Id.: EN 28

SPENT FUEL POOL CRITICALITY ANALYSIS THROUGH MCNPX 2.6.0 CODE

Jean A. D. Salomé[1], Clarysson A. M. Silva[1], Raphael H. M. Silva[1], Cláubia Pereira[1]

[1]Universidade Federal de Minas Gerais, Escola de Engenharia, Departamento de Engenharia Nuclear
Av. Antônio Carlos, 6627 Campus Pampulha – CEP 31270-901 – Belo Horizonte – MG
jadsalome@yahoo.com.br, clarysson@nuclear.ufmg.br

Keywords: Spent Fuel Pool; Criticality Analysis; MCNPX 2.6.0 Code

ABSTRACT

The management of the fuel cycle is an important stage for power generation through the nuclear matrix. The back-end strategies encompass all post-unloading procedures of the spent fuel from the reactor core. The handling of irradiated material can be carried out in two different ways: the final deposition in some type of repository, in the open fuel cycle, called once-through; or an intermediate deposition, waiting some reprocessing technique in the closed fuel cycle. In both cases, the water pools have been used as an intermediate part of the process, since they provide satisfactory shielding for possible radioactive emissions from the stored fuel, as well as its cooling. Thus, studies that produce knowledge about wet repositories, whether temporary or permanent, are very useful, because they generate data and information regarding the behavior and evolution of spent fuel after its removal from the reactor core. Such surveys allow better material management, adoption of more efficient projects and the observance of parameters and legal norms that provide safety and feasibility in the use of this type of energy resource. In this research, a model of spent fuel pool has been simulated using the MCNPX 2.6.0 code. The geometry and design of this installation, the most relevant components and constituent materials are according to the dimensions and compositions found in the references. Several configurations of different loads have been simulated in the modeling. In these loads, mixtures of fresh and irradiated fuel elements have been made as indicated in the literature. The fuel bundles have been burnt at levels of 33 and 52 GWd/tHM. Three types of fuels/geometries, obeying the respective compositions of each case, have been inserted in the pool to be investigated. The first type consists of solid UO_2 rods, enriched to 5.0% by weight of ^{235}U before burning; the second type, also of solid geometry, composed of a MOX with the percentage of 5.0% of fissile material, and the third, with annular geometry and the same initial composition as the first type. The purpose of this study is to present an analysis of the criticality safety of the installation. The results obtained demonstrate that the models, even for innovative fuels, such as MOX and annular geometry rods, are in accordance with the limits proposed by the norms that regulate the matter.

1. INTRODUCTION

In this research, a model of spent fuel pool has been simulated using the MCNPX 2.6.0 code. The geometry and design of this installation, the most relevant components and constituent materials are according to the dimensions and compositions found in the reference [1]. Several configurations of different loads have been simulated in the models. In these loads, mixtures of fresh and irradiated fuel elements have been made. The purpose of this study is to determine, by executing MCNPX 2.6.0 code, an analysis about criticality safety of the installation. The results obtained demonstrate that the models, even for innovative fuels, such as MOX and annular geometry rods, are in accordance with the limits proposed by the norms that regulate the matter [2].

2. METHODOLOGY

This work has considered two geometries for the fuel elements studied. A typical 16 x 16 fuel rods PWR lattice and a 13 x 13 fuel assembly with rods of annular geometry [3, 4]. For convenience, all fuel elements have the same external sizes. The geometric characteristics and the constituent materials of the analyzed fuel elements are presented in Fig. 1 and Tab. 1, respectively.

