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Dosimetry assessment during the sipping test in the IPR-R1 TRIGA reactor using MCNPX

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ABSTRACT

A sipping test device recently constructed at *Centro de Desenvolvimento da Tecnologia Nuclear* – CDTN (Nuclear Technology Development Center - CDTN), Belo Horizonte, Brazil, will be used to inspect irradiated fuel elements cladding in the IPR-R1 TRIGA reactor. The device is important to check the integrity of the irradiated fuel elements cladding of this reactor, which can have been affected by corrosion over long periods of time. This paper describes the methodology used in the characterization of gamma doses received by the personnel during the manipulation of the sipping test device. Results are presented for dose rates calculated in the reactor pool surface using the Monte Carlo code MCNPX. Validation of MCNPX calculations of gamma doses were carried out thought comparison with experimental measurements. The difference between the measured and calculated values is lower than 13%. The results provided confirmatory evidence that during the sipping test the manual handling is safety.

1. Introduction

Brazil has three nuclear reactors in operation of similar characteristics: the Argonauta 500 W reactor of IEN (Rio de Janeiro), the IEA-R1 4.5 MW reactor of IPEN (São Paulo), and the IPR-R1 TRIGA 100 kW reactor of CDTN (Belo Horizonte). These three reactors are utilized for training, teaching, scientific research, and technological applications including sipping tests (Zeituni et al., 2004).

IPR-R1 is an open pool type TRIGA reactor cooled by natural circulation of light water, and having as fuel an alloy of zirconium hydride and uranium enriched at 20% in U-235. The reactor core host 58 aluminum clad fuel elements (FE) and 5 stainless steel-clad fuel elements located in five rings around the central thimble. The aluminum clad fuel elements (FE type 1) are consisted of two cylindrical graphite slugs at the top and bottom ends (reflectors) positioned below and above the active region, and two burnable poison composed by samarium oxide disks (Mesquita and Souza, 2010).

A recent sipping test device was constructed at *Centro de Desenvolvimento da Tecnologia Nuclear* – CDTN, Belo Horizonte, Brazil in order to allow the inspection of reactor fuel elements cladding of the TRIGA IPR-R1 reactor (Rodrigues, 2016). Three irradiated fuel elements (FE) will be selected and the integrity assessment will be performed by using the sipping test device. The defective fuel cladding can be identified by detecting the occurrence of radioactive fission products leaked to the sipping tank water, which is based on the detection of radioactive fission products by means of gamma-ray spectroscopy.

For the sipping test device design and safety analysis it is necessary to calculate the gamma radiation isodose around it, where the experiments will be conducted (Mesquita and Gual, 2015). In this work, the Monte Carlo MCNPX code simulation of the new sipping device is performed in order to verify the gamma irradiation dose that the reactor operator will be submitted at the time of device manipulation when sipping test is being performed in the reactor pool (Hendricks et al., 2008).

This first work will support the structural integrity evaluation of the TRIGA fuel elements, concerning fission product escape. The calculation of the isodose distribution curves around of the reactor surface is also a requirement for the elaboration of the Safety Analysis Report (SAR), necessary in the licensing process. This initiative to develop and improve the methods for structural integrity evaluation will be of relevance to disseminate safe use of nuclear reactors technology at CDTN. The experience will be employed in the design and operation of the RMB reactor (Brazilian Multipurpose Reactor) as this project is component part of the multi-annual programming of CDTN/CNEN, and consistent with the

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country's strategic objectives.

International Commission on Radiological Protection recommends the cumulative dose limit of 20 mSv/y averaged over five years for occupational exposed workers and 1 mSv/y for the public (ICRP, 2007). These limit values are adopted by the Brazilian Nuclear Energy Commission Basic Guidelines on Radiological Protection, in the norm CNEN-NN-3.01 (CNEN, 2014). The nuclear reactor is within a radiologically controlled area (RCA) and therefore the reactor operators are considered as occupationally exposed individuals (OEIs). The sipping test is expected to be held at the IPR-R1 research reactor for up to 10.5 h per year. The workers in controlled areas should not exceed 3/10 of the occupational dose limits. Just 6 mSv is 3/10 of the annual dose limit (20 mSv) for exposed workers (CNEN, 2011).

The objective of this study is to verify, through MCNPX calculations, the OEIs absorbed dose do not exceed the limits established by regulations. The resulting value will then be compared against the investigation level (allowed 3/10 of the annual limit for IOEs) and against experimental measurements for validation purposes. Dose rates of gamma radiation are estimated at the top surface of the reactor pool, where the experiments will be conducted. This position corresponds to the worst case scenario of an operator handling the sipping test device.

2. Materials and methods

The sipping device is made of 2.68 g/cm^3 aluminum. Fig. 1 shows a picture of the shipping device constructed. The component parts are described in a previous work (Rodrigues, 2016). The ribbings of the base were modeled by using the coordinate transformation card (TR) of MCNPX code.

