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Preliminary study to propose a PWR Small Modular Reactor for a Nuclear Powered Submarine

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1. Introduction

Nuclear Powered Submarines are advanced military ships desired by many navies around the world. Nevertheless, currently, only six countries possess such ships: United Sates of America, United Kingdom, Russia, France, China and India [1]. Brazil is trying to develop its first nuclear powered submarine, the "Álvaro Alberto" [2]. Although this particularly project is already in advanced phase, the study for new technologies is very important for future designs more compact and reliable.

The Small Modular Reactors (SMRs) are reactors with electrical output varying from 10 to 300 MW, with integral design (i.e. all the components of primary circuit assembled within one big pressure vessel) and intrinsically safe. Nowadays, there are many projects in course like the NuScale (USA), CAREM (Argentine), SMART (South Korea), UK SMR (United Kingdom), among others [3], that envisage to increase the lifetime and the safety of the reactors and to reduce the initial implementation costs, in order to enhance the competiveness of nuclear power throughout the world, when compared with reactors from Generation II [4].

Due to the main advantages of size and safety, the conceptual idea of the SMRs is very interesting to be implemented onboard nuclear powered ships, in which there is a restriction of space to accommodate the necessary equipment and huge need for safety, not only focused on the crew that is living around a nuclear plant, but also on the rest of the world, as it is a mobile nuclear power plant.

In this context, the aim of this research is to propose a Pressurized Water Reactor (PWR) core for the nuclear reactor of a military submarine intended for thirty years of operation. At this work, the objective is to present the first steps taken in the design by the qualitative analysis and comparison of cores with different sizes, fuel enrichments, volume fractions of components (moderator, fuel and cladding) and reflectors. It was adopted as premises the PWR type because it is a stablished technology that Brazil already has the design knowledge [5].

The motivation for this study arises from the desire of Brazil of having nuclear powered submarines and from the continue need of preservation of the knowledge acquired in the design process of the shore test plant (LABGENE) and the "Álvaro Alberto" plant. In more expanded view, the study of SMR technology and proposal of this type of power plant can bring several advantages for Brazilian electrical grid, as, due to the continental dimensions of the country, there are still places in which the electrical energy is provided by small diesel generators and only for a limited period of the day. Therefore, this kind of power plant could be implemented near these remote communities to improve their life quality.

2. Methodology

In this work, different combinations of nuclear cores are analyzed and compared through the multiplication factor (k_{eff}). The combinations are made varying some parameters of the core like the total volume, the volume fraction of moderator (light water H_2O), fuel (uranium dioxide UO_2) and cladding (Zircaloy-4 Zr4), the fuel enrichment and the material and thickness of radial reflector, according to Table I. The model of the nuclear reactor was simplified to a homogenous core. Although the expected differences in the value of the k_{eff} compared to a real heterogeneous PWR core, for this first comparative analysis, the simplification allows to run more cases, and, therefore, explore more possibilities.

It was used a Monte Carlo Method based code, OpenMC, to run the cases and obtain the values of k_{eff} through the eigenvalue calculations. The main input parameters were defined as 15000 particles per generation, 10 generations per batch, 20 discarded batches and 150 total batches.

Table I: Parameters used to specify the reactors cores to be analyzed.

Core volume	Components Volume Fraction H_2O – UO_2 – Zr4	Fuel Enrichment	Radial Reflector
0.6 to 2 m^3, steps of 0.2 m^3 Relation H/D= 1.16	60% - 28.57% - 11.43% 55% - 32.14% - 12.86% 50% - 35.71% - 14.29% 45% - 39.29% - 15.71%	3.5%, 7%, 10%, 15% and 19.5%	Material: water and AISI304 Thickness: varying from 10% to 50% of core diameter

3. Results and Discussion

In the first step of the study, the homogeneous core was simulated to evaluate the effect of changing the parameters expressed in Table I, obtaining the results according to Fig. 1.