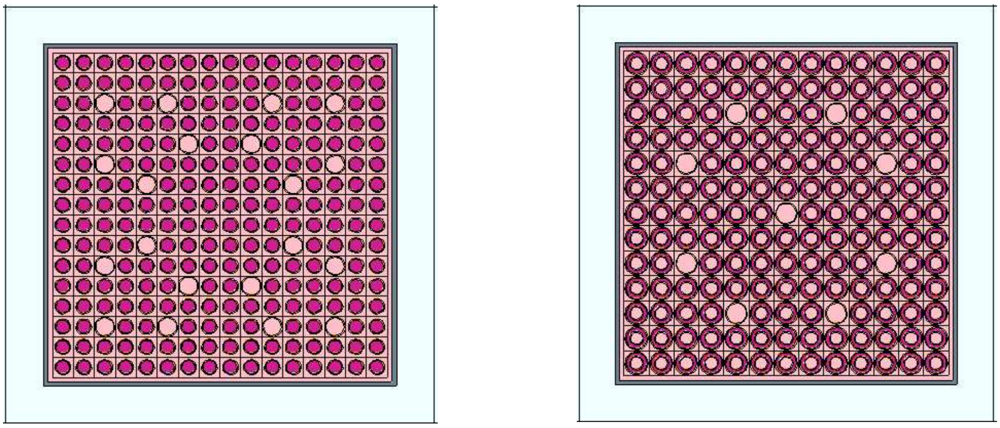

Fig. 1. Cross-sectional view of the simulated model by MCNPX 2.6.0. At left is the typical 16 x 16 PWR lattice and at right is 13 x 13 fuel assembly with annular geometry.

Tab. 1. Geometric data and materials for the analyzed assemblies.

Data	Solid fuel (16 x 16)	Annular fuel (13 x 13)
Pin pitch (cm)	1.43000	1.52533
Rod inner radius (cm)	—	0.33665
Inner clad outer radius (cm)	—	0.39380
Fuel inner radius (cm)	—	0.40000
Fuel outer radius (cm)	0.45830	0.61550
Outer clad inner radius	0.46590	0.62310
Rod outer radius (cm)	0.53850	0.69570
Active fuel height (cm)	391.6	391.6
Cladding material	Zircaloy-4	Zircaloy-4
Gap material	Helium	Helium
Fuel rods per assembly	236	160

Three configurations for these arrays have been adopted. The first two structures are composed for solid pin cells. One has been filled with UO_2 enriched to 5.0% by weight of ^{235}U. The other has been composed of a MOX with the percentage of 5.0% of fissile material. The third arrangement, with annular geometry, has the same initial composition that the first assembly. In all the cases the tubes for the control rods are filled with water. Tab. 2 presents the isotopic compositions of the fresh fuels.

Tab. 2. Isotopic composition (atoms/barn.cm) of UO_2 and MOX fuels.

Nuclide	UO_2	MOX
^{235}U	1.17430E-04	5.33524E-06
^{238}U	2.20300E-02	2.13880E-02
^{238}Pu	–	4.47779E-05
^{239}Pu	–	9.80991E-04
^{240}Pu	–	4.64944E-04
^{241}Pu	–	1.66477E-04
^{242}Pu	–	1.26932E-04
^{16}O	4.64090E-02	4.68992E-02

The irradiation of a single fuel bundle from each one of these arrays has been simulated through the SCALE 6.0 code in TRITON/KENO-VI control module. The SCALE 6.0 code is a computer software system developed by the Oak Ridge National Laboratory. It has been used largely to solve problems about neutron transport and to produce lattice physics models [5]. In this research, the fuel bundles were burnt continuously and the levels of 33 and 52 GWd/tHM have been considered. The burnup of 33 GWd/tHM covers the largest part of the fuel elements stored in the water pools [6]. The burnup of 52 GWd/tHM represents the profile change in the nuclear reactor fleet aiming the efficiency at energy generation [7, 8].

To model the pool and loads, and to perform the calculations of criticality the MCNPX 2.6.0 code has been used. It is a code used in several situations involving particle physics. The MCNPX 2.6.0 code has capabilities, especially in transmutation, burnup and delayed particle production. However, from the point of view of this research, the main feature of the code is the ability to track almost all particles in practically all ranges of energy [9]. Regarding the criticality of a system, such tool is essential for tracking neutrons.

The spent fuel pool, which has been evaluated for this research, is an ordinary light-water pool model that has been found in several plants around the world. The installation must ensure the subcriticality and structural integrity of the bundles during normal operating or under accident conditions, to avoid corrosion due to the surrounding fluid or thermal effects, to remove the heat generated for the spent fuel assemblies and to shield the external environment and operators from possible radioactive emissions [1, 10].

Basically, the spent fuel pool consists of two contiguous rectangular regions filled with a total of 1084 specific racks to support the fuel elements according to each situation. The Region 1 is divided into 4 cells, named A1 to A4 and receives a maximum load of 264 racks filled with fresh fuel assemblies. The Region 2 is divided into 8 cells, called C1 to C8 and gets until 820 irradiated fuel bundles. The water of the pool is borated at 2300 ppm of boron [1]. Fig. 2 shows top and transversal views of the spent fuel pool and respective regions. Tab. 3 indicates the geometric information and constituent materials of the spent fuel pool and the racks.