The Monte Carlo code MCNPX v 2.6.0 using ENDF/B-VII nuclear data evaluation was used for modeling of the experiment. The simulated model of the IPR-R1 reactor was based on previous studies performed by Dalle (2005). The sipping device was introduced inside the reactor pool in the original model composed of; steel, aluminum and concrete vessel, surrounding nuclear reactor core. Table 1 summarizes the material compositions of aluminum,

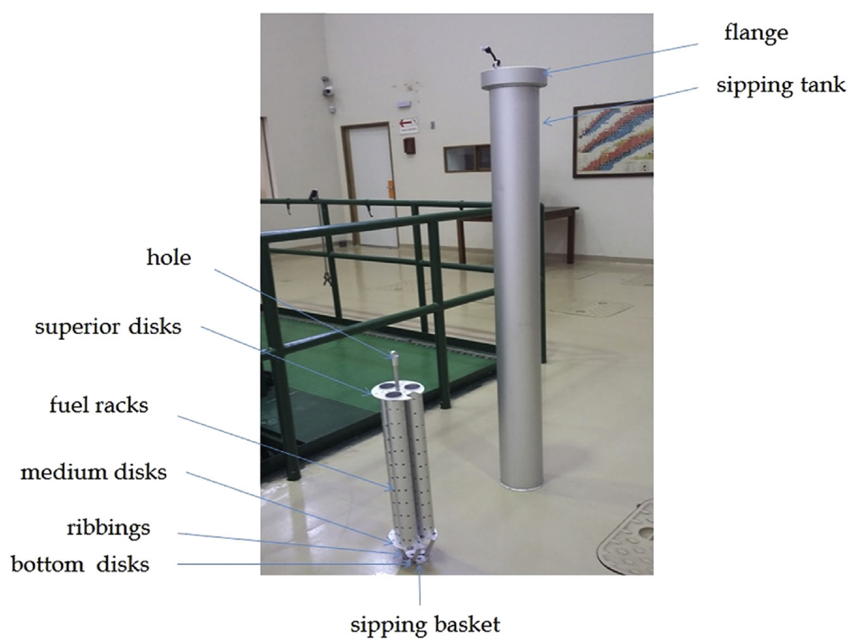

Fig. 1. Details of the basket within the tank of the sipping device positioned in the reactor hall.

Table 1
Material specification of aluminum, ordinary concrete and stainless steel.

Aluminum	
Density (g/cm^3)	2.68
Material composition w_t (%)	Al-27(1.000000)
Concrete	
Type	Ordinary
Density (g/cm^3)	2.453
Material composition w_t (%)	H-1(0.0221); C-12(0.002484); O-16(0.57493); Na-23(0.015208); Mg-nat(0.001266); Al-27(0.019953); Si-nat(0.304626); K-nat(0.010045); Ca-nat(0.042951); Fe-nat(0.006435)
Stainless Steel	
Type	SS 304/Fe + Cr + Ni (Cr:−0.18; Fe:−0.74; Ni:−0.08)
Density (g/cm^3)	8.02
Material composition w_t (%)	Cr-50(0.007513); Cr-52(0.150659); Cr-53(0.017412); Cr-54(0.004416); Fe-54(0.041777); Fe-56(0.68007); Fe-57(0.015987); Fe-58(0.002165); Ni-58(0.053758); Ni-60(0.021421); Ni-61(0.000947); Ni-62(0.003068); Ni-64(0.000807)

ordinary concrete and stainless steel used in the MCNPX models.

Table 2 shows major design parameters of the sipping device.

The mixture composition of the fuel element (U-ZrH) aluminum type and graphite as reflector used in the calculations is shown in Table 3.

It is recognized that the convergence of Monte Carlo (MC) calculation for large-scale systems is very time-consuming. Complete information for individual MC particle tracks are required in order to perform accurate calculations. Aiming to reduce variance, an equivalent reactor physics model was developed. To construct equivalent model, the complete geometry was subdivided, into two separated regions:

I- The segment bottom regions (only with reactor core);
II- The superior region (only with sipping device).

This equivalent reactor model is adopted in order to reduce computing time. In the case of simulating the non-separated model, for an acceptable running time (less than 36 h), the number of fission neutrons that reach the superior region is negligible, resulting unsatisfactory statistical results. Using this procedure the calculation of gamma radiation isodose will be calculated around the sipping test device, where the experiments will be conducted.

The isodoses curves will be obtained from "TMESH" mesh type 3 tally using "total" option to superimpose a rectangular meshgrid over the simulation geometry. The "total" allows scoring the equivalent to an F6 heating tally for gammas rays in units of MeV/cm^3 normalized by source particle. A rectangular mesh is defined with 1 mm resolution. MATLAB software was adopted to generate views.

Table 2
Major design parameters of the sipping device.

Characteristics	Dimension
Guide tube	
External radius of guide tube	2.6225 cm
Number of guide tube	3
Thickness	0.3 cm
Height	57.5 cm
Angle	120°
Radius from the center of the basket to the tube guide	4 cm
Strap with a hole	
External diameter	1.3 cm
Height	59.50 cm
Ribbing of basket	
Radius of superior and lower slab	7.0 cm
Radius of base slab	3.8 cm
Radius from the center of the basket to the ribbing	3.0 cm
Number	4
Thickness	0.3 cm
Height	10.0 cm
Angle	35°
Basket	
Height	57.5 cm
Tank	
External radius	7.49 cm
Thickness	0.16 cm
Height	171.6 cm
Reinforcement ring	6.0 cm
Diameter of hole at water level	1.2 cm
Distance to the reinforcing ring	1.5 cm

Table 3
Fuel element (U-ZrH) aluminum type compositions of the TRIGA reactor.

