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Estratégias de gerenciamento de reatividade para um reator nuclear de água pressurizada integral.

Reactivity Management Strategies for an Integral Pressurized Water Nuclear Reactor.

L. Y. Rojas Mazaira[1], M. Cecilia Betancourt[2], C. R. García Hernández[3], J. A. Rosales García[4], C. A. Brayner de Oliveira Lira[5]

[1] Doutor em Tecnologias Energéticas e Nucleares. Centro Regional de Ciências Nucleares do Nordeste.

E-mail: leored1984@gmail.com

[2] Mestre em Engenharia em Instalações Energéticas e Nucleares. Universidad de la Habana e Universidade Federal de Pernambuco.

[3] Doutor em Ciências Técnicas. Universidad de la Habana.

[4] Doutor em Tecnologias Energéticas e Nucleares. Universidad de la Habana.

[5] Doutor em Engenharia Nuclear. Centro Regional de Ciências Nucleares do Nordeste.

RESUMO

Nos últimos 15 anos, tem havido um crescente interesse nos reatores nucleares modulares pequenos (SMR) e suas aplicações. Os SMRs oferecem vários benefícios com destaque para as características de segurança intrínsecas. Reatores SMRs estão sendo desenvolvidos para as principais tecnologias de reatores nucleares, entretanto, por suas características de segurança excepcionais um papel importante será desempenhado pelos chamados reatores modulares de água pressurizada integrais (iPWR). Os iPWR foram propostos para atingir alta resistência à proliferação de armas nucleares e utilizar ciclos de combustível estendidos. A obtenção de um ciclo com longo comprimento temporal, usando baixo enriquecimento de combustível, sem necessidade de embaralhamento dos elementos combustíveis e com um núcleo de tamanho pequeno é um grande desafio para o projeto neutrônico do núcleo do reator. Neste trabalho, foi desenvolvido um modelo computacional neutrônico para o núcleo de um reator iPWR baseado no código Serpent, que permitiu simular e avaliar o desempenho de diferentes configurações do núcleo. Múltiplas alternativas de absorventes queimáveis integrais em um ciclo de combustível de 48 meses foram estudadas, e o peso da reatividade dos grupos de barras de controle foi estimado. Um ciclo de combustível estendido de mais de 1400 dias é alcançado, e uma penalidade de reatividade mínima foi obtida.

Palavras-chave: SMR, iPWR, absorventes queimáveis, ciclos de combustível

ABSTRACT

In the last 15 years, there has been a growing interest in the small modular nuclear reactors (SMR) and their applications. SMRs offer numerous benefits, including inherent safety features. SMRs are under development for all main reactor lines, but an important role for its outstanding safety features will be played by the so-called integral pressurized water nuclear reactors (iPWR). To achieve high proliferation resistance, extended fuel cycles for iPWR have been proposed. Obtaining such a large cycle length, using low fuel enrichment, without shuffle, and with a relatively small core size is a challenge for the neutronic design of the reactor core. In this work, a full-core neutronic computational model based in Serpent code was developed, it permitted to simulate the performance of the proposed reactor core configurations. Different variants of use of integral burnable absorbents in 48-months fuel cycle were studied, and the reactivity worth of the control rod groups was estimated. A fuel cycle extended of more than 1400 EFPDs is reached, and a minimal reactivity penalty was obtained.

Keywords: SMR, iPWR, burnable absorber, extended fuel cycle

1. INTRODUCTION

In the last 15 years, there has been a growing interest in the small modular nuclear reactors (SMRs) and their applications. SMRs are newer generation nuclear reactors designed to generate electric power up to 300 MW, whose components and systems can be shop fabricated and then transported as modules to the sites for installation as demand arises. SMRs offer numerous benefits, including inherent safety features such as passive heat removal capabilities for decay heat, increased security and proliferation resistance with integrated safeguards, and underground construction to address the threats of sabotage, airplane impact, and some natural hazard scenarios (IAEA, 2018).

SMRs are under development for all main nuclear reactor lines: water-cooled reactors, high-temperature gas-cooled reactors, liquid-metal, sodium, and gas-cooled reactors with fast neutron spectrum, and molten salt reactors. An important role for its outstanding safety features will be played by the so-called integral pressurized water nuclear reactors (iPWR).

The main drivers of SMR development are interested in developing power facilities with flexible power generation for a broader range of users and applications, replacing older fossil-fired units, improving safety performance, and offering better economic affordability.

Currently, there are more than 50 SMR designs under development for different applications. Three industrial demonstration SMRs are in an advanced stage of construction or concluded: in Argentina (CAREM, an iPWR), in the People's Republic of China (HTR-PM, a high-temperature gas-cooled reactor), and in the Russian Federation (KLT40s, a floating power unit). They were scheduled to start operation between 2019 and 2022. KLT40s had his first grid connection on the 19 of December 2019, and the rest remain under construction (PRIS,2020). In addition, the Russian Federation has already manufactured six RITM-200 reactors (an iPWR) with four units already installed in the Sibir and Arktika icebreakers, to be in service in 2020.

At present three designs of iPWR type reactor appear on the website of the Advanced Reactor Information System (ARIS, 2020), the first is IMR (Japan), his full name is Integrated Modular Water Reactor of Mitsubishi, is under design, and it has a commercial purpose. Second, Nuscale (USA), his full name is NuScale Power Modular and Scalable Reactor of NuScale Power Inc.; it is under design and has a commercial purpose too. Third and last SMART (Republic of Korea), his full name is System-Integrated Modular Advanced Reactor of Kaeri, is licensed and has a commercial purpose too.

In (IAEA, 2018) was published a summary of the main design features and status of SMRs included in the category water-cooled small modular nuclear reactors land-based. Among the projects described are some that have been extensively studied and have unfortunately been archived, although they remain very promising.

Between the most important advanced iPWR projects are IRIS (IAEA, 2018), of IRIS Consortium, mPower (Erighin, 2012) of BWX Technologies, SMR-160 of Holtec International, and Westinghouse SMR of Westinghouse Electric Company LLC, USA (IAEA, 2018). Table 1 summarizes some interesting features of the most important previous advanced square lattice iPWR projects.

All of them have some common features such as the use of UO_2 fuel with less than 5 percent enrichment in ^{235}U, reduced core height, square lattice with 17 X 17 fuel elements.