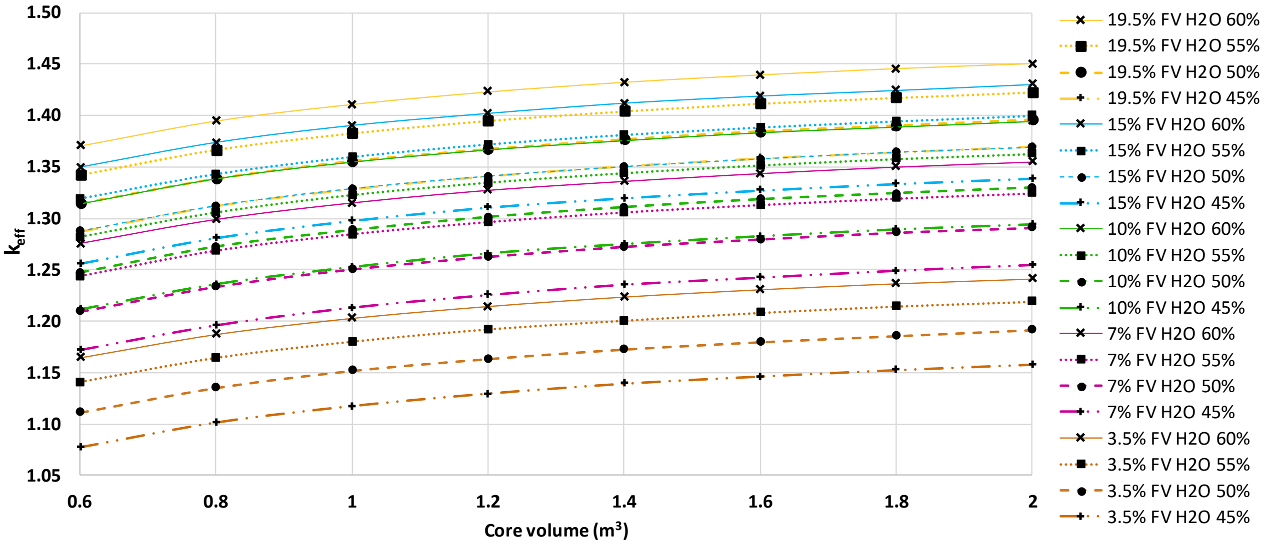

Figure 1: Variation of the k_{eff} with the core volume for different components fractions and fuel enrichment.

One can observe from Figure 1 that the values of k_{eff} increase with the increase of the fuel enrichment and the moderator fraction. This behavior complies with the expected from literature as higher enrichments elevates the number of available fissile nuclides and more moderator improves the moderation, elevating the thermal neutrons flux [6]. Other important point is that all the cores are already supercritical. As in this first step, there are no control elements (i.e. neutrons absorbers) to regulate the chain reaction and keep the reactor critical, it is expected to achieve values of $k_{eff} > 1$. Actually, it is desired, in this stage of the design, to have higher values of k_{eff} to enable longer core duration life.

Continuing the work, it was evaluated the effectiveness of including a radial reflector in the core. For that, it was tested reflectors of water and stainless steel AISI304 with different thickness varying from 10% to 50% of core diameter. First, it was verified if the reflection behavior would not be affected differently regarding the core volume, fuel enrichment and volume fraction of the components. This analysis was performed to guarantee that the evaluation of the reflector only for some selected cores could be extended to the others without jeopardizing the results. The results exhibited a very predictable behavior of k_{eff}, increasing proportionally with core volume, fuel enrichment and the moderator volume fraction. Therefore, for the comparison between the reflectors, it was simulated only cores with 45% of moderator volume fraction and 0.6 m^3 of volume. The results are presented in Figure 2.

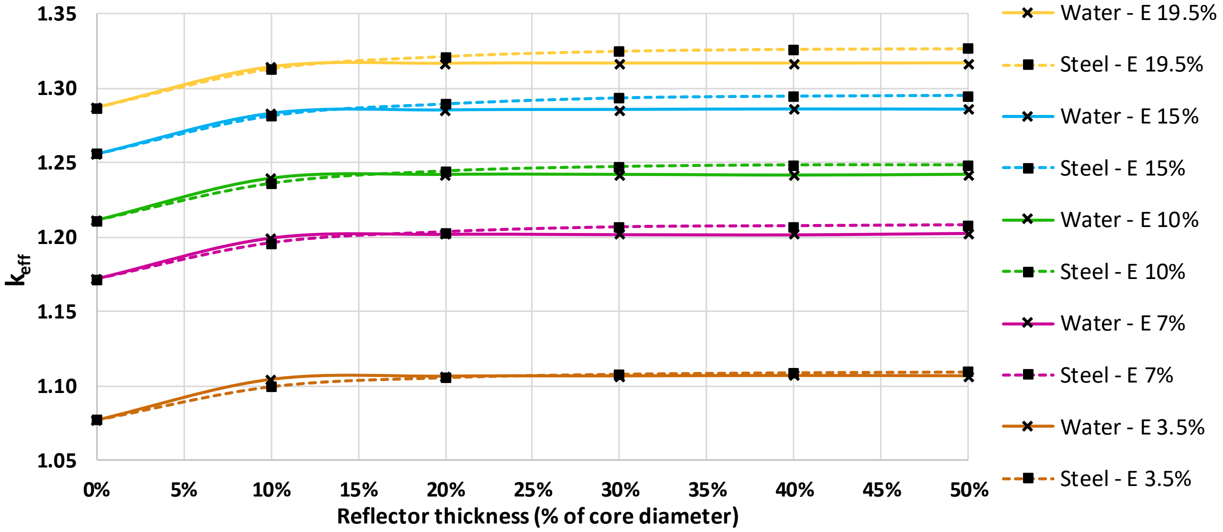

Figure 2: Reflector efficiency comparison for different fuel enrichments.