Chemical element	Atomic number density (atoms/barn cm^2)
U-235	2.3693 e−04
U-238	9.5445 e−04
Zr	3.6256 e−02
H	5.8724 e−02
Total	7.6881 e−02

Detailed photon physics treatments including photoelectric effect with the fluorescence production, incoherent and coherent scattering and pair production, has been considered in the range of energy between 0.001 and 15 MeV through the card "PHYS: P".

To determine the gamma doses (D_γ), different tallies were employed: average surface fluence tally type F2 (in particles/cm^2); track-length type F4 in a given cell (in particles/cm^2), and Energy Fluence *F5 type (in MeV/cm^2). For the cases that used F2 and F4 tallies, a surface was delimited on the top of the reactor pool to account for the particles.

The following hypotheses were taken into account for the simulations: (i) only gammas transport (mode p), (ii) spectrum of gamma radiation from the reactor TRIGA FE obtained by decay during 1 day after reactor shutdown.

The source term of the TRIGA reactor was calculated using ORIGEN-ARP isotopic depletion and decay analysis system (Bowman, 2011). For reactor power history registered in the reactor operation data logbook. The evolution of IPR-1 TRIGA reactor since the begin of life in November 6, 1960 until current days and decay of 1 day after shutdown was considered. Table 4 presents the source term spectra and intensity obtained. The values are for 1 fuel element occupying the B position of the inner ring (central rings). These fuel elements are the most demanded from the viewpoint of heat generation, as reported by Mesquita et al., 2012.

The gamma dose rates (rem/h) are calculated using the neutron flux-to-dose equivalent rate conversion factors and DE/DF cards, according to NCRP (1971) Report 38 presented in Annex H of MCNPX manual. The normalization is obtained by multiplying the total gamma source (S) and the dose rates are calculated from the equation:

$$\dot{D}(rem/h) = ftally \cdot T \cdot S \tag{1}$$

where:

T: neutron flux-to- dose equivalent rate conversion factors (rem/h)/(particles/cm^2-s);
S: source emission (particles/s).

When *F5 tally is used, values of the mass attenuation coefficients (μ_{tr}/ρ) to the air tabulated in Attix (1991) were introduced

Table 4
Gamma source for 1 FE.

Gamma source		1 day after shutdown
Group	MeV	Bq
1	2.74E+12	2.74E+12
2	8.39E+11	8.39E+11
3	1.29E+12	1.29E+12
4	1.45E+12	1.45E+12
5	3.86E+11	3.86E+11
6	1.17E+12	1.17E+12
7	2.15E+12	2.15E+12
8	3.41E+11	3.41E+11
9	3.44E+11	3.44E+11
10	3.61E+11	3.61E+11
11	7.43E+10	7.43E+10
12	2.39E+10	2.39E+10
13	9.17E+09	9.17E+09
14	1.94E+08	1.94E+08
15	3.26E+07	3.26E+07
16	1.37E+05	1.37E+05
17	2.54E-02	2.54E−02
18	5.34E-03	5.34E−03
TOTAL (photons/s)		1.12E+13

using DE/DF cards. The result was converted from MeV/g to mGy with the multiplier tally (FM) by multiplying by 1.6021E -07 and 3600 s/h. It is used the equivalence that 1 J/kg (physical unit) of gamma absorbed dose corresponds to 1 Sv of equivalent dose.

As the results of MCNP are normalized to a particle emitted by the source, it is necessary to scale the calculated to actual doses of system multiplying the obtained results with the above equation to the total neutrons emission per second (source). For the normalization of the results the source 2.714e+12 gamma/s/FE presented in Table 4 was used.

Criticality calculations were performed using the KCODE card available in MCNPX code to estimate the neutron multiplication factor (k_{eff}). The convergence of the results together with small computational uncertainty (~30 pcm) was obtained for the code running 950 cycles in total, with 5000 neutrons per cycle and 900 active cycles. In order to speed-up the calculations, the MCNP code was run in parallel in a 20 processors computer cluster.

3. Results

The equivalent reactor physics model is presented in Fig. 2. In the Fig. 2 diagram (I) is the segment bottom regions (only with reactor core), and (II) is the superior region (only with sipping device). Vised 22S visualization tool was used to illustrate the MCNPX model (Schwarz et al., 2008).

Fig. 3 shows the MCNPX XY view of upper and medium disk (in the left part) and the picture (in the right part) of the sipping test basket.

Fig. 4 shows the MCNPX XY view of bottom disk (in the left part) and the picture (in the right part) of the sipping test basket.

Fig. 5 shows the MCNPX XY view of the base (in the left part) and the picture (in the right part) of the sipping test basket.

As seen from the picture, the model is very close to real sipping test device, as the simulation using the Monte Carlo MCNPX reproduces with extensive detail the original geometry and composition. Fig. 6 illustrates the complete MCNPX model of the sipping test device in the calculation position in the reactor's pool.