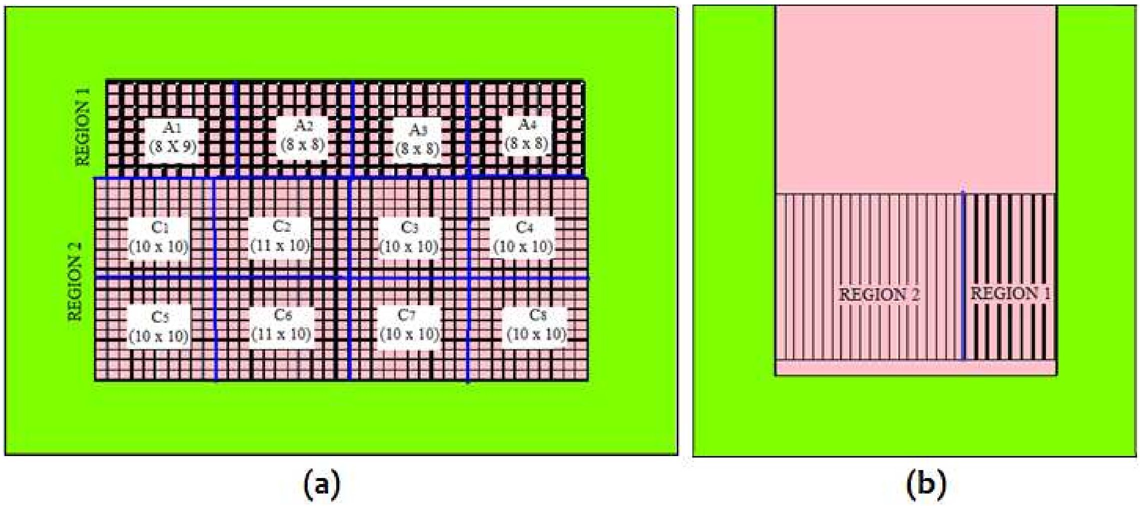

Fig. 2. (a) Top and (b) Transversal views of the spent fuel pool and respective regions simulated by MCNPX 2.6.0.

Tab. 3. Geometric data and materials for the spent fuel pool model.

Dimensions (cm)		**Region 1**	**Region 2**
Spent Fuel Pool	Width	234.50	487.50
	Length	969.05	999.30
	Depth	1190.00	1190.00
	Lining thickness	2.54	2.54
	Walls thickness	180.00	180.00
Rack	Cell 8 x 8	234.95 x 234.95 x 533.60	–
	Cell 9 x 8	234.95 x 234.95 x 533.60	–
	Cell 10 x 10	–	243.60 x 243.60 x 533.60
	Cell 11 x 10	–	267.90 x 243.60 x 533.60
	Wall thickness	2.565	0.090
Pitch distance	Storage cells	29.25	24.30
	Fuel Assemblies	0.64	0.74
Borated sheet steel	Width	23.60	23.70
	Thickness	0.30	0.25
Material		**Region 1**	**Region 2**
Spent Fuel Pool	Coolant	Light water	
	Lining	304 Austenitic Stainless Steel	
	Walls	Reinforced Concrete	
Neutron absorber steel		304B Borated Stainless Steel	
Storage cell		316 Austenitic Stainless Steel	

Mixed loads of fresh and burnt fuel elements have been simulated inside the adopted storage pool to investigate if the arrangements are in accordance with the criticality safety limits required by the rules for storage of nuclear fuel [2]. The loading of the regions and cells of the spent fuel pool and the analysis of the criticality of these systems have been carried out in two different steps. On the first step, the insertion of fresh fuel in Region 1 has been evaluated for filling cells A1 to A4. Region 2 and its respective cells has been remained empty at this stage since this area is used for storage of spent fuel. On the second step, the Region 2 has been filled through insertion of irradiated fuel, gradually, in cells C1 to C8. On this stage of criticality assessment, the cells in Region 1 have been considered to be fully charged with fresh fuel. The nomenclature criteria for each of these cases have been clarified ahead.

