Neutronic Analysis of Dual Cooled Annular Fuel with OpenMC Monte Carlo Code

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1. Introduction

The dual cooled annular fuel is an innovative PWR fuel design composed of internal and external cladding tubes in which annular fuel pellets are stacked. The coolant flows both in the interior and the exterior of the fuel rod. The dual cooling and the reduced fuel pellet thickness promotes a more effective heat transfer from the fuel rod to the coolant. Therefore, from the thermal-hydraulic point of view, the annular fuel design allows an increase in core power density, while maintaining or improving safety margins [1].

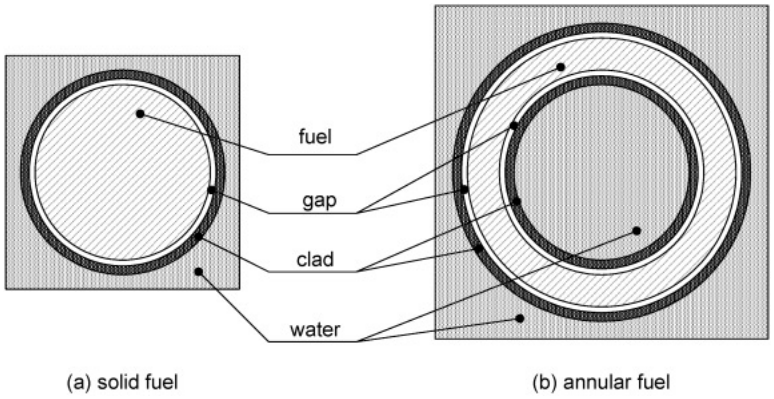

Figure 1: Horizontal cross section of fuel rods [2].

However, the geometric changes from the usual PWR solid fuel and the fact that the coolant (which is also the moderator) flows both on the surroundings and in the interior of the fuel rod demands a detailed neutronic analysis of this new fuel design. Most deterministic neutron transport codes are not suitable for this analysis, since the equivalence relations for heterogeneous resonance integrals adopted in these codes were formulated only for solid cylindrical rods [2]. This fact demands ad hoc modifications in those deterministic codes or the employment of the more general Monte Carlo stochastic approach, which is the option chosen for this research. OpenMC is a Monte Carlo neutron and photon transport code developed by the MIT Computational Reactor Physics Group [5]. Unlike most Monte Carlo codes, OpenMC is an open source software, allowing any person in any country to download, run and even

modify it without restrictions. In addition to that, OpenMC allows Doppler Broadening of nuclear cross sections to arbitrary temperature through the Windowed Multipole Method, automatically runs in parallel in multi-core computers and simplifies pre and post-processing through a Python API, making it very suitable for the task in hand.

2. Methodology

The purpose of this analysis it to evaluate the neutronic behavior of the dual cooled annular fuel in many operational conditions and compare it to the usual PWR solid fuel. In addition to the solid fuel, two annular fuel designs will be considered: a 15×15 assembly (labeled PQN-01) and a 13×13 assembly (labeled PQN-02). Table I shows the cold ($298\,K$) dimensions of the three fuel designs considered.

Table I: Dimensions for considered fuel lattices (units of cm) [1].

	Solid Fuel (17×17)	PQN-01 (15×15)	PQN-02 (13×13)
Fuel rod pitch	1.260	1.431	1.651
Inner channel radius	-	0.3365	0.4315
Inner clad outer radius	-	0.3935	0.4890
Inner gap outer radius	-	0.4000	0.4950
Fuel outer radius	0.4096	0.5990	0.7050
Outer gap outer radius	0.4178	0.6050	0.7110
Clad outer radius	0.4750	0.6600	0.7685

It is important to notice that in both annular fuel designs the overall assembly size is approximately the same as the reference solid fuel 17×17 assembly. Fuel assembly as well as fuel rod and pin cell models will be simulated. In all models, the fuel is UO_2 at 4.95% enrichment and the coolant is light water. Moreover, the gap is filled with pure helium and the clad is made of a zirconium alloy. To take into account the thermal expansion of the fuel rods (solid and annular) as the temperature of the model changes, the dimensions presented in Table I and the densities of all materials will be adjusted following the equations described in references [6] and [7]. Nuclear cross sections for all the nuclides present in the models will be obtained from the nuclear data library ENDF/B VII.1, which is available at [8] already in the format readable by OpenMC and comes with the Windowed Multipole data library required for Doppler Broadening the nuclear cross sections.

Several different scenarios will be analyzed: scenarios with homogeneous temperature distributions, scenarios with a higher fuel temperature than coolant temperature and scenarios with varying boric acid (H_3BO_3) concentrations in the coolant. In addition to that, a fuel rod model will be depleted using OpenMC's depletion module in order to assess the fuel burnup and fuel cycle length in the high power density régime.

The quantities that will be calculated (tallied, in the Monte Carlo jargon) in those scenarios include the effective multiplication factor k_{eff}, the conversion ratio C (defined as the ratio of U^{238} capture rate and the U235 fission rate), the soluble poison absorption rate, the fuel rod power distribution, the neutron flux spectrum and the temperature reactivity feedback, among others.

In every simulation, k_{eff} and neutron source distribution convergence will be verified, as it should be done in any Monte Carlo simulation to guarantee the accuracy of the results [4].

3. Preliminary Results

As a first step of the dual cooled annular fuel neutronic analysis, the OpenMC code was benchmarked against MCNP, another Monte Carlo neutron transport code. The conversion factor C and the effective multiplication factor k_{eff} were compared considering both annular fuel geometries and the solid fuel. Table II shows the mean values obtained by OpenMC, by MCNP and their relative discrepancy.

Table II: Neutronic parameters comparison between OpenMC and MCNP [2].

	C			k_{eff}		
	OpenMC	MCNP	Rel. disc.	OpenMC	MCNP	Rel. disc.
Solid Fuel	0.495	0.503	1.59%	1.3998	1.3867	0.94%
PQN-01	0.537	0.539	0.37%	1.3589	1.3556	0.24%
PQN-02	0.504	0.505	0.20%	1.3819	1.3759	0.44%

The results show very good agreement between OpenMC and MCNP. It is also possible to notice the differences in C and k_{eff} between the solid fuel and the annular fuel designs, confirming the necessity of a careful neutronic analysis of the annular fuel.

The solid fuel has the greatest k_{eff} and the smallest C from the three geometries, indicating that the annular fuel has a larger U^{238} capture rate. This was expected, since the water presence inside the annular fuel reduces the self-shielding effect, increasing the U^{238} capture rate and therefore reducing the multiplication factor.

4. Acknowledgments

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5. References

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