Fig. 7 shows the fuel element type 1 (aluminum cladding, graphite reflectors positioned below and above the active region, burnable poison composed by two samarium oxide disks) in the left side. In the right side, the basket of sipping test device loaded with three FE inside the tank.

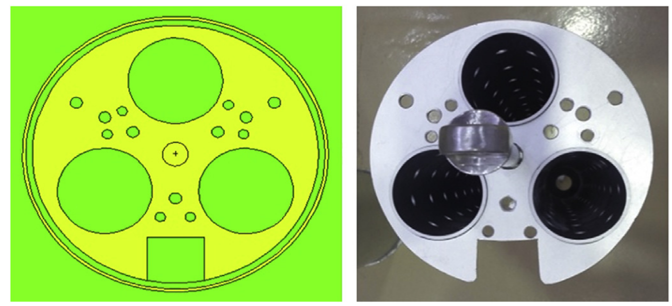

Fig. 3. Upper and medium disk of the sipping test basket.

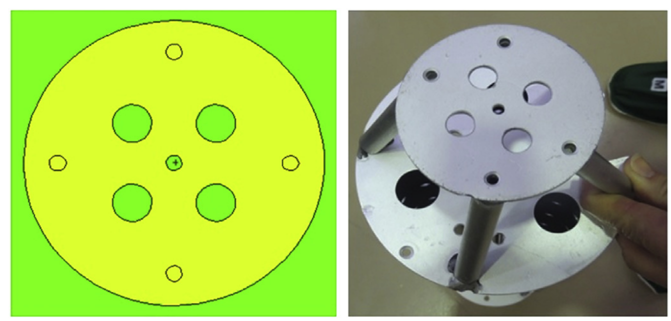

Fig. 4. Bottom disk of the sipping test basket.

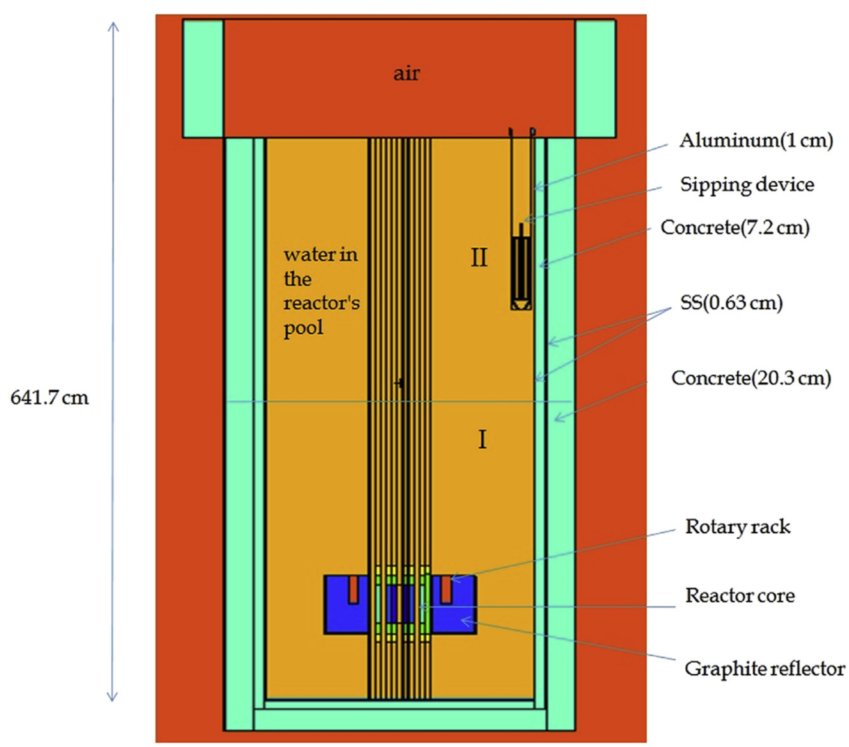

Fig. 2. MCNPX model with the three separated regions.

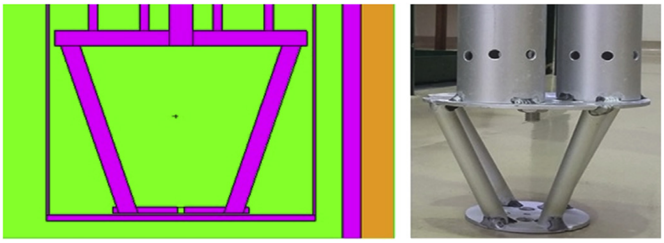

Fig. 5. Base of the sipping test basket.

Criticality calculations were carried out for the bottom of the reactor pool (I) with all 63 FEs in the reactor core and without the 3 FE in the B ring. The results are presented in Table 5. Criticality in the upper part (II) of sipping test device with three fuel elements of the B position of the inner ring (central rings) of the reactor core IPR-R1 TRIGA was also calculated. Inner ring was chosen because it hosts the FE with the highest amount of fissile material.

The results show that the reactor core stay subcritical when 3 FE of B ring are removed. It means the reactivity excess of the core is not enough to start the reactor during the sipping experiment not even if the control bars are removed from the core. Subcriticality condition is also observed in the basked of the sipping device, namely Part II with 3 FE of B ring.