3. RESULTS

3.1. Criticality Analysis of the Fuel Elements with the Burnup

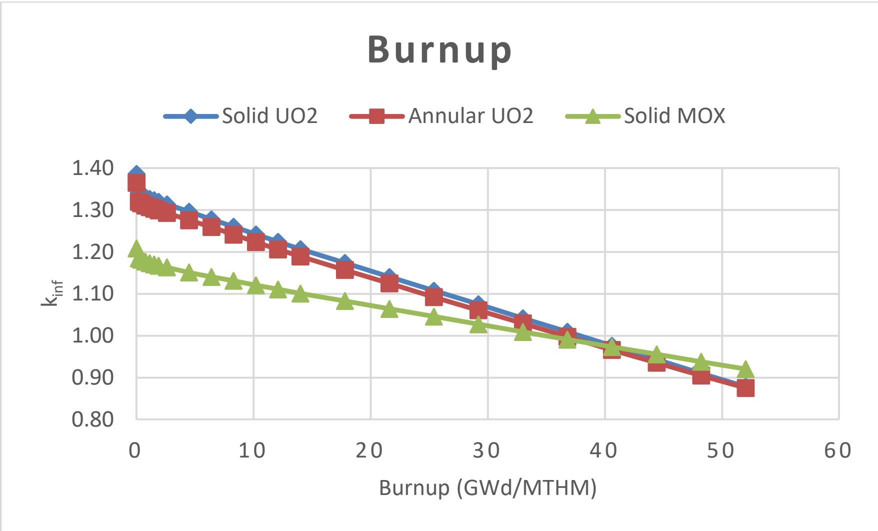

Fig. 3. The infinite multiplication factor (k_{inf}) versus burnup for the studied assemblies.

The infinite multiplication factor (k_{inf}) versus burnup for the studied assemblies is plotted in Fig. 3. As it has been observed, at the beginning of life, the fuel elements filled with UO_2 have a criticality higher than the criticality for the assembly filled with MOX. However, after a burnup at 40 GWd/tHM, the behavior of the curves has gotten an inverse aspect and the criticality for the MOX system reaches higher values. This behavior has been verified in other assessments [11, 12]. This happens due to the presence of plutonium isotopes (*e.g.* ^{240}Pu and ^{242}Pu) found in the MOX at the beginning of life, which act as neutron absorbers, hardening the neutron spectrum and reducing the fuel reactivity. It is also seen that the reduction in the criticality of the MOX fuel and, consequently, its reactivity, is less affected with the burnup along the time. The main cause is the transmutation of ^{240}Pu into fissionable ^{241}Pu, demanding less excess in reactivity for the MOX [13].

3.2. Criticality Analysis of the Fuel Elements Inside the Spent Fuel Pool

Following, the values of the effective multiplication factor (k_{eff}), calculated by the MCNPX 2.6.0 code for the loadings of the fuel elements inside the storage pool have been shown for both fresh and irradiated bundles in Tab. 4. The standard deviation estimated by the used code is in the order of 2.0 x 10^{-4}. Each case has a particular configuration. For instance, the first couple of numbers determines the level of the burnup (00 fresh fuel, 33 GWd/tHM and 52 GWd/tHM). After the letter P, the first number represents the geometry/fuel (1 - solid UO_2 rod, 2 – solid MOX rod and 3 – annular UO_2 rod). On the sequence, the condition of the fuel (1 – fresh fuel or 2 – irradiated fuel). Finally, the respective region and cell of the last loading (A1 to C8).

Tab. 4. k_{eff} calculated by MCNPX 2.6.0 for the evaluated cases

Fresh Fuel Assemblies		**Fuel Assemblies Burnt at 33 GWd/tHM**		**Fuel Assemblies Burnt at 52 GWd/tHM**	
Case	**k_{eff}**	**Case**	**k_{eff}**	**Case**	**k_{eff}**
00P11A1	0.29633	33P12C1	0.79538	50P12C1	0.66350
00P11A2	0.29679	33P12C2	0.79683	50P12C2	0.66441
00P11A3	0.29666	33P12C3	0.79719	50P12C3	0.66472
00P11A4	0.29679	33P12C4	0.79696	50P12C4	0.66529
00P21A1	0.93255	33P12C5	0.79703	50P12C5	0.66507
00P21A2	0.93348	33P12C6	0.79776	50P12C6	0.66584
00P21A3	0.93357	33P12C7	0.79774	50P12C7	0.66606
00P21A4	0.93399	33P12C8	0.79848	50P12C8	0.66591
00P31A1	0.28395	33P22C1	0.93342	50P22C1	0.93377
00P31A2	0.28414	33P22C2	0.93377	50P22C2	0.93357
00P31A3	0.28414	33P22C3	0.93236	50P22C3	0.93394
00P31A4	0.28423	33P22C4	0.93339	50P22C4	0.93391
		33P22C5	0.93266	50P22C5	0.93406
		33P22C6	0.93189	50P22C6	0.93382
		33P22C7	0.93247	50P22C7	0.93363
		33P22C8	0.93203	50P22C8	0.93333
		33P32C1	0.79103	50P32C1	0.71038
		33P32C2	0.79254	50P32C2	0.71133
		33P32C3	0.79250	50P32C3	0.71167
		33P32C4	0.79326	50P32C4	0.71197
		33P32C5	0.79313	50P32C5	0.71196
		33P32C6	0.79393	50P32C6	0.71227
		33P32C7	0.79421	50P32C7	0.71337
		33P32C8	0.79419	50P32C8	0.71298