The thermal power produced ranges from 160 to 800 MWt. Although, in (IAEA, 2018), the mPower project is shown with a 24-month fuel cycle, in (Erighin, 2012) a 48-month extended fuel cycle for the B&W mPower™ small modular reactor was proposed. This option is a once-through fuel cycle in which the entire core is discharged and replaced after four years. There is no refueling and no shuffle of fuel assemblies during the cycle, and this feature prioritizes proliferation resistance.

Obtaining such a large cycle length, using low fuel enrichment, without shuffle, and with a relatively small core size is a challenge for the neutronic design of the nuclear reactor core, despite the low power density near to 65 kW/liter.

The mPower nuclear reactor core concept does not use soluble boron for reactivity control during normal operation, SMR-160 also don't use it. Not using chemical shim produces a large negative value of the moderator temperature reactivity coefficient, mainly at the beginning of the cycle (BOC), which provides a robust inherent reactivity control mechanism, like a boiling water nuclear reactor (BWR) (Erighin, 2012).

However, controlling the large excess reactivity of the core using only burnable absorbers and control rods is another major challenge for core design. Several studies about mPower fuel cycle design and reactivity management strategies have been published (Kitcher and Chirayath, 2016; Erighin, 2012 and Rosales et al, 2018), in these papers the neutronic analysis of burnup depletion were made using models that simulate the fuel assembly lattice.

In this paper, the results of a full-core neutronic simulation based in Serpent code for an extended fuel cycle of an iPWR are presented. Different variants of use of integral burnable absorbents were analyzed for a 48-month extended fuel cycle. Hot excess reactivity, not compensated by burnable absorbers, shall be suppressed by at least two mutually exclusive control rod patterns, the worth of control rod patterns was determined. The goal of this work is to design the reactor core to achieve a fuel cycle time of 48 months using the fewest number of inserted control rods.

Table 1. Major technical parameters of most important iPWR project.

Major technical parameters	M Power	Nu-Scale	SMR-160	Westinghouse SMR
Thermal/ electrical capacity, MW(t)/MWe	575/195	160/50	525 / 160	800/>225
Fuel type /assembly array	UO_2 pellet /17x17/ square	UO_2 pellet /17x17/ square	UO_2 pellet / square array	UO_2 pellet /17x17/ square
Number of fuel assemblies	69	37	112	89
Fuel cycle (months)	24	24	18–24 (flexible)	24
Main reactivity control mechanism	Control rods	Control rods, boron	control rods	Control rods, boron

An outline of the remainder of this paper follows. In the next section, the analyzed reactor core parameters and characteristics of the previous designs are described. In Section 3, several main characteristics of Serpent code, used to develop the full-core neutronic model for an extended fuel cycle of an iPWR are presented. The fuel cycle design and several reactivity management strategies using burnable absorbers are described in Section 4. Numerical results are presented and discussed in Section 5, and Section 6 gives a number of concluding remarks.

2. REACTOR CORE PARAMETERS

The analyzed reactor core is made up of 69 fuel assemblies, loaded in a square lattice of 21.5 cm pitch. The core has an octant mirror-symmetric configuration, and it is designed to produce the energy required for a 4-year once-through fuel cycle. No refueling and no shuffle of fuel assemblies during the cycle are permitted.

In (Kitcher and Chirayath, 2016), a simple calculation was made to determine the mass of uranium metallic necessary to reach a fuel burnup near to 40 GWd/ THM and thereby guarantee a fuel cycle time of 48 months. An optimization study of the reactor core dimensions was also performed. The thermal power and the active fuel length were taken of data reported in (IAEA, 2016) because they match better with a 48 months fuel cycle. In Table 2 are illustrated the main parameters of iPWR nuclear core analyzed.

Table 2. Main characteristics of the analyzed reactor

Parameter	Value
thermal power	530 MW
number of fuel assemblies	69
core equivalent diameter	200 cm
fuel active length	240 cm
assembly lattice pitch	21.5 cm
fuel type /assembly array	UO_2 pellet / 17x17 square
Fuel rods per assembly	264
fuel enrichment	< 5%
average fuel burnup	40 GWd/ THM

VERA Core Physics Benchmark Progression Problems (Godfrey, 2014) is a valuable document where detailed information on the geometry and materials used in the PWR cores is reported. In this work several benchmark problems were selected to assist developers and analysts of nuclear engineering in the development of calculation methods and tools to describe the behavior of the nuclear reactor cores in normal operation and transient states. The data published in (Godfrey, 2014) are based on actual fuel data from plants with nuclear reactors of the PWR type, and all are public and current. For our simulations, geometric and material data were taken of Westinghouse-designed 17x17 fuel assembly reported here.

The core is formed by a 17x17 fuel rod geometry within the 69 fuel assemblies. It contains an axially uniform UO_2 fuel stack contained within Zircaloy-4 cladding. Each 17x17 assembly contains 24 guide tubes serving as structure and as a location for discrete inserts such as control rod clusters. There is also one instrument tube at the lattice center for the insertion of an in-core neutron flux detector. Each of these tubes is made of Zircaloy-4.

Figure 1 shows the core and assembly layout and core fuel enrichment distribution used in this study, five assemblies at 4.0 wt % ^{235}U enrichment, and 64 assemblies at 4.95 wt % ^{235}U enrichment (Rosales et al, 2018). Table 3 provides the detailed specification of the fuel rod and 17x17 lattice.

Table 3: Fuel Rod and 17x17 Lattice Specification.

Pellet Radius	0.4096 cm
Inner Clad Radius	0.418 cm
Outer Clad Radius	0.475 cm
fuel rod lattice pitch	1.26 cm
Pellet Material	UO_2
Clad / Plugs Material	Zircaloy-4
Inner Guide Tube Radius	0.561 cm
Outer Guide Tube Radius	0.602 cm
Inner Instrument Tube Radius	0.559 cm
Outer Instrument Tube Radius	0.605 cm
Tube Materials	Zircaloy-4
Inter-Assembly Half Gap	0.04 cm
Fill Gas Material	Helium

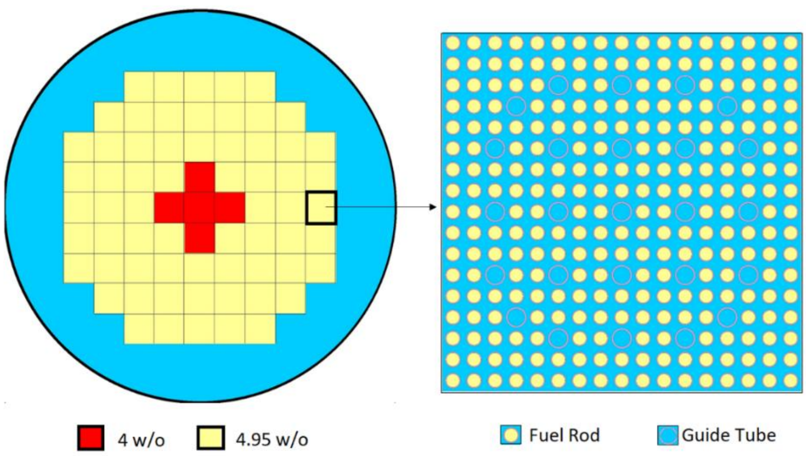

Fig. 1. Core and assembly layout.

