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A PWR Fuel Assembly Analysed with SERPENT, KENO VI, MCNPX and NEWT Codes

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1. Introduction

The aim of this work is to analyse PWR fuel assemblies loaded with standard enriched UO_2 and reprocessed fuel spiked in thorium (TRU) using several nuclear computational codes and neutronic libraries. Then compare the obtained results and assess the sensibility between codes and libraries. The parameters of interest are the k_{eff}, the delayed fission neutron fraction (DNF), the power distribution and the isotopic concentration in the Beginning of Cycle (BOC) and End of Cycle (EOC). The final results are expected to differ to a certain degree, as the different codes and libraries add new uncertainties to the models.

To accomplish this, the data obtained with the SERPENT-2 code [1] and MCNPX [2] codes are compared to already established data from SCALE codes KENO-VI (3D Monte Carlo transport code) and NEWT (2D deterministic transport code) [3, 4]. Standard UO_2 fuelled PWR assemblies based on the Angra-2 FSAR [5] were modelled and used as the reference case. These assemblies are loaded with UO_2 with three different enrichment values: 1.9%, 2.5% and 3.2%. So, other three cases were modelled by directly replacing each UO_2 fuel by the TRU fuel. The fissile concentration of each of the TRU fuel was adjusted to match the criticality of the standard UO_2 fuel.

For this purpose, the codes used for the calculations were NEWT for the 2D neutron transport and ORIGEN-S for burnup and decay in SCALE 6.0. The TRITON module prepares problem-dependent cross-sections and couples both codes. These assemblies were also modelled in SERPENT Monte Carlo code for the neutronic calculations using 3D Monte Carlo method. The 3D Monte Carlo neutron transport KENO-VI was used as reference case for the criticality calculations. NEWT, ORIGEN-S and KENO-VI are all part of the SCALE6.0 code package [6]. The neutronic cross-section libraries used were the ENDF B-VII collapsed in 238 groups (SCALE) and using continuous energy (SERPENT).

2. Methodology

The fuel assemblies were modeled based on the available data provided by the Final Safety Analysis Report of Angra 2 nuclear reactor [5]. On the standard Angra 2 core, there are 6 types of fuel assemblies, each of which with different enrichment levels and number of Integral Fuel Absorber shown on Table 1.

Table 1: Assembly types in the Angra 2 reactor core.

UO_2 enrichment	Number of IFBA rods	Identifier
1.9%	0	ELE01
2.5%	0	ELE02
	12	ELE03
3.2%	0	ELE04
	8	ELE05
	12	ELE06

A reference model in KENO-VI was used to compare the k-eff values of UO_2 assemblies in NEWT and KENO-VI. Previous works [3] show that when using the TRU fuel, the burnable poison can be removed from the core. The assemblies containing IFBA rods were not part of the work, therefore the only assemblies to be evaluated are ELE01, ELE02 and ELE04.

Table 2 contains the parameters of the fuel elements used. These elements are cooled and moderated with light water, and for the development of this work, it was considered that there was no diluted boron in any of the burning stages. The modelled assemblies are shown in Figure 1.

Table 2: Parameters of the fuel assemblies

Parameter	Value
Assembly type	16x16
Assembly pitch	23.11 cm
Number of rods per assembly	236
Number of guide tubes per assembly	20
Rod diameter	1.43 cm
Active length	391.6 cm
Cladding and guide tube material	Zircaloy 4