The visualization of gamma dose distribution with MATLAB software obtained with a 100 × 100 × 100 rectangular mesh grid (RMESH) centered at the center of sipping device with 1 mm resolution are presented in Figs. 8 and 9. It is showing MCNPX XY views of the gamma doses distributions of the reactor core with and

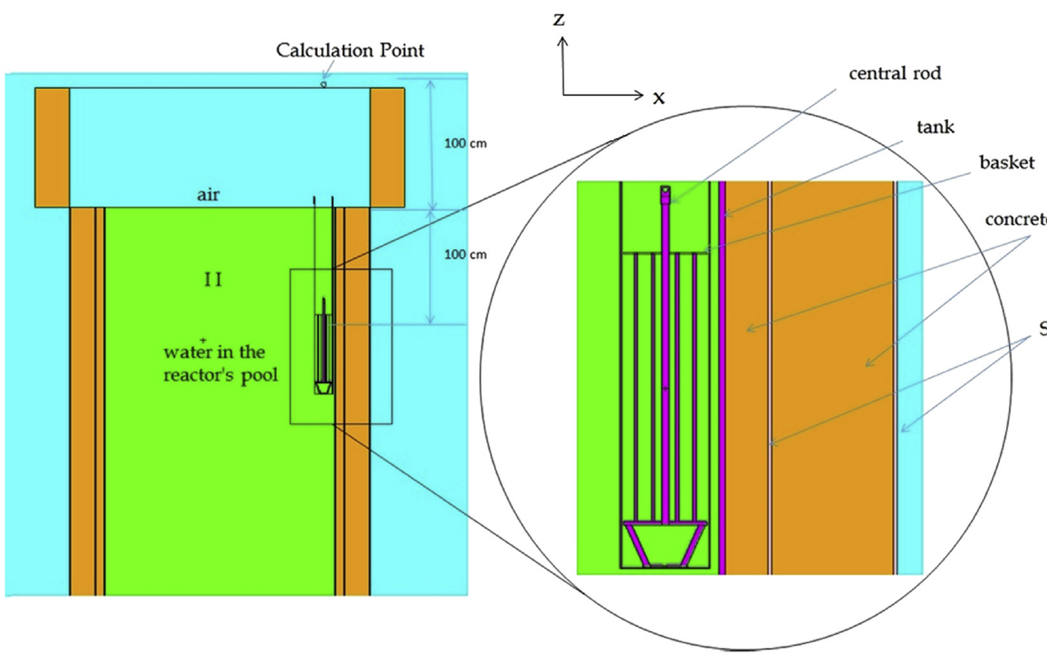

Fig. 6. Full MCNPX model of the sipping test device in the reactor pool (part II).

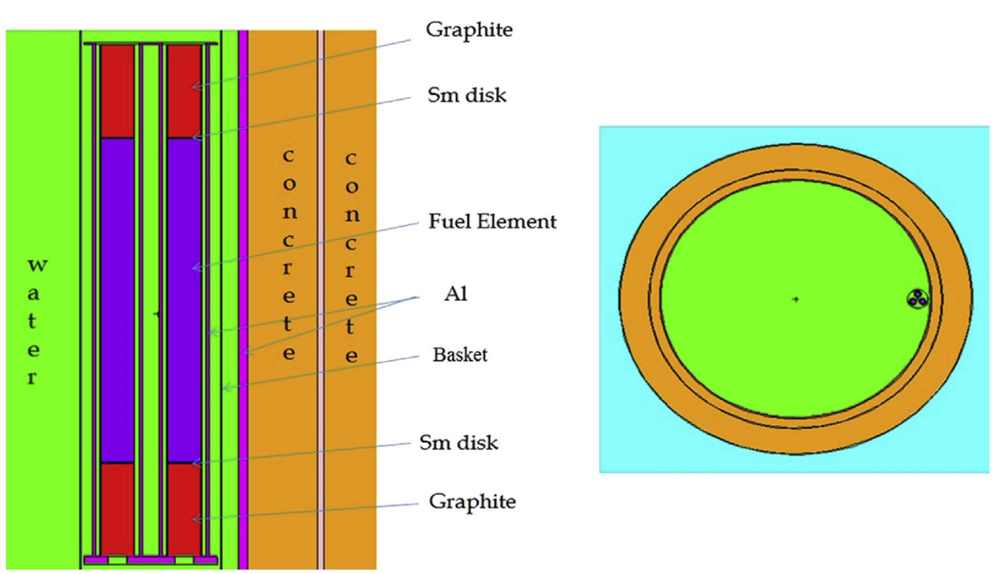

Fig. 7. The fuel element type 1 of IPR-1 TRIGA within the basket of sipping test device.

Table 5
Comparison of neutron multiplication factor for different configurations.

Position	$k_{eff} \pm \sigma$
Part I with the full FEs	1.00907 ± 0.00004
Part I without 3 FE of B ring	0.97000 ± 0.00004
Part II with 3 FE of B ring	0.33485 ± 0.00114

without the three FE of the B ring (central rings) that will be under investigation in the sipping device.