Figs. 4 and 5 show the normalized criticalities to fresh and mixed (fresh and irradiated) bundles into the spent fuel pool. In these graphs, for mere adjustment, the values of k_{eff} calculated by MCNPX 2.6.0 code have been normalized for the criticality value of the maximum loading for each situation. In other words, to fresh bundles, the normalization has been made through the respective values of k_{eff} of the cell A4. To mixed loadings, the values of k_{eff} have been normalized for the values of k_{eff} of the cell C8. This procedure only allows that the criticality values calculated in this research to be evaluated on an approximate numerical scale. It does not mean that assemblies inside storage pool to be in critical or supercritical conditions.

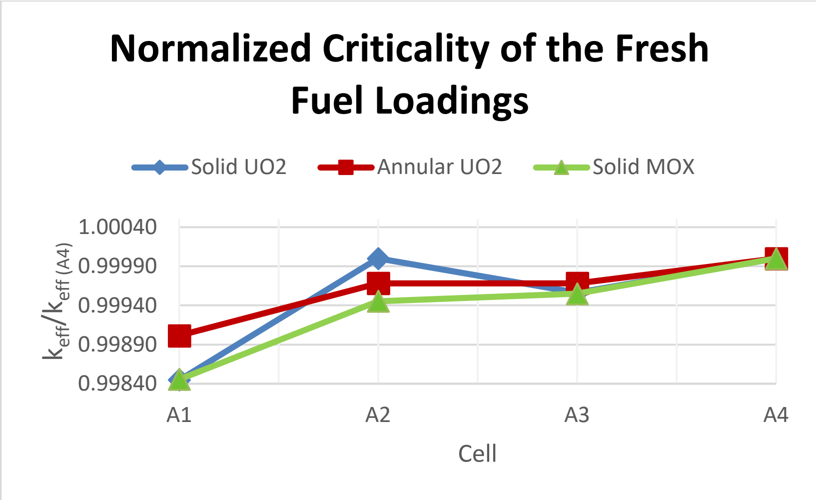

Fig. 4. The normalized criticality ($k_{eff}/k_{eff(A4)}$) of the fresh fuel loadings.

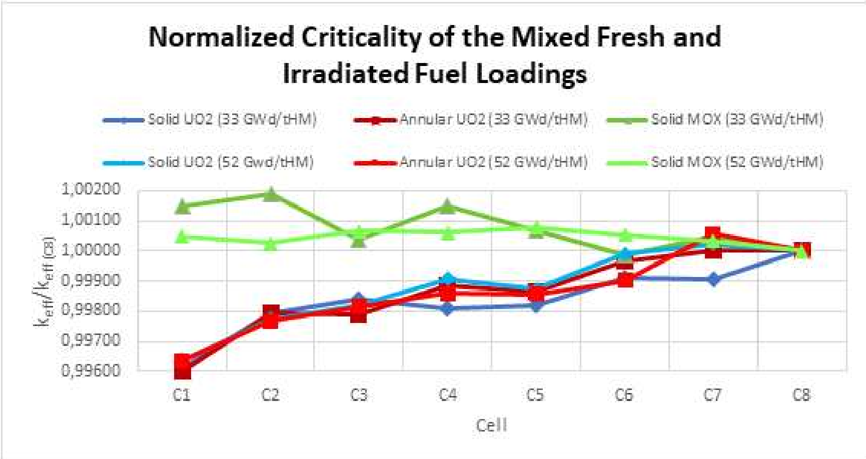

Fig. 5. The normalized criticality ($k_{eff}/k_{eff(C8)}$) of the mixed fresh and irradiated fuel loadings.