3. CORE NEUTRONIC SIMULATION USING SERPENT CODE

Serpent is a multi-purpose three-dimensional continuous-energy Monte Carlo particle transport code, developed at VTT Technical Research Center of Finland, Ltd. The development started in 2004, and the code has been publicly distributed by the OECD/NEA Data Bank and RSICC since 2009. Serpent started as a simplified reactor physics code, but the capabilities of the current development version, Serpent 2, extend well beyond nuclear reactor modeling (Leppanen J., 2015). During its sixteen years of life, the Serpent code has been remarkably developed; it has evolved from a simplified Monte

Carlo lattice transport code into a versatile nuclear reactor physics calculation tool. The code can solve multiple problems from group constant generation to fuel cycle analysis, nuclear reactor modeling and coupled multi-physics calculations.

SERPENT code uses a universe-based combinatorial solid geometry, which allows the description of practically any two or three-dimensional fuel or reactor configuration. The geometry consists of material cells, defined by elementary quadratic and derived macro-body surface types as spheres and cylinders.

The burnup capability in SERPENT is based on its own routines, without any external coupling. The number of depletion zones is not restricted, although memory resources become a limiting factor. No additional user effort is necessary to select the fission and activation products and actinide daughter nuclides for depletion calculations. The burnable materials can be sub-divided into depletion zones automatically. The irradiation history is defined in units of time or burnup.

The reaction rates can be normalized to total power, specific power density, flux, fission, or source rate, and the normalization can be changed by dividing the irradiation cycle into a number of separate depletion intervals. We calculated the power peak factors normalized to total power.

ACE format cross-section libraries based on JEF-2.2, JEFF-3.1, JEFF-3.1.1, ENDF/B-VI.8, and ENDFB/B-VII evaluated data files are included. Interaction data is available for 432 nuclides at six temperatures between 300 and 1800K. The nuclear data library used for the cross-section of the materials in these calculations was JEFF-3.1. Detailed information on the possibilities of the Serpent code can be found at (Leppanen J., 2019).

For the simulations, it was used Serpent 2.1.29 running in parallel with four processors. A PC with an Intel(R) Core (TM) I7-3770 CPU, 3.40 GHz with 16 GB of memory and Linux Mint 19.3 were used. A full-core neutronic computational model based in Serpent code was developed. The Serpent code has great advantages for carrying out the neutronic calculations proposed in this study, because in addition to allowing the simulation of the whole nuclear reactor core with relative ease, it also performs accurate depletion calculations considering a large number of isotopes and material zones.

4. FUEL CYCLE DESING AND HOT REACTIVITY MANAGEMENT STRATEGIES

The fuel enrichment values and the loading pattern must produce sufficient reactivity for the cycle to achieve the planned energy. The proposed core loading pattern reaches an effective multiplication factor Keff value for BOC of 1.33804. The BOC state is considered at nominal power, hot core, and Xenon equilibrium concentration.

The large value of initial excess reactivity (25264 pcm) causes the need to introduce a large amount of absorbent material to balance the excess reactivity. As the proposed reactor design concept does not use a chemical shim, suitable handling of the burnable absorbents guarantees the smallest possible amount of control rods introduced into the reactor core during the fuel cycle. Two hot reactivity management strategies using burnable absorbents in the analyzed reactor core have been studied. First, using the integral fuel burnable absorbers (IFBA) and second, the gadolinia integral burnable absorbers.

4.1 REACTIVITY CONTROL WITH INTEGRAL FUEL BURNABLE ABSORBERS

The use of integral burnable absorbers type IFBA makes it possible to obtain an optimized reactor core hot reactivity control and flattened core power distributions during the fuel cycle. The IFBA technique is based on the use of a very thin ZrB_2 coating on fuel rods that were previously selected in the fuel assembly. The ^{10}B contained in the coating layer does not displace fuel material and depletes rapidly, for this reason there will be no residual reactivity penalty (Godfrey,2014).

In (Sanders and Wagner, 2002) the effect of integral fuel burnable absorber on reactivity for various designs was discussed. In that work, the fuel assembly performance was simulated. Several combinations of distributions of IFBA in the assembly and fuel enrichment values were studied.

In (Godfrey, 2014) several distributions of IFBA in the fuel assembly were proposed, and considerations about IFBA data were made, that were used by us.

In this paper, 104, 128, and 156 IFBAs rod by assembly were simulated in the loading pattern. The assembly layouts for the considered configurations are shown in Fig. 2. In Table 4 the IFBA fuel rod specification, used in the fuel burnup calculations is described. Other than the ZrB_2 coating, the remaining geometric data of the fuel rod appear in Table 3.

Table 4. IFBA Fuel Rod Specification.

Poison Material	ZrB_2
^{10}B Loading	0.927 mg/cm
^{10}B Enrichment	50 %
Coating Density	3.85 g/cm^3
Poison Height	240 cm
Poison Location	Centered axially

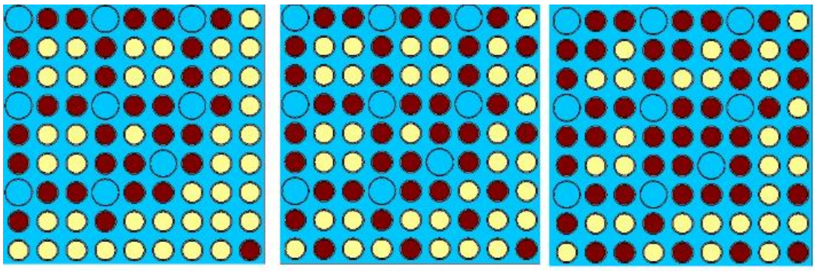

Fig. 2. 104,128 and 156 IFBA rod configurations, square symmetry.