Mode n p (neutrons and gamma rays induced by the interaction of neutrons in the structural materials are transported) is not used because of neutron that reach the top of the reactor pool is negligible. Only the calculation in mode p was made. This calculation did not consider the gamma rays originated from the activated cladding, because in ^{28}Al ($T_{1/2}$ = 2.45 min), the activation is negligible. The results are restricted to fuel elements with aluminum cladding.

The distance between the basket of the sipping device and reactor pool wall is 31.41 cm. Further studies may investigate the influence of decreasing the distance between the sipping device and reactor pool wall.

In the calculation with SDEF the relative error associated to the results were kept lower than 5% for all points of interest, requiring 150 million stories for the *F5 tally. For other tallies the relative error remained lower than 10%. The computation time was 1.5 days.

For validation of simulations, the results of gamma doses rate are compared with the measured dose presented in the technical report CDTN/CNEN NI-SERTA-02/13 (Oliveira et al., 2013). In these experimental measurements the decay time after reactor shutdown was 1 day. This gamma source is presented in Table 4. This document is respecting an experiment of the manipulation made from a single fuel element, which has been lifted from the core

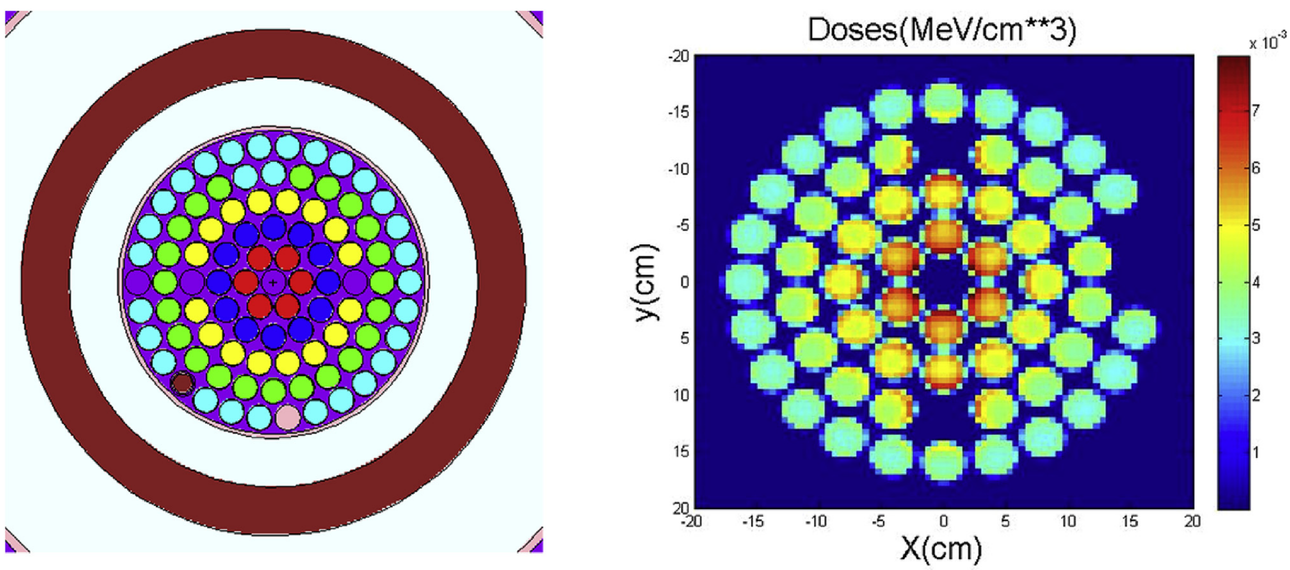

Fig. 8. MCNP model of the reactor core in 2D of gamma doses distribution at z = 0 using MCNPX.