Considering the respective errors, a tendency to increase the criticality of the storage pool with the filling of the cells by the fuel elements has been observed. This behavior is clearly seen in Figs. 4 and 5, especially when the deposition of the fresh fuel elements is considered. The exception has occurred for some cases of assemblies filled with irradiated MOX, mainly for burnup at 33 GWd/tHM. Probably, this behavior can be explained by a lower availability of neutrons due to less intense burnup combined with an increase in the concentration of absorbing isotopes, especially fission products and heavy nuclei, which become more pronounced as the loadings of irradiated MOX bundles become larger [14].

According to the results of this research, it has been concluded that the fuel elements filled with UO_2 have very similar behavior when deposited in the storage pool, even for the annular geometry rods, which has 29.4% less fuel stored when compared to the rods of solid geometry. The studied cases have shown that the assemblies of rods of solid geometry have higher criticality than assemblies of annular geometry rods. The exception to this behavior has been observed for the loadings of the bundles of annular geometry irradiated at 52 GWd/tHM, since that criticality for this geometry is higher than the criticality of the solid geometry. Possibly, this has happened due to maximization of the buildup of ^{239}Pu by capture reactions in ^{238}U into the spent fuel irradiated at a higher burnup [14]. The literature has reported that the capture effects of ^{238}U are more intense on the contact surface between fuel and coolant [15]. For assemblies of annular geometry this contact is double in virtue of internal and external surfaces of the fuel rods, therefore, it is reasonable to have a higher criticality for this geometry.

Observing the criticality values obtained for the loadings of bundles filled with irradiated MOX inside the storage pool, it is possible to say that their behaviors have been expected. Firstly, Tab. 4 has shown that the criticalities of the MOX systems are higher than the criticalities that have been achieved for the other sets. This is obvious, since in the criticality calculations of nuclear systems, the neutron absorption cross section is extremely important. For plutonium isotopes these parameters are almost twice those of the uranium isotopes in thermal energy range [13]. This characteristic corroborates the results of this research. In the second place is important the concerns about the burnups for the MOX systems. Fig. 5 shows that at 52 GWd/tHM the calculated criticality is higher than at 33 GWd/tHM. Such behavior should also be expected, since, at higher burnups, there is a greater availability of neutrons, which contributes to a higher criticality [14]. The softer slope of the criticality curve for burnup at 52 GWd/tHM confirms this conclusion.

3.3. Applicability of the Results in the Storage of Fuel Material

The CNEN (Comissão Nacional de Energia Nuclear) is a Brazilian Federal Authority that determines norms and rules about the National Policy on Nuclear Energy including the legislation that establishes the conditions for storage of fuel material, irradiated and non-irradiated, under normal operating condition or in cases of accidents. According to the CNEN-NE-5.02 norm, section 6.2.2, the value of k_{eff} for non-irradiated fuel material must be less than 0.95 under normal operating condition and less than 0.98 in case of an accident. Furthermore, in the section 6.3.3, the norm requires that the value of k_{eff} must be less than 0.95 in both cases, operation at normal condition or in case of an accident, when dealing with irradiated fuel material [2]. These limits agree with internationally accepted references [16].

Analyzing the criticality values (k_{eff}) simulated by MCNPX 2.6.0 code for this research, it has been found that all calculated values are according with the limits required by Brazilian legislation for storage of nuclear fuel elements, for both irradiated and non-irradiated material, under normal operating conditions or in case of an accident. All results that have been found are below to the limits required by the norms. It means that, according to this research, the storage pool studied model admits a maximum fill of its volume without exceeding the criticality safety limits imposed by Brazilian rules.

A relevant aspect, but which is not the focus of the present work, is the influence of the temperature on the criticality of nuclear fuels. As is known, an increase in the temperature of the nuclear fuel implies an increase in the neutron absorption, and consequently reduction in the reactivity by the Doppler effect. The insertion of the negative-feedback by Doppler effect in the reactivity is very important about the criticality safety for nuclear reactors [13]. In this study, the spent fuel pool and the stored fuel bundles are at room temperature and not at operating temperature of a PWR nuclear reactor (moderator/coolant ≈ 600 K and nuclear fuel rod center > 1500 K) [1]. The storage of nuclear fuel into spent fuel pools differs strongly from the conditions found at operating reactor core. In the spent fuel pools, there is no decrease in the reactivity due to Doppler effect. However, no concerns are necessary about the results of this paper, the lack of the Doppler effect has not acted on them. As said before, all the values of this research are according to the norms.