4.2 REACTIVITY CONTROL WITH GADOLINIA INTEGRAL BURNABLE ABSORBER

The gadolinium oxide, called gadolinia too, has been used successfully for reactor core reactivity control in light water nuclear reactors (LWR). Gadolinia is mixed homogeneously within the UO_2 fuel pellets for a few select rods in the assembly in predetermined concentrations, usually ranging from 2-8% by weight (Godfrey, 2014).

In the natural gadolinium composition, there are two isotopes with very large values of neutronic thermal absorption cross-section. They are ^{155}Gd with natural abundance equals 15% and ^{157}Gd with 16%. The thermal absorption cross-sections are $6.1x10^4$ barns and $2.5x10^5$ barns, respectively. Gadolinium isotopes do not contribute significantly to the radiotoxicity of spent nuclear fuel. The melting temperature of Gd_2O_3 is 2350 °C (Galahom, 2016).

Uranium-gadolinium fuel rod has lower heat conductivity than the non-poison fuel rod, so the ^{235}U enrichment is reduced to meet the design criterion regarding maximum fuel temperature. That's why fuel rods containing gadolinia are usually lower enriched in ^{235}U than non-poison rods in the same assembly. The above has an economic basis and to ensure sufficient safety margins.

However, decreasing the enrichment of the non-poison fuel rods leads to a lower initial mass of fissile fuel, which conspires against the duration of the desired extended fuel cycle. Because the proposed core loading pattern has a fuel enrichment near to 5 wt %, we considered using a value of 3 wt % of ^{235}U for all fuel rods containing gadolinia in the different variants analyzed.

In (Cudrnak and Necas, 2011), several burnup variants of the 17x17 standard PWR assembly were studied, using ^{235}U enrichment values at the gadolinium-poison rod in the range of 2.6 to 3.2 wt %. In Japan, the maximum fuel assembly burnup is up to 55 GWd/THM, which corresponds to 17 x 17 fuel assembly with an average ^{235}U enrichment of 4.8 wt %, containing 24 gadolinia fuel rods with an enrichment of 3.2 wt % of ^{235}U. In Table 5 the gadolinia fuel rod specification, used in the fuel burnup calculations is described.

Table 5. Gadolinia Fuel Rod Specification.

Poison Material	Gd_2O_3
Gadolinia Concentration	2-8 wt %
Fuel Density	10.111 g/cm^3

In this work, several loading patterns using gadolinia burnable absorbers were analyzed. First, the reactor full core burnup calculations were made considering all fuel assemblies using 32 poison rods with gadolinia enrichment of 2, 4, and 6 wt %, cases 1, 2, and 3, respectively. The assembly layout with 32 gadolinia rods was taken off (Sanders and Wagner, 2002) and is presented in Fig. 3.

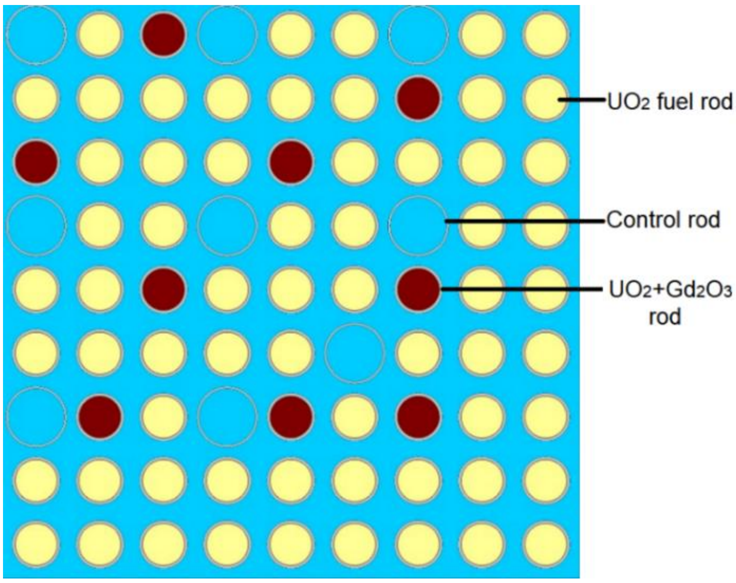

Fig. 3. Assembly layout with 32 UO_2+Gd_2O_3 rods, square symmetry.

Table 6. Fuel assembly data for the several core designs using gadolinia poison rods.

cases	No. of assemblies - UO_2 fuel enrichment wt **%**	No. of UO_2 fuel rods/ assembly	No. of UO_2-Gd_2O_3 rods/ assembly	Gd_2O_3/^{235}U wt **%** for UO_2- Gd_2O_3 rods
1	64–4.95 % 5-4 %	232	32	2 % / 3 %
2	64–4.95 % 5-4 %	232	32	4 % / 3 %
3	64–4.95 % 5-4 %	232	32	6 % / 3 %
4	64–4.95 % 5-4 %	236	28	12- 4 % / 3% 16- 8 % / 3%
5	64–4.95 % 5-4 %	236	28	12- 6 % / 3 % 16- 8 % / 3 %

4.3 CONTROL RODS REACTIVITY MANAGEMENT

In normal reactor operation, the critical state is reached with the number necessary of control rods inserted to balance excess reactivity. In analyzed core design, no soluble boron to suppress excess reactivity is used. This means that the excess of reactivity not balanced by burnable absorbents must be assumed by the inserted control rods. That is why the number of assemblies with control rods is higher than in standard PWR of the same size and similar to a boiling water nuclear reactor (BWR).

Hot excess reactivity (HER) is the amount of reactivity that must be balanced during the fuel cycle at normal operating conditions using control rods. The larger HER, the greater the control rod number needed to maintain the reactor at the full power critical state. Too much HER is disadvantageous because it would require a large number of control rod inserted to control the reactivity. Too little HER is also undesirable, since it may leave the core with limited capability to overcome xenon transient states.

Two main control rod patterns were formed, like in BWR reactor type. The rod control assemblies (RCA) were grouped in A and B, distributed in a checkerboard pattern, as shown in Fig. 5 (Erighin, 2012). Several sequences of movement of control rods can be used throughout the cycle. In order to estimate if it is possible to balance the maximum values of HER during the fuel cycle with a different strategy of use of burnable absorbers, the reactivity worth of the RCAs groups at the BOC was calculated. In addition to groups the A and B, a third group formed by nine assemblies belonging to group A, and located at positions (C,3), (C,5), (C,7), (E,3), (E,5), (E,7), (G,3), (G,5) and (G,7) was considered and called A1.

