Analysis of the methods for group constants generation for calculation of a large SFR core using Serpent 2 and CriMR codes

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Abstract. This work aimed at generating homogenized group constants using the Serpent code and then using the CriMR diffusion code to model the large SFR OECD 3600 MWth MOX core. The results were compared with a full core reference Monte Carlo solution by Serpent. Reactivity feedback parameters were also considered. Generating the group constants from separate fuel assemblies allows for simultaneously carrying out calculations and then using the results as input in diffusion codes rather than waiting so long for a 3D full core Monte Carlo calculation to be completed. From the results of the integral parameters we see a close agreement in the calculation codes. The differences can be attributed to the errors that could arise from generating the constants from individual sub-assemblies. The differences in the underlying physics and approximations used in development of the codes could also be a factor. Another way the errors could be reduced is by checking to see that the sub-assembly configurations used in the non-multiplying zones are as close as possible to the real layout in a full 3D core.

1. Introduction

Reactor neutronic analysis methods today have come a long way since the beginning of the development of the first nuclear power reactors. Previously, understanding reactor steady-state and transient processes at given conditions took a lot of time to achieve as well as high costs.

But these days, computer codes developed by researchers have significantly reduced calculation time of important neutronic parameters and the heavy costs formerly required. This has really boosted the understanding of researchers of various reactor performance and safety scenarios.

Reactor safety analysis can be carried out simultaneously by various groups of engineers and scientists thereby reducing the time required to study certain results during reactor operations. However, these codes must be verified and validated extensively to ensure a high degree of accuracy and similarity with experimental results [[1]].

Some of the methods used today involve codes which are fundamentally different in principles behind their development. Basically, there are two main kinds of codes today. One group operates on

the principle of nodal diffusion analysis and the other is based on lattice physics analysis. The nodal diffusion codes are also known as deterministic codes while the lattice physics codes are stochastic codes that operate using the Monte Carlo method of probabilistic analysis.

There are possibilities of using advantages of both approaches for safety analysis improvement, for example Monte Carlo based codes can be used as group constant generating tool for deterministic codes.

This work has two aims. First is to investigate the possibilities of Serpent [2] Monte Carlo code to generate group constants. Second is to validate nodal diffusion code CriMR [3] using those group constants as input. For that two methods of group constants generation were considered: one using full core simulation and another with simulation of few subassemblies of chosen types.

2. Models and methods

The OECD/NEA specified benchmark [4] for the SFR 3600 MW thermal MOX core was chosen as reference for this work.

2.1. SFR MOX Core description

SFR core consists of 453 fuel assemblies and 330 radial reflector assemblies, in turn fuel assemblies are presented in two types – 225 with low enrichment MOX fuel and 228 with high enrichment MOX fuel. Each fuel rod has upper and lower axial reflector and gas plenum. Core layout is shown on figure 1.

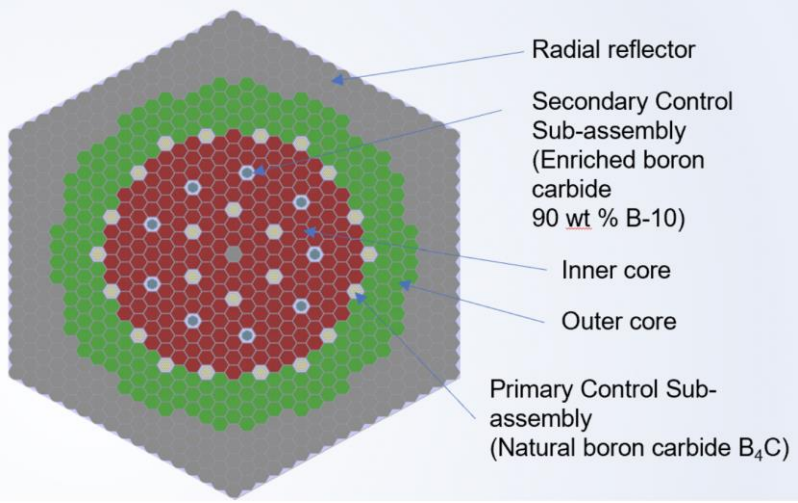

Figure 1. Radial Core Layout of the 3600 MWth SFR core, oxide fuel.

2.2. Group constants generation methods

The methodologies to be employed for the group cross section generation involves using Serpent code [5] generate them from a 3D core, and secondly from the fuel assembly and the non-multiplying regions – the axial and radial reflectors, CSD, DSD and the empty control rod channel.

Two methods for group constant generation are considered:

- method involves generating the constants from full core 3D Monte Carlo simulation (it also can be used as a reference solution for further investigation);
- method involves generating constants from the component fuel sub-assemblies and non-multiplying sub-assemblies of the core.

The following group constants are to be generated for the fuel sub-assembly and for the non-multiplying zones of the reactor core:

- Macroscopic cross sections

- Group to group scattering matrices
- Diffusion coefficient

For both methods standard Russian grid of 26 energy groups was used [6, 7].

Subsequently, using the results generated a 3D model of the core will be developed using the diffusion code – CriMR. This method for core analysis and verification of the use of Serpent for generating group constants was adopted by Nikitin et al. [8]. Nikitin et al. demonstrated the feasibility of this approach in their study of the OECD/NEA SFR 3600 MWth benchmark. They carried out their work using DYN3D and PARCS deterministic codes and Serpent Monte Carlo code.

