Influence of void fraction in the power distribution for a GE-12 fuel assembly

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Abstract. Analysis of the influence of void fraction distribution is very important to learn about the fission process and heat produced by the fuel assembly, here in this study several void fraction (VF) values along different burnup values have been considered in order to observe their influence in power distribution, uranium consumption, neutron flux and behaviour for a GE-12 fuel assembly. For this study, burnups up to 60 MWd/kg and VF values up to 0.8 were considered setting the uranium enrichment at 3.5 weight percent at the start of every VF scenario, results show that higher void fractions reduce the thermal flux decreasing thermal fission and limiting heat production.

1. Introduction

Fuel assembly design plays a very important role in reactor core performance. A nuclear reactor is designed for heat production in the core. The heat is extracted through the coolant by forced convection. Core thermal design seeks to extract the highest amount of fuel heat within temperature limits of the cladding maximizing the fuel-coolant contact area (Rust, 1979). This implies a large number of fuel rods in the fuel assembly. Nowadays, some fuel assembly designs are built in a ten by ten rods array basis, like the GE-12 fuel assembly shown in Fig. 1 (Dalrymple, 2012).

![Figure 1. GE-12 fuel assembly radial cut array.](image)

As the fuel heat is being extracted from the rod, the amount of bubbles in the water surrounding the fuel rod increases and accumulates, going through different flow regimes. In a BWR, typical VF values go from 0.38 to 0.75, reaching an annular flow regime where the liquid flows as a film along the cladding while the steam goes through among the fuel rods.
On the other hand, current fuel assembly designs use inner water rods to provide a more homogeneous burnup in the fuel. The GE-12 design has two inner water rods occupying the space of 8 fuel rods.

2. Methodology
In order to perform the neutronic analysis of the GE-12 fuel assembly, the CASMO-4 (Rhodes, 2007) code from Studsvik was used. CASMO-4 is a multigroup two-dimensional transport theory code capable of performing burnup calculations for BWR and PWR lattices or simple pin cells by solving the transport equation using the characteristic method.

The code handles a geometry consisting of cylindrical fuel rods of varying composition in a square pitch array that allows for fuel rods loaded with gadolinium, erbium, integral fuel burnable absorber (IFBA), burnable absorber rods, cluster control rods, in-core instrument channels, water gaps, and cruciform control rods in the regions separating the fuel assemblies.

In general, a GE-12 fuel assembly has up to 100 pins. In particular, in this study 92 fuel rods with 3.5% enrichment and 8 water rods are considered. The input file was run considering an assembly with diagonal symmetry. The initial power density considered was 50.66 KW/lit, with a fuel and moderator temperature of 739 K and 559 K, respectively. Every input file was run for burnups up to 60 MWd/kg in every VF scenario considered.

3. Results and discussion
In the present study, three VF values are evaluated for a GE-12 fuel assembly. They are of 0.2, 0.4, and 0.8 VF values. Also, three burnup values are considered namely 20, 40, and 60 MWd/kg.

Figures 2-4 show the fast neutron flux for burnups over 60 MWd/kg with 0.2, 0.4, and 0.8 VF values in coolant. Figures 5-7 show the thermal neutron flux for the same parameters. The fast neutron flux energy range studied goes from 0.625 eV to 10 MeV, while the thermal neutron flux goes from 0.005 eV to 0.625 eV. Fast neutrons are the first born from fission, so the highest concentration is seen on the fuel rods zone in every scenario studied (Figures 2-4). The thermal neutrons are moderated fast neutrons, so the highest concentration is seen on the water rods and the boundaries of the assembly (Figures 5-7).

As void fraction increases, fewer fast neutrons are moderated, so the quantity of thermal neutrons is lesser in 0.8 than 0.2 VF values. This means that there will be less thermal neutrons colliding with uranium 235, producing less fissions and fast neutrons. The uranium consumption is shown in Figures 8-10.

![Figure 2](image_url). Fast neutron flux for a 20 MWd/kg burnup with different VF values.
Figures 3-7 show the uranium 235 consumption for the same parameters seen in the neutron flux. At the start of every scenario studied, the uranium enrichment is stated at 3.5 weight percent. As void fraction increases, less uranium 235 is consumed due to thermal neutron flux decrease. However, the uranium 235 consumption at the boundaries is still the highest consumption of the whole assembly. This could be due to neutrons first impact with the external fuel rods, while inner fuel rods suffer from self-shielding. The fast neutron flux does not affect the uranium 235 consumption, but other elements consumption not shown in this study.
Figures 8-10 show the normalized power distribution for 20, 40, and 60 MWd/kg burnups with 0.2, 0.4, and 0.8 VF values in coolant. As the VF increases for every scenario studied, the power distribution in the boundaries increases while in the central zone decreases. The highest values of the power distribution in every graphic agree with the lowest values of the weight percent of uranium 235.
Figure 12. Power distribution for a 40 MWd/kg burnup with different VF values.

Figure 13. Power distribution for a 60 MWd/kg burnup with different VF values.

The infinite medium multiplication factor ($k_{inf}$) is the ratio of the number of neutrons produced in reactor and the number of neutrons loss by absorption in reactor with no leakage (Duderstadt, 1976). The behavior of the $k_{inf}$ for the three VF scenarios studied is shown in Fig. 14. As burnup increases, the $k_{inf}$ decreases for the three VF values. At the beginning of the burnup steps, the $k_{inf}$ has a higher value for the VF scenario of 0.2 than the VF scenarios of 0.4 and 0.8. This is because uranium enrichment has the same value at the beginning of the burnup steps for all the scenarios studied, so there are more fissions produced when the moderation is higher for the VF scenario of 0.2 than in the others two scenarios. At the end of the burnup steps this behavior is inverted, so the highest $k_{inf}$ corresponds to the VF scenario of 0.8 while the lowest $k_{inf}$ value corresponds to the VF scenario of 0.2 since the moderation is lower. This means that there are less thermal neutron colliding with the uranium 235 but there are more fast neutrons colliding with another fissionable materials, like uranium 238 which can also produce plutonium 239 when absorbs a thermal neutron.

Figure 14. $k_{inf}$ vs burnup for different VF values.
4. Conclusions
A study on the effect of void fraction’s change with burnups over 60 MWd/kg on neutron flux, uranium consumption, power distribution and $k_{inf}$ behavior for a GE-12 fuel assembly has been conducted.

For higher VF values in the same burnup scenario studied, fast and thermal neutron fluxes decrease while uranium 235 weight percent increases. The thermal neutron flux shares a similar behavior with the power distribution behavior due to its contribution to the fission production. In all the VF scenarios studied, the $k_{inf}$ value decreases as the burnup increases. The $k_{inf}$ higher decrease is seen for the VF value of 0.2, while the lower decrease is seen for the VF value of 0.8. This might be due to the rate consumption of uranium 235.

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