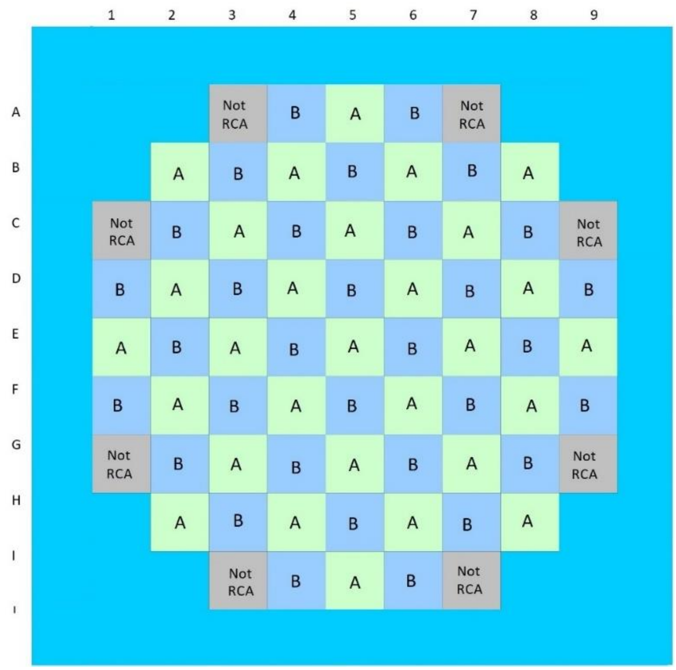

Fig. 5. The two main rod control assemblies' sequences used at the analyzed nuclear reactor core.

Although in the mPower design was proposed the Ag-In-Cd as absorber material in the rod control assemblies (IAEA, 2018), here the B_4C was considered too. In Table 7 the RCA rod specifications for the Ag-In-Cd and B_4C absorbers are described. B_4C data was taken from (Godfrey, 2014). The ^{10}B enrichment is 20 %.

Table 7. RCA rod specifications for the Ag-In-Cd and B_4C absorbers

Parameters	Ag/In/Cd	B_4C
Composition	80/15/5%	100%
Poison Density	10.2 g/cm^3	1.76 g/cm^3
Poison Radius	0.382 cm	0.373 cm
Poison Height	240 cm^3	240 cm^3
Cladding Material	SS304	SS304
Cladding Inner Radius	0.386 cm	0.386 cm
Cladding Outer Radius	0.484 cm	0.484 cm
Plenum Material	Helium	Helium

5. RESULTS AND DISCUSSION

The different number of IFBA rods had been investigated to select the suitable number of the IFBA rods that can be used in the iPWR core design, which guarantees a minimum excess of reactivity during the fuel cycle and the desired extended length. Figure 6 illustrates the variation of Keff values with Effective Full Power per Days (EFPD) at different numbers of IFBA rods.

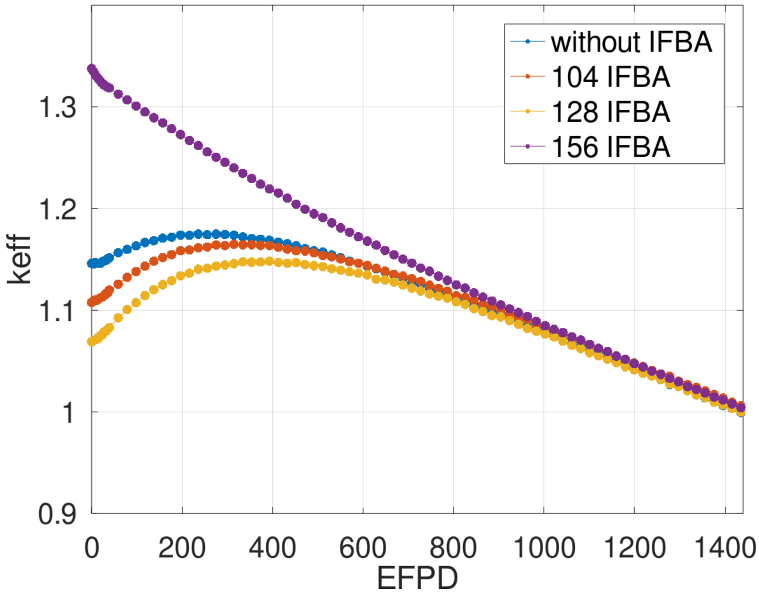

Fig. 6. Variation of Keff values with EFPD at different numbers of IFBA rods.

In all cases, a fuel cycle extended of more than 1400 EFPDs is reached. A very small reactivity penalty at the end of life is obtained. The absorbent isotope ^{10}B is practically totally burned at the end of the fuel cycle. Figure 7 shows the variation of ^{10}B mass throughout its lifetime in the reactor core. The HER value at BOC with 156 IFBA rods is 6 860 pcm, but the maximum value, reached between 300 and 500 days is near to 14 800 pcm.

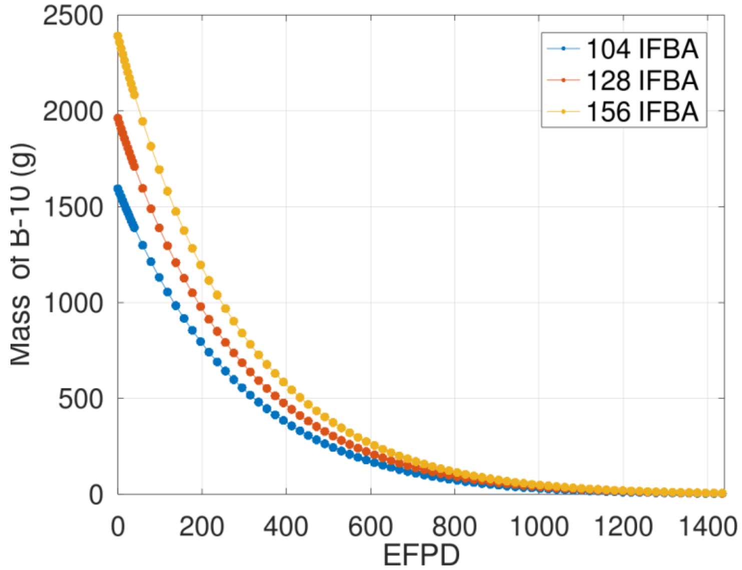

Fig. 7. Variation of ^{10}B mass versus burnup at different numbers of IFBA rods.

