEDROWSKI, B., Brown, F., Bull, J., 2010. MCNP-1.60 Feature Enhancements and Manual Clarifications, Los Alamos National Laboratory, LA-UR-10-06217,(2010).
7. MESQUITA, A. Z., "Proposta de utilização do reator nuclear de pesquisa triga ipr-r1 do Centro de Desenvolvimento da Tecnologia Nuclear (CDTN) em pesquisa e treinamento", VI Congresso Nacional de Engenharia Mecânica, 2010.
8. DE CORTE, F., "The k0 - standardization method:;a A move to the optimization of neutron activation analysis", Ryksuniversiteit Gent, Faculteit Van de Wetenschappen, (in English),(1986).
9. MENEZES, M.A.B.C., JAĆIMOVIĆ, R. "Optimised k[sub]0-instrumental neutron activation method using I IPR-R1 reactor at CTDN/CNEN", Nucl. Instrum, Methods Phys Res., Sect. A, Accel. v. 564, Issue 2 (2006).
10. SOUZA, R. M. G. P., RESENDE, M. F. R., "Experiments proposed to 100kW and 250kW to increase power of the IPR-R1 TRIGA reactor", CNEN/CDTN, NI-AT4-005/98 Rev.1, Belo Horizonte (1998). (in Portuguese).

11. SOUZA, R. M. G. P., RESENDE, M. F. R., MESQUITA, A. Z. “Final Test Procedure to Increase the IPR-R1 TRIGA Reactor Power”, CNEN/CDTN, NI-AT4-03/02, Belo Horizonte (2002). (in Portuguese).
12. MESQUITA A. Z., “Experimental Investigation on Temperatures Distributions in a Research Nuclear Reactor TRIGA IPR-R1”, Ph.D Thesis, Universidade Estadual de Campinas, São Paulo, (in Portuguese), (2005).
13. DALLE, H.M., “Neutronic Evaluation of the IPR-R1 TRIGA Reactor for 250kW Operation”, CNEN/CDTN, NI-CT4-04/99, Belo Horizonte (1999). (in Portuguese).
14. GUERRA, B.T. “Obtenção do fluxo neutrônico nos terminais de irradiação do reator TRIGA IPR-R1 utilizando o Método de Transporte Monte Carlo”. Thesis, Departamento de Engenharia Nuclear, Escola de Engenharia da UFMG, Belo Horizonte, (in Portuguese), (2011).
15. POSTON, D. I.; TRELLUE, H. R., “User's Manual, Version 2.0 for Monteburns”. LA-UR-99-4999, 1999.
16. PELOWITZ, D.B., “MCNPX User's Manual”, LA-CP-07-1473, 2008.
17. ISOTALO, A. and Aarnio, P. (2011) "Higher order methods for burnup calculations with Bateman solutions." Ann. Nucl. Energy, 38 (2011) 1987-1995.
18. ISOTALO, A. and Aarnio, P. (2011) "Substep methods for burnup calculations with Bateman solution." Ann. Nucl. Energy, 38 (2011) 2509-2514.
19. ISOTALO, A. (2013) "Computational methods for burnup calculations with Monte Carlo neutronics." D.Sc. Thesis, Aalto University, 2013.
20. Triga Manual,General Atomics
21. MESQUITA, A.Z., Rezende, H.C., Tambourgi, E.B., 2007. Power calibration of the TRIGA Mark I nuclear research reactor. Journal of the Brazilian Society of Mechanical Sciences and Engineering XXIX, 240–245.
22. MESQUITA, A.Z., Souza, R.M.G.P., Rezende, H.C. “Thermal Power calibrations of the IPR-R1 TRIGA nuclear reactor” In: 20th International Congress of Mechanical Engineering - COBEM 2009, 2009, Gramado - RS. 20th International Congress of Mechanical Engineering - COBEM 2009. Rio de Janeiro:Associação Brasileira de Engenharia e Ciências Mecânicas - ABCM, 2009.
23. MESQUITA, A.Z., Hugo Cesar Rezende, Rose Mary Gomes do Prado Souza. Thermal power calibrations of the IPR-R1 TRIGA reactor by the calorimetric and the heat balance methods Progress in Nuclear Energy 53 (2011) 1197e1203