Baiocco et al. used a similar method for validation of PARCS, a diffusion code [9]. However, they used Helios (a deterministic code) and Serpent for the generation of group constants which were used in developing a 3D model of a small PWR core.

The 3D model developed was compared with a reference 3D model of the core developed by Serpent. They also concluded that the group constants generated by Serpent and Helios had good agreement. The need for validation of codes such as Serpent is extremely important since it is relatively new in its development and the developers call for users to apply it in calculations cautiously [10].

In order to obtain cross sections of non-multiplying subassemblies for second method they were surrounded by fuel assemblies.

2.3. CriMR code

CriMR nodal diffusion code has been under development at the National Research Nuclear University MEPhI. It is written with FORTRAN programming language.

Like other nodal diffusion codes, it solves the Boltzmann neutron transport equation [11].

CriMR uses the following group constants from Serpent to carry out neutronic analysis of a reactor core:

- Absorption cross section
- Scattering cross section
- Diffusion coefficient
- Production cross section

3. Results

First task is verification of serpent model against benchmark. As shown in table 1, results for SFR MOX Core model developed for this work are in close agreement with reference.

Table 1. Comparison of benchmark and Serpent results.

Parameter	Benchmark	Serpent Model	Benchmark vs Serpent (pcm)
K-eff	1.0294	1.0369	-750
β-eff (pcm)	367	361	-6
Δρ$_{Na}$ (pcm)	1937	1976	39
Δρ$_{CR}$ (pcm)	-6041	-5623	418

Next task is simulation of SFR MOX Core with diffusion code using cross sections produced by Serpent. Results obtained by diffusion simulation are presented in table 2. It could be seen that results for both methods of constant generation are almost the same and also in good agreement with Serpent results.

Table 2. Comparison of Serpent and CriMR results.

Parameter	Serpent Model	CriMR (full core constansts)	CriMR vs Serpent (pcm)	CriMR (subassembly constansts)	CriMR vs Serpent (pcm)
K-eff	1.0369	1.0369	0	1.0361	72
ΔρNa (pcm)	1976	2060	84	2072	96
ΔρCR (pcm)	-5623	-5963	340	-5921	298
ΔρKD (pcm)	-1251	-1273	22	-1270	19

Radial power distribution for Serpent and CriMR is shown on figure 2, errors do not exceed 5%.

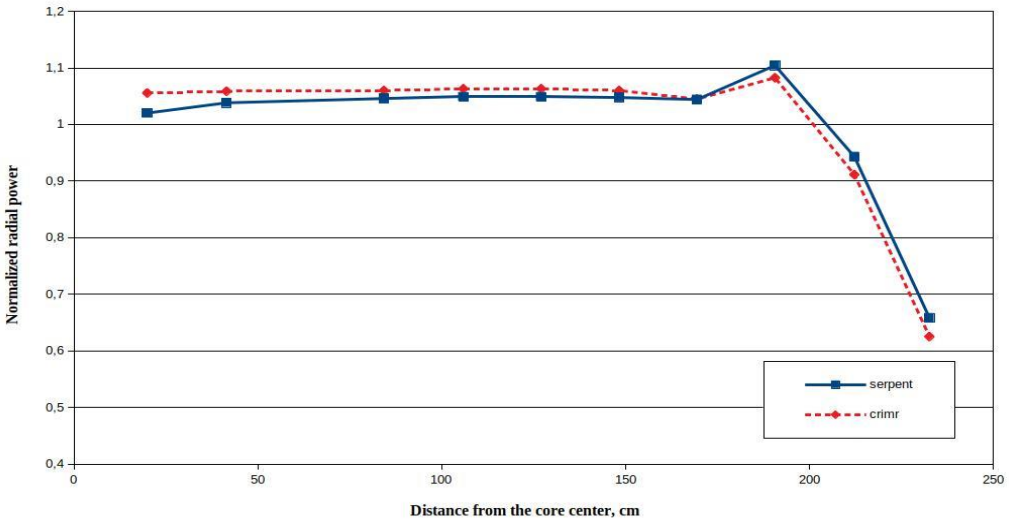

Figure 2. Radial power distribution for Serpent and CriMR.

4. Conclusion

Serpent Monte Carlo code was used to generate group constants for diffusion code CriMR. Two methods were considered, one using full core simulation and another with simulation of few subassemblies of chosen types.

Generating the group constants from separate fuel assemblies allows for simultaneously carrying out calculations and then using the results as input in nodal diffusion codes rather than waiting so long for a 3D full core Monte Carlo calculation to be completed.

From the results of the integral parameters can be seen a close agreement in the calculation codes. The differences can be attributed to the underlying physics and approximations used in development of the codes. It should be noticed that for the sake of time constraints the neutron histories and active cycles were kept to the barest minimum recommended by researchers for calculation of parameters from fuel assemblies [12]. An increase in this could also help in improving the results generated from the CriMR nodal diffusion code.

Acknowledgement

This work was supported by the National Research Nuclear University MEPhI Academic Excellence Project (agreement with the Ministry of Education and Science of the Russian Federation of August 27, 2013, project no. 02.a03.21.0005), the framework of 5-100.

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