In Fig. 8 the core power peak factors at BOC, without and with different numbers of IFBA rods are shown. The BOC is considered at hot and ^{135}Xe equilibrium state, without control rods inserted. The core power peak factors are calculated as the actual assembly power between the core average assembly power, in the same form are calculated the pin power peak factors. The eigenvalue calculations, where tallies were used to estimate the energy released in the different zones and calculate the peak factors, were made with 100 000 histories and 500 cycles. Relative errors of less than 5% were obtained in all the energy tallies.

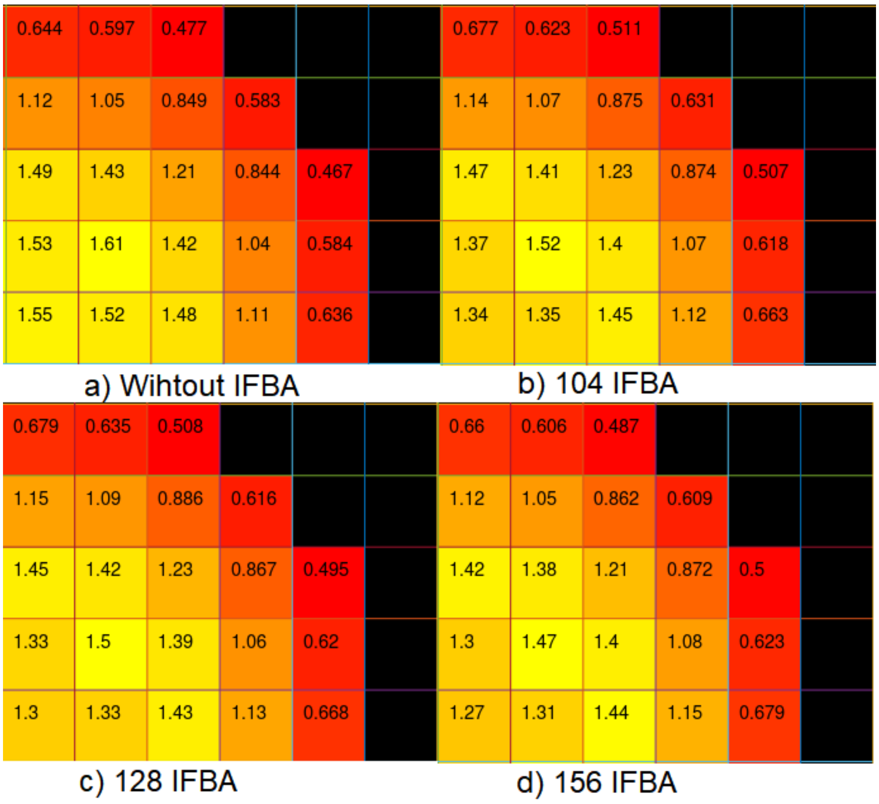

Fig. 8. Core power peak factors at BOC without and with 104,128 and 156 IFBA rods, square symmetry.

As can see the greatest value of power peak factor at loading pattern without IFBA rods is 1.61, in the rest of the cases, it is always less. The use of IFBA rods no perturbs the core power distribution. The pin power peak factor distributions with different numbers of IFBA rods no are shown, but the maximum value in the 156 IFBA case is 1.77, which ensures sufficient safety margins.

Different gadolinia configurations had been investigated to select a suitable configuration that can be used in the iPWR core design with an extended fuel cycle. That is the number of UO_2+Gd_2O_3 rods by assembly and concentration of the absorbent material. Figure 9 describes the Keff performance versus fuel cycle length for five cases analyzed.

In the cases with 32 gadolinia rods by assembly, the minimum values of HER needed to balance are reached with 6 % of gadolinia concentration. Near of the 600 days is necessary to balance 13000 pcm of reactivity approximately. The reactivity penalty obtained at the end of life in all cases using gadolinia absorber is caused by the decrease in the mass of fissile fuel, and it is less than 1400 pcm, remember that the uranium gadolinia fuel use a lower initial fuel enrichment. The above produces a shorter duration of the fuel cycles. However, the lengths of the cycles continue to be considered of 48 months. In the cases with 28 gadolinia rods by assembly, lower values of excess reactivity need to be compensated.

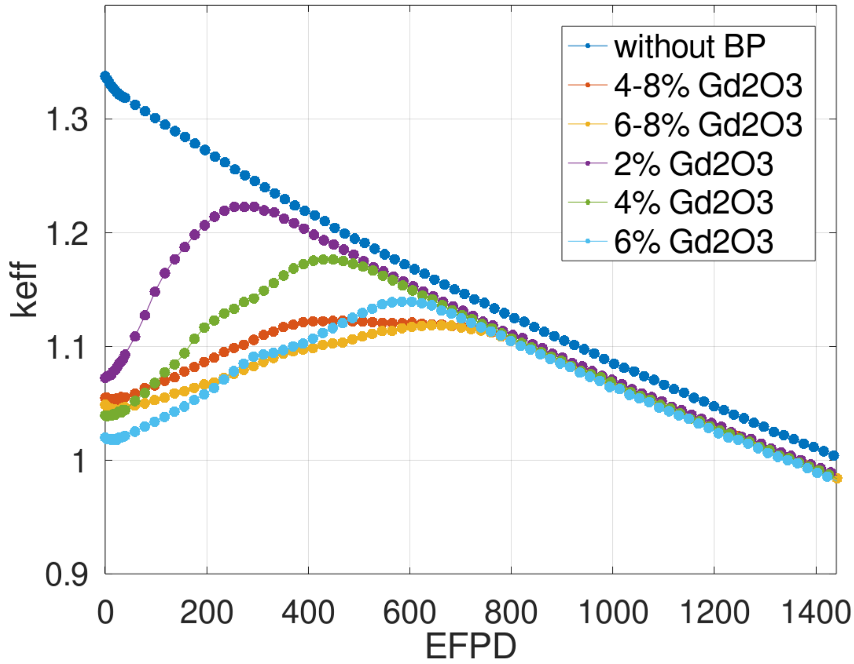

Fig. 9. Variation of Keff with EFPD at different gadolinia concentrations for 28 and 32 UO_2+Gd_2O_3 rods by assembly.

In Figs. 10 and 11, the variation of ^{155}Gd and ^{157}Gd masses with EFPD at different cases are shown. Similar to ^{10}B performance, the ^{155}Gd and ^{157}Gd masses decrease rapidly over time.