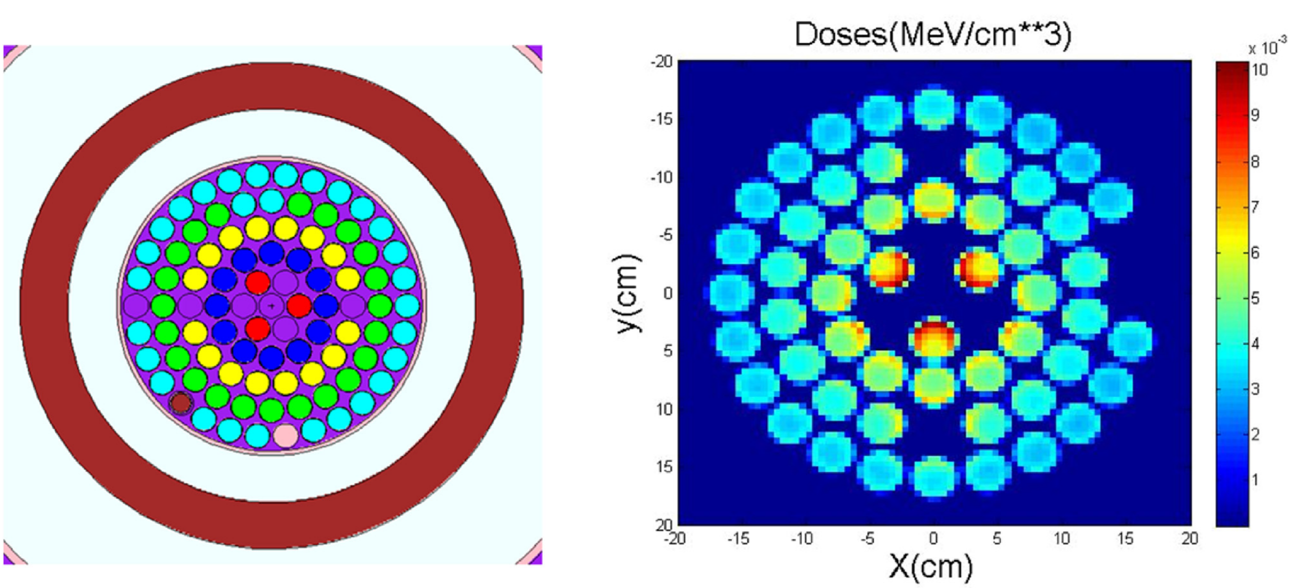

Fig. 9. MCNP model of the reactor core without the three fuel elements that will be investigated in 2D of gamma doses distribution at z = 0 using MCNPX.

position next to the edge of the pool. These measurements were performed by reactor operators in order to predict the dose that an individual worker would be exposed if a fuel element is handling in different positions. Table 6 summarizes MCNPX calculated gamma dose rates at the interest points during the sipping test. The tank is loading a basket containing 3 FEs for 100 cm of water layer between the FE and the surface and 100 cm of air between the surface and the calculation point.

The gamma doses rate was 0.08 mSv/h for a single fuel element loading in the experiment. For the comparison the considered gamma dose rate will be 0.24 mSv/h because three fuel elements are stored in the sipping test device under investigation. Due to the self-shielding being disregarded in the sum, this assumption is conservative respect gamma rate. With regard to the neutron multiplication factor, separate calculations were executed to demonstrate the assembly keeps subcritical (Table 5).

Table 6
Comparison between measured and calculated gamma doses rate for 100 cm of water layer and 100 cm of air.

tally	MCNPX D (mSv/h)	Measured D (mSv/h)	% Difference
F2	0.316	0.24	32
F4	0.314	0.24	31
*F5	0.268	0.24	12

The results of F2 and F4 tallies are equivalent and the difference with respect to experimental result is less than 33%. The results of the *F5 tally were the closest to the experimental results, being the difference less than 13%. The simulations have overestimated experimental dose rate values and the most conservative estimation points to a difference less than 33%. Fig. 10 shows the 3D representation of the gamma dose rate distribution and Fig. 11 shows the normalized isodose distribution.

It is observed the maximum gamma rays dose coincide with the position where the sipping device is positioned and decrease rapidly with as a function of distance from the sipping device to the pool surface. The total accumulated dose received by the reactor

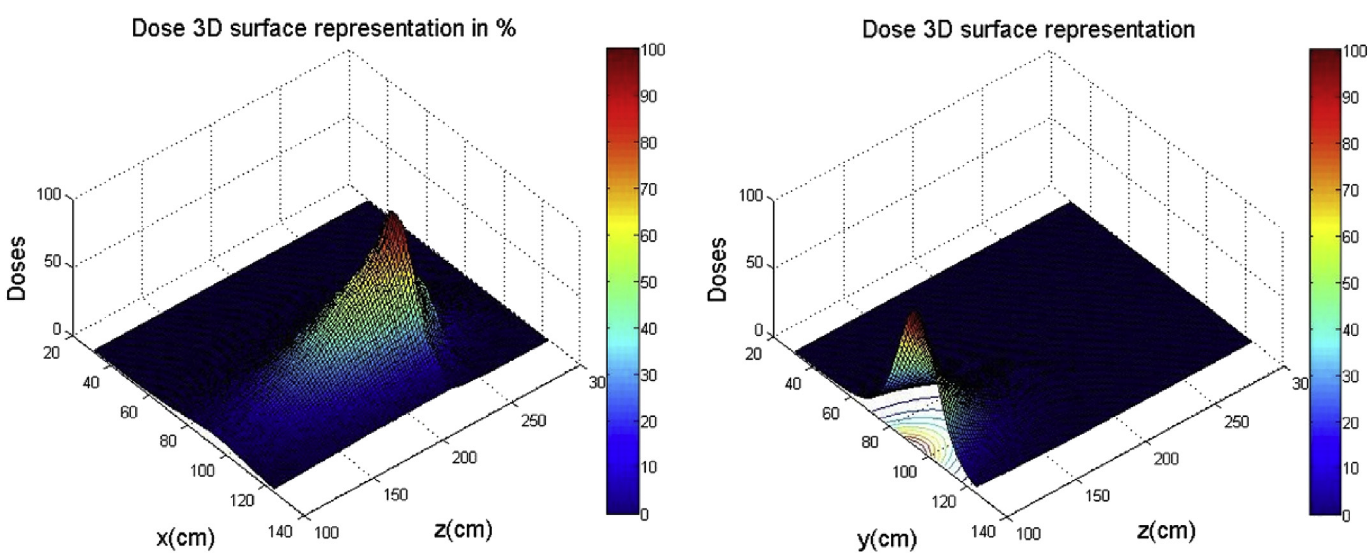

Fig. 10. 3D representation of XZ and YZ view of the gamma dose rate distribution using MATLAB.