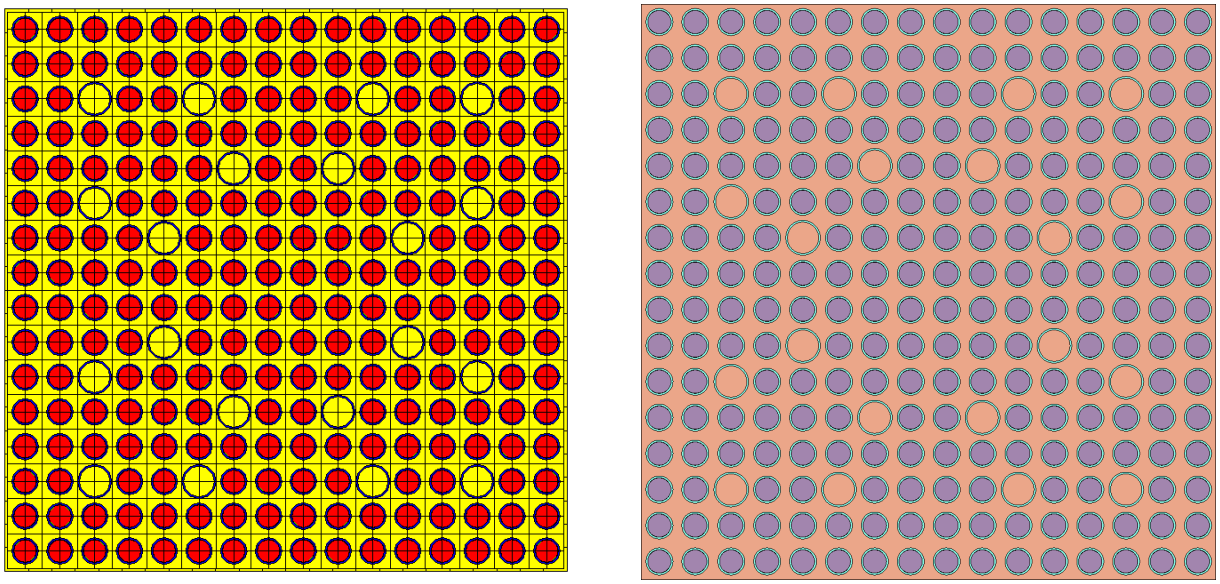

Figure 1: Fuel assembly modelled by NEWT (left) and SERPENT (right)

The NEWT code allows the collapsing of energy groups and the homogenization of materials used in the model. For the generation of cross-sections and DNF, 238 groups were collapsed into 2 groups, the first being the fast group with energies between 20MeV to 4eV and the second of 4eV to 10E-5eV. The temperatures used for the burnup phase were 873 K for the fuel, 618 K for the cladding and 583 K for the moderator. The burnup time was 868.5 days at constant specific power of 38 MW / MTU without the addition of cooling time at the end of this period.

3. Results and Discussion

The partial results of the k_{eff} calculated by KENO-VI, NEWT and SERPENT are shown in Table 3. The results using MCNPX are currently being obtained and will be added and analysed in the final version of this paper. The fissile concentrations of the TRU fuels are 14%, 20.5% and 25% corresponding to the assemblies ELE01, ELE02 and ELE03 respectively, and were obtained on previous works [3, 4]. The values of the k_{eff} obtained show close accordance between KENO-VI and NEWT. The small difference can be explained by the different methods each code applies to obtain the k_{eff}. The results using SERPENT overestimate the values of k_{eff}. This difference can be due to the different libraries used by the codes, as well as the methods to treat these cross-sections before the transport calculations, especially the temperature. As the fissile material concentration arises, the difference becomes larger, suggesting a correlation that can be further studied.

Table 3: k_{eff} calculated by KENO-VI, NEWT and SERPENT for UO_2 and TRU fuel.

	KENO-VI	NEWT		SERPENT		MCNP	
Fuel type	UO_2	UO_2	TRU	UO_2	TRU	UO_2	TRU
ELE01	1.22773 ± 0.00057	1.22512	1.22603	1.23113 ± 0.00022	-	-	-
ELE02	1.30480 ± 0.00058	1.30060	1.30450	1.30668 ± 0.00020	-	-	-
ELE04	1.36360 ± 0.00060	1.35912	1.35619	1.36488 ± 0.00028	-	-	-

For the final paper, the SERPENT model will be improved in order to obtain the criticality comparable to the other codes. With this parameter settled, the TRU fuel will be inserted in these models and the results will also be compared with NEWT. The DNF, power distribution, k_{eff} and major actinides compositions will also be compared after a burnup cycle.

4. Conclusions

In this work, the behaviour of reprocessed fuel spiked in thorium during a burnup cycle is being evaluated for differente nuclear computational codes. Three types of fuel assemblies of a PWR were analysed using the KENO-VI, NEWT and ORIGEN-S codes, coupled by the TRITON module, and with the SERPENT-2 and MCNPX Monte Carlo code. The cases using standard UO_2 were used as reference to obtain the composition of the TRU fuel.

The partial k_{eff} obtained indicate that SERPENT overestimates de criticality of the systems in comparison to SCALE. These models will be improved by generating cross-sections libraries in NJOY with the temperature used in SCALE. With the models in accordance, the TRU fuel will be evaluated in SERPENT and in KENO-VI using a continuous energy library, as well as in the MCNPX code.

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