4. CONCLUSION

After all considerations about this study, it has been concluded that the results obtained by this research demonstrate that the investigated nuclear storage pool model fulfills the requirements of Brazilian rules for storage of nuclear material under all conditions that has been stipulated by the authorities from Brazil and international agencies. Furthermore, other assessments can corroborate this statement and enrich the results found in this study. For instance, an investigation about the effects of the temperature on the criticality of the spent fuel pool.

ACKNOWLEDGMENTS

The authors are grateful to the Brazilian research funding agencies, CNEN (Comissão Nacional de Energia Nuclear), CNPq (Conselho Nacional de Desenvolvimento Científico e Tecnológico), CAPES (Coordenação de Aperfeiçoamento de Pessoal de Nível Superior), FAPEMIG (Fundação de Amparo à Pesquisa do Estado de Minas Gerais) and FUNEC (Fundação de Ensino de Contagem), for the support.

REFERENCES

[1] IAEA – International Atomic Energy Agency, Storage of Water Reactor Spent Fuel in Water Pools: Survey of World Experience, Technical Reports Series nº 218 (1982).

[2] CNEN – Comissão Nacional de Energia Nuclear, NE-5.02 Transporte, Recebimento, Armazenagem e Manuseio de Elementos Combustiveis de Usinas Nucleoelétricas, CNEN/PR 08/03, Rio de Janeiro, Brasil (2003).

[3] Westinghouse Electric Corporation, The Westinghouse Pressurized Water Reactor Nuclear Power Plant, pp. 14-26 (1984).

[4] M. S. Kazimi *et al.*, High Performance Fuel Design for Next Generation PWRs: Final Report, MIT-NFC-PR-082, pp.127-128 (2006).

[5] USNRC – United States National Regulatory Commission, SCALE/TRITON Primer: A Primer for Light Water Reactor Lattice Physics Calculations, NUREG/CR-7041, pp. 1-7 (2012).

[6] Z. Xu, Design Strategies for Optimizing High Burnup Fuel in Pressurized Water Reactors, thesis, Department of Nuclear Engineering, Massachusetts Institute of Technology, US (2003).

[7] EPRI - Electric Power Research Institute, Optimum Discharge Burnup for Nuclear Fuel: A Comprehensive Study of Duke Power's Reactors, TR-112571 (1999).

[8] J. R. Secker *et al*., Optimum Discharge Burnup and Cycle Length for PWRs, *Nuclear Technology*, Vol. 151:2, pp. 109-119 (2005).

[9] D. B. Pelowitz, MCNPX 2.6.0: Monte Carlo All-Particle Transport Code System and MCNPDATA, LA-CP-07-1473 (2008).

[10] C. Ye *et al.*, The Design and Simulation of a New Spent Fuel Pool Passive Cooling System, *Annals of Nuclear Energy*, Vol. 58, pp. 124-131 (2013).

[11] Y. Zhang, Transmutation of Transuranic Elements in Advanced MOX and IMF Fuel Assemblies Utilizing Multi-Recycling Strategies, thesis, Office of Graduate Studies, Texas A&M University, US (2009).

[12] M. Salam and C. J. Hah, Comparative Study on Nuclear Characteristics of APR1400 Between 100% MOX Core and UO_2 Core, *Annals of Nuclear Energy*, Vol. 119, pp. 374-381 (2018).

[13] W. O. Stacey, Nuclear Reactor Physics, New York, Wiley-VCH, 2nd ed., pp. 223-224 and 249 (2007).

[14] A. Sasahara *et al.*, Neutron and Gamma Ray Source Evaluation of LWR High Burn-up UO_2 and MOX Spent Fuels, *Journal of Nuclear Science and Technology*, Vol. 41:4, pp. 448-456 (2004).

[15] Y. Yuan, The Design of High-Power Density Annular Fuel for LWRs, thesis, Department of Nuclear Engineering, Massachusetts Institute of Technology, US (2004).

[16] IAEA – International Atomic Energy Agency, Storage of Spent Fuel from Power Reactors, *Proceedings of a Symposium held in Vienna*, *9-13 November 1998*, p. 471 (1999).