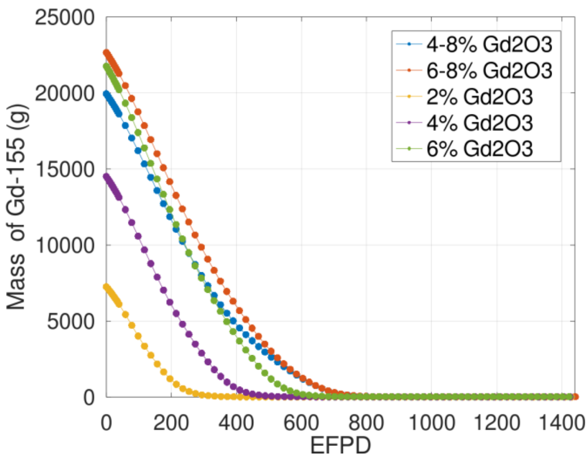

Fig. 10. Variation of ^{155}Gd masses with EFPD in different cases.

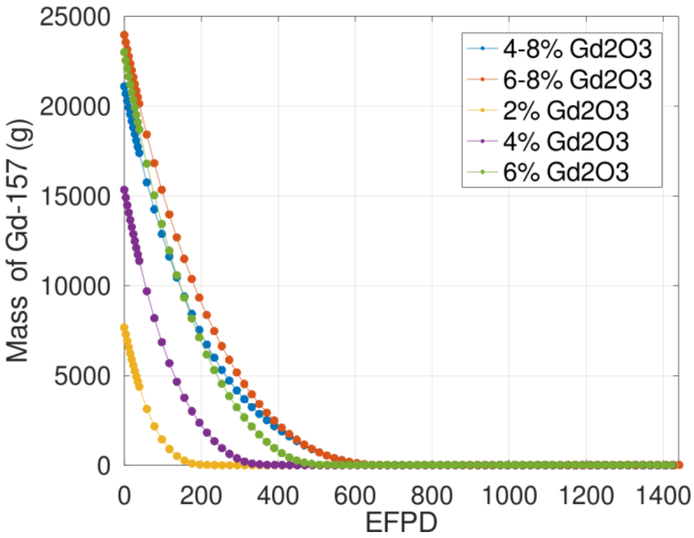

Fig. 11. Variation of ^{157}Gd masses with EFPD in different cases.

The reactivity worth of the different considered groups of control rod assemblies was calculated. Table 8 shows the reactivity worth values for several numbers of control rod assemblies fully inserted into the core at BOC. In this case, the BOC is considered at hot and ^{135}Xe equilibrium state without burnable poison in the core. The Keff value obtained for cold zero power condition is 1.46474 +/- 0.00012, which means that a reactivity value of 12.670 pcm must be compensated to bring the core to the hot full power condition.

In this study, Ag-In-Cd and B_4C absorbers were considered. With the B_4C absorber, higher values of reactivity worth are reached. But only at BOC with the group A+B of rod control assemblies fully inserted the subcritical state is reached.

Table 8. Reactivity worth values for several numbers of control rod assemblies fully inserted into the core at BOC, using Ag-In-Cd and B_4C absorbers.

Number of assemblies with control rods	Keff		Reactivity worth (pcm)	
	Ag-In-Cd	B_4C	Ag-In-Cd	B_4C
0	1.33804	1.33804	----	----
9 (A1)	1.27417	1.26610	6 391	7 194
29 (A)	1.17430	1.14939	16 374	18 865
61 (A+B)	0.99668	0.95096	34 136	38 708

The reactivity worth of group A should be enough to balance the maximum HER during the burnup fuel cycle when 128 or 156 IFBA rods are used. If cases 3, 4, or 5 of gadolinia burnable absorbers are used the control rod group A permits to balance the highest value of HER too. With cases 1 and 2, a greater number of control rods must be inserted during a large section of the fuel cycle.

In Fig. 12 the reactor core power peak factor distributions at BOC without and with 9, 29, and 61 assemblies with 24 control rods fully inserted are shown. The assemblies with control rods inserted have lower power peak factors, while now the peripheral fuel assemblies have higher peak factors. The higher power peak factor with 29 assemblies with control rods fully inserted is 1.46, less than the 1.61 obtained for the non-poisoned core. The flattest core radial power distribution is obtained for the cases with 9 and 29 assemblies with control rods fully inserted.

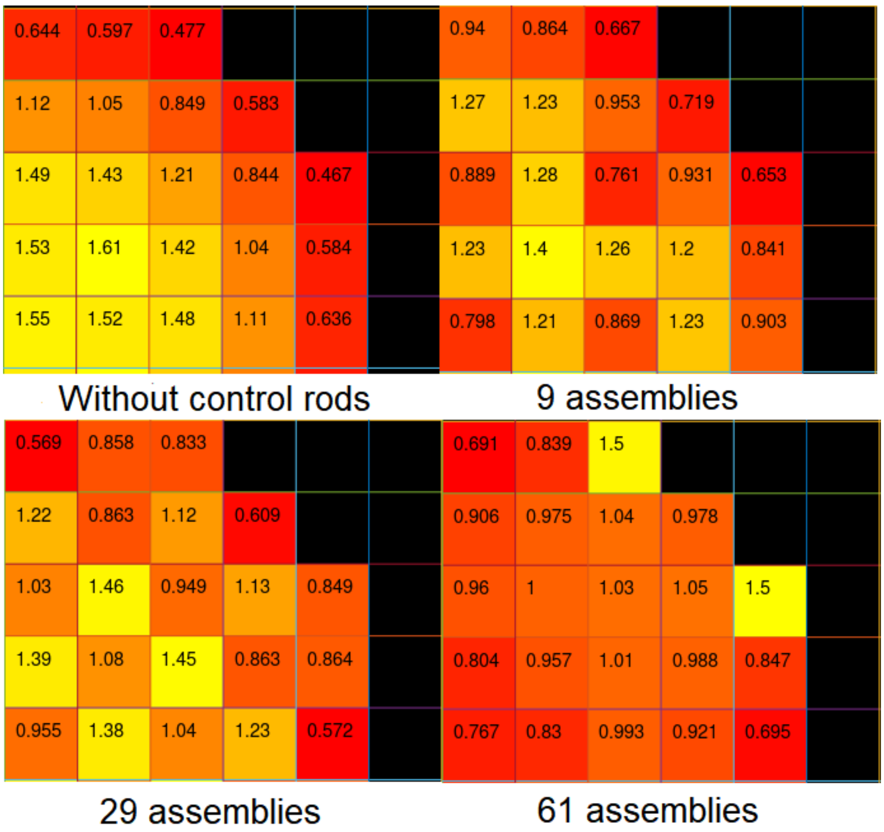

Fig. 12. Core power peak factors at BOC without and with 9, 29 and 61 assemblies with control rods fully inserted into the core, square symmetry.