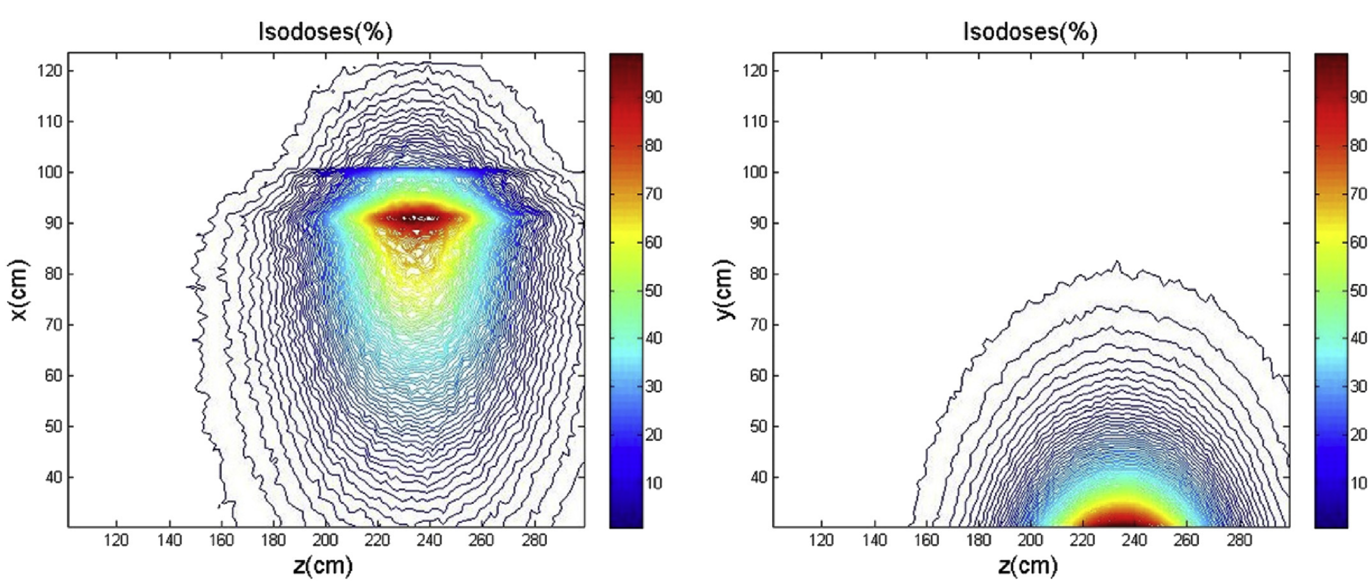

Fig. 11. Isodose curve in % representation of XZ and YZ view of the gamma dose rate distribution using MATLAB.

operator in the experiment is expressed as:

$$\dot{D}_{\text{exp}} = \dot{D} \cdot N \cdot t = 0.316 \text{ mSv/h} \cdot 0.5\text{h} \cdot 21 \text{ time/year} \\ = 3.318 \text{ mSv/year}, \quad (2)$$

where:

N: Numbers of times the sipping test device is manual handled (i.e. 3 FE loading in the sipping device 21 times resulting the total 63 FE in the nuclear core.
t: time of exposure (assumed 0.5 h)

According to Equation (2), the experiment with sipping test device involves 3.318 mSv/year (taking into account the most conservative value calculated). Ensuring that the operator performs the sipping test once per year at the IPR-R1 research reactor, the total cumulative dose will not exceed the limit established by the standards for OEIs in controlled area, namely 6 mSv/year (CNEN, 2011). The maximum hours that operators of sipping device can work will be:

$$t = \frac{\dot{D}_{\max}}{\dot{D}_{\exp}} = \frac{6mS\nu/year}{0.316mS\nu/h} = 19 \text{ hour/year of working} \quad (3)$$

Therefore, the total time that could be spent performing the experiment is 19 h per year.

4. Conclusions

The results showed the estimated cumulative absorbed dose that the operator will be exposed when the sipping test device is positioned for 100 cm of water layer, 100 cm of air, and at 31.43 cm distance from the reactor pool wall is 3.318 mSv/year. This value, validated by comparison with experimental measurements, does not exceed the limits for occupational exposure of 6 mSv/year in controlled area. Therefore, the manual handling of the sipping test device by the workers is secure from dosimetry assessment.

The dosimetry assessment reported here is important for safe operation of either research reactors or power reactors. The sipping test device used for research reactors can be later applied to power reactors.

An MCPX model of sipping test device was implemented for the first time in CDTN, which is necessary for calculations of gamma radiation isodose at the top surface of the reactor pool, where experiments will be conducted.

Further research in this area may include the calculation of stainless steel cladding fuel elements in order to take in account the activated cobalt, which could be significant in the result. It is also recommended to perform a study of distance variation between the basket of sipping device and reactor pool edges to analyze the effects on irradiation doses.

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