Analyzing the results from Figure 2, the steel reflector becomes more efficient than the water one for all cases after around 20% of core diameter and the increase of k_{eff} practically stabilizes after 30% of the core diameter. Thus, the reflector of AISI304 and thickness equivalent to 30% of core diameter was chosen to proceed with the studies. The k_{eff} for the all cores combination was calculated again and only the ones with $k_{eff} < 1.26$ was chosen for next step of the study. This value is based on existing reactors and it complies with the expected maximum total negative reactivity possible to be inserted by the control rods.

For the next stage of the work, it was included in the core the neutron absorber material. It was considered the alloy of Ag-In-Cd (80%-15%-5% of concentration). In the first approach, the alloy was also included in the core homogeneously diluted, varying its fraction from 0.1% to 2.2% of core volume, discounting this value from the water volume fraction showed in Table I. The goal was to determine the absorber concentration necessary to achieve the core criticality. Nevertheless, it was observed a sharp decrease in the values of k_{eff}, achieving criticality with only 0.1% of absorber volume fraction. These results reveals that including absorbers in homogeneous model of the reactor does not bring results coherent with the reality of the heterogeneous reactors. The physics changes considerable when lumping the components in the reactor core and, particularly in the case of the absorber, the probability of this material to capture one neutron is enlarged if it is homogenously diluted in the core.

As a new attempt of still keeping a simplified reactor core model but aiming to improve the results obtained, the absorber alloy was inserted in the core lumped in two concentric rings of same thickness placed one at 1/3 and other at 2/3 of the radius. New values of k_{eff} were, then, calculated with the absorber, varying the corresponding volume fraction from 0.1% to 4% of the core. With this approach, the results, presented in Figure 3, exhibited a better behavior, not strongly decreasing the k_{eff}. However, the cores with higher enrichment are still supercritical and additional measures, either increase the fraction of Ag-In-Cd or inclusion of burnable poison or boron in the water, are necessary to have a critical core and, thus, ensure the safety of the reactor.

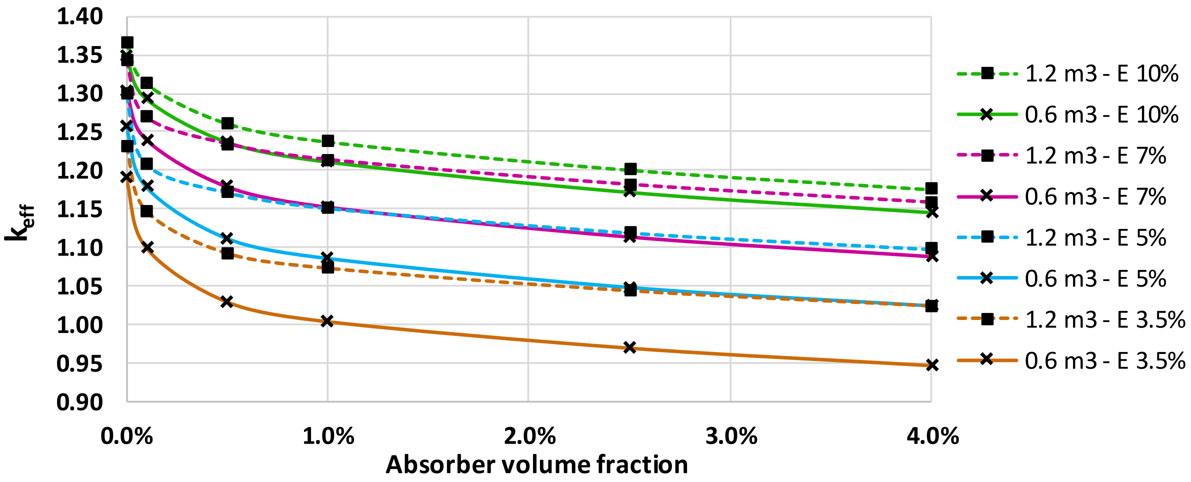

Figure 3: Variation of k_{eff} with the volume fraction of the ring absorbers for different fuel enrichment and core volumes.

4. Conclusions

All the configurations proposed for the nuclear reactor core are already supercritical, which is important to ensure a longer duration of the core, however, it is still necessary to properly include the control elements so the reactor can safely operate and close one preliminary design for the core.

Despite being simple, the homogeneous model proved to be suitable for this very preliminary conception of the core, enabling to explore various possibilities in a short time. However, from the inclusion of the absorbers on, the model already presents discrepancies from what expected based on real PWR commercial plants. For that, these results must be used with caution, knowing its limitations.

For future evolutions on this study, it is necessary to select very few most promising configurations (i.e. high k_{eff} value, but with possibility of being controlled using minimum resources due to onboard space constraint) to evolve the model to heterogeneous core. In this one, other aspects must be verified as the design of the fuel assembly to ensure proper moderation ratio, position and quantity of the control elements and the thermal hydraulics feasibility. Finally, the study of the core burnup and, consequently, estimation of the cycle duration between refueling shall be determined.

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