The analyzed reactor concept design does not use soluble boron for reactivity control during normal operation, which produces a large negative moderator temperature coefficient of reactivity. The moderator temperature reactivity coefficients (MTRC) were calculated at BOC. The coefficient values were calculated using the following equation, Where $Keff^{T1}$ and $Keff^{T2}$ are the Keff values for the temperatures T_1 and T_2, respectively, and they are the temperature values at interval extreme points. The results are presented in Table 9.

$$MTRC = \frac{\rho^{T2} - \rho^{T1}}{T2 - T1} = \frac{\frac{1}{keff^{T1}} - \frac{1}{keff^{T2}}}{T2 - T1}$$

The moderator temperature reactivity coefficient values obtained agree with reported in (Kitcher and Chirayath, 2016) and (Rosales et al, 2018), in these papers was used MCNP code. The high values of MTRC obtained show the large negative moderator temperature reactivity effect in this proposed reactor core design.

Table 9. The moderator temperature reactivity coefficients at BOC.

Temperature (K)	MTRC (pcm/K)	std (pcm/k)
300-400	-4.43716734	± 0.54
400-500	-11.8005001	± 0.55
500-600	-28.712401	± 0.59

6. CONCLUSIONS

In this paper, a full-core neutronics computational model based in Serpent code was developed, in order to design an extended fuel cycle for an iPWR. Several distributions with IFBA rods were investigated to select the suitable number of the IFBA rods in the iPWR core design, that guarantees a minimum excess of hot reactivity during the fuel cycle and the desired extended length.

In all cases, a fuel cycle extended of more than 1400 EFPDs is reached and a very small reactivity penalty at the end of life was obtained. The values of HER obtained during the fuel cycle using 156 IFBA rods can be balance with the group A of RCA. The use of IFBA rods no perturbs the core radial power distribution.

The control hot reactivity strategies using gadolinia were presented in five cases. The reactivity penalty obtained at the end of life in all cases is less than 1400 pcm, and greater than in the case of control hot reactivity strategy with IFBA rods. The above represents a decrease in the fuel cycle length of about 60 days, considering a load factor plant of 0.95 means that match with a 48-month extended fuel cycle. The cases 4 and 5 are the most advantageous because they reach maximum values of HER below 12 000 pcm, that can be balanced with group A of control rods partially inserted in the reactor core.

The reactivity worth of the considered groups of control rods with Ag-In-Cd and B_4C absorbers were calculated. With the B_4C absorber higher values of reactivity worth were reached. The reactivity worth of group A permits to balance the HER during the fuel cycle in most of the control hot reactivity strategies analyzed. In all cases with control rods fully inserted, more flattened core radial power distributions were obtained.

In future works, thermohydraulic analyzes will be carried out to complete the conceptual design of the proposed iPWR reactor core.

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REFERENCES

Anonymous, **Latest news related to PRIS**. *https://pris.iaea.org*. Retrieved in June – 2020.

Anonymous, **Latest news related to ARIS**. *https://aris.iaea.org*. Retrieved in June – 2020.

CUDRNAK, P.; N., VLADIMIR. Optimization of fuel assemblies with gadolinium for LWRs. **Journal of Power Technologies**, v. 91, n. 4, p. 186-190, 2011.

ERIGHIN, Madalina Aimee. A 48-month extended fuel cycle for the B and W mPower TM small modular nuclear reactor. 2012.

GALAHOM, A. Abdelghafar. Investigation of different burnable absorbers effects on the neutronic characteristics of PWR assembly. **Annals of Nuclear Energy**, v. 94, p. 22-31, 2016. *http://dx.doi.org/10.1016/j.anucene.2016.02.025*

GALAHOM, A. Abdelghafar. Simulate the effect of integral burnable absorber on the neutronic characteristics of a PWR assembly. **Nuclear Energy and Technology**, v. 4, p. 287, 2018.) *https://doi.org/10.3897/nucet.4.30379*.

Godfrey, Andrew T. **VERA Core Physics Benchmark Progression Problem Specifications. Revision 4**. CASL-U-2012-0131-004. Oak Ridge National Laboratory. *https://www.casl.gov/sites/default/files/docs/CASL-U-2012-0131-004.pdf*

IAEA. **Advances in small modular reactor technology developments** Supplement to Advanced Reactors Information System (ARIS). 2016 Edition. IAEA. *https://aris.iaea.org/Publications/SMR-Book_2016.pdf*

IAEA. **Advances in small modular reactor technology developments**. Supplement to Advanced Reactors Information System (ARIS). 2018 Edition. IAEA. *https://aris.iaea.org/Publications/SMR-Book_2018.pdf*

KITCHER, E.D.; CHIRAYATH, S. S. Neutronics and thermal hydraulics analysis of a small modular reactor. **Annals of Nuclear Energy,** v. 97, p. 232-245, 2016. *http://dx.doi.org/10.1016/j.anucene.2016.07.019*

LEPPÄNEN, J et al. The Serpent Monte Carlo code: Status, development and applications in 2013. In: **SNA+ MC 2013-Joint International Conference on Supercomputing in Nuclear Applications+ Monte Carlo**. EDP Sciences, 2014. p. 06021.*http://dx.doi.org/10.1016/j.anucene.2014.08.024*

LEPPÄNEN, J. **Serpent – a Continuous-energy Monte Carlo Reactor Physics Burnup Calculation Code.** VTT Technical Research. Centre of Finland. (last update June 3, 2019). Available online at *http://serpent.vtt.fi/mediawiki/index.php/Main*

GARCÍA, J. A. R. et al. New stage on the neutronics and thermal hydraulics analysis of a small modular reactor core**. International Journal of Nuclear Energy Science and Technology**, v. 12, n. 4, p. 400-417, 2018.

SANDERS, C. E.; WAGNER, J. C.**Study of the Effect of Integral Burnable Absorbers for PWR Burnup Credit**. NUREG/CR-6760. Oak Ridge National Laboratory. (2002) *https://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6760/*