3D Neutronic Calculations: CAD-MCNP Methodology
applied to Vessel Activation in KOYO-F

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Abstract. This paper presents a methodology for 3D neutronic calculations suitable for complex and extensive geometries. The geometry of the system design is first fully modelled with a CAD program, and subsequently processed through a MCNP-CAD interface in order to generate an MCNP geometry file. Neutronic irradiation results are finally achieved running the MCNPX program, where the geometry input card used is directly the MCNP-CAD interface output. This methodology enables accurate neutronic calculations for complex geometries characterised by high detail levels. This procedure will be applied to the Fast Ignition Fusion Reactor KOYO-F to determine first neutron fluxes calculations along the blanket as well as the material activation in the reduced martensitic 9Cr-1Mo steel vessel.

1. Introduction

Monte Carlo codes have become an essential tool for transport analysis in fusion experiments design and validation. Indeed, MCNP has been chosen by the ITER Organization as the neutronic transport reference code. Although complex geometry can be accurately modeled in Monte Carlo codes by combining surface interactions to form bodies (cells), extensive effort is needed to generate the appropriate input geometry cards. In order to achieve realistic complex input geometries, several organizations are currently developing geometry conversion systems from CAD designs to different transport codes, for both deterministic and probabilistic programs.

This new methodology, recently implemented for ITER, can also be applied for inertial fusion experiments. Moreover, neutronic calculations can be carried out for complex inertial fusion fast ignition chambers; along the paper will be presented, following this methodology, the first blanket neutronic results of the new redesigned fast ignition fusion reactor KOYO project: KOYO-F.

2. CAD-MCNP Geometry Conversion Systems

CAD system models are constructed mainly for visualization and manufacturing purposes, and although they represent a helpful tool for modeling, they are often over detailed and present excessive geometric complexity for transport calculations. In this sense, CAD models need to follow different preprocessing steps - detail suppression, geometric and model simplification- in order to obtain suitable inputs for geometry conversion systems. Additionally, a check and repair process must be performed to solve minor errors –interferences, gaps and overlaps- which are insignificant for visualization applications but crucial to meet the rigorous needs of neutronic applications. A second
key element in CAD-MCNP geometrical conversion arises for the void cell generation in MCNP. The CAD model contains only the solid cells, but particle simulation codes need the void space to be defined.

There are currently four different groups developing CAD-MCNP conversion systems [1]. The most mature code is MCAM, developed by the FDS Team (China), an integrated modeling environment system which allows CAD model preprocessing, analyzing and editing prior to CAD-MCNP geometric conversion. Furthermore, MCAM handles backward conversions (MCNP-CAD) and can also be applied to other simulation codes. Currently, a stable version –MCAM 4.2- is finished and tested, reliable for various applications such as fusion research, fission industry, accelerator analysis and medical physics.

Geometry conversion systems enable to model experiment geometries faster and with much higher complexity. However, manual editing of the generated MCNP geometry input file is not easy, appearing to be a better solution to implement new changes directly on the CAD model. In this sense, it seems that nuclear analysts willing to take advantage of these new geometry conversion systems should also become skilled in the use of CAD systems.

In the following sections will be developed an application of MCAM usefulness for projects requiring high geometric complexity. KOYO-F fast ignition fusion reactor blanket CAD model will be converted with MCAM in order to achieve first neutronic calculations with MCNPX.

3. Fast Ignition Fusion Reactor KOYO-F

3.1. KOYO-F basic design parameters

Recent progress in fast ignition has enabled to redesign KOYO IFE power plant as KOYO-F, a 4 module reactor driven by a unique cooled Yb-YAG ceramic laser system. Each module integrates 32 compression beams, one heating laser and 2 target injectors alternatively operated at 2 Hz, as shown in figure 1. Table 1 summarizes the basic specifications for the power plant [2].

![Figure 1: KOYO-F modular reactor.](image)

| Table 1: Basic KOYO-F parameters. |
|-----------------------------------|
| Net output                        | 1200 MWe (300 MWe x 4) |
| Laser Energy                      | 1.1 MJ                 |
| Target gain                       | 165                    |
| Fusion output per pulse           | 200 MJ                 |
| Pulse rep-rate in reactor         | 4 Hz                   |
| Blanket energy multiplication     | 1.2                    |
| Thermal output per reactor        | 916 MWh                |
| Total output at plant             | 3664 MWh               |
| Thermal to electricity efficiency | 41.5% (LiPb at 500 ºC) |
| Total electric output of plant    | 1519 MWe               |
| Laser efficiency                  | 8%                     |
| Rep-rate of laser                 | 16 Hz                  |
| Recirculating power of laser      | 240 MWe                |
| Total plant efficiency            | 32.7%                  |

3.2. KOYO-F Blanket concept

Fusion reactor KOYO-F presents a wetted wall chamber scheme where the renewable liquid LiPb provides neutron shielding for structural components. The surface film design must avoid simple laminar flow; otherwise surface temperature could become too hot to evacuate the ablated vapor by cryogenic pump effect. In order to achieve a turbulent flow, a liquid LiPb cascade flow design is adopted. This way, the LiPb flow is renewed both through a cascade type configuration and a porous front first wall panel, continuously forming a protective 3mm film. As shown in figure 2, each blanket is composed of 3 different steps, where the hot LiPb surface flow of each step is mixed with inner cold LiPb from the next step. A key point of this concept is the void –filled with low pressure D₂ gas- at the top of each step to allow mixing the hot surface flow with the inner cold one.
The blanket cooling system grants 2 different flows to save the electric power. One of them is the inner open surface flow described above, and the second is an outer closed loop. Finally, the blanket completes with a graphite reflector and a 9Cr-1Mo martensitic steel vessel.

4. Blanket modelling and calculations

4.1. Blanket design
The CAD blanket model –shown in figure 3- has been carried out for the equatorial blanket, in front of the target fire position, in order to evaluate maximum neutronic fluxes and activation parameters. The blanket dimension is 1m long –the inner loop is 30 cm-, while the graphite reflector and the 9Cr-1Mo martensitic steel vessel are 45 cm and 20 cm long respectively. The structural material chosen for the blanket is Eurofer97, except for the first wall, made of porous SiC/SiC.

4.2. Neutronic results
The neutron source, located 3m away from the SiC/SiC front panel, assumes a fusion output of 200 MJ, 160 MJ corresponding to neutron energy. The target repetition rate is 4 Hz, therefore the emerging neutron intensity –designed isotropical for the MCNPX calculation- rises up to \(2.84 \times 10^{20}\) n/s.

The neutronic results, calculated for a separate blanket, are detailed in figure 4. The greater neutron flux, achieved at the SiC/SiC first wall is \(\approx 4.4 \times 10^{14}\) n/cm²/s. Once through the inner loop, the flux has only dropped to \(\approx 2.0 \times 10^{14}\) n/cm²/s; the influence of the outer loop and the reflector is much more important, considerably reducing the neutronic flux (\(\approx 7.6 \times 10^{12}\) n/cm²/s) as well as the energy spectra. The energy neutron spectra at these same locations are finally shown in figure 4.
4.3. Vessel material activation

Vessel material activation after 30 years of nominal operation has been analyzed using the ACAB code, the EAF2005 data library, and the neutronic flux and energy spectrum in the vessel from previous MCNPX calculations. The total activity after 30 years of operation rises up to 0.64 Ci/cm³, downshifts to 0.28 Ci/cm³ after the first year decay and finally stabilizes around 5x10⁻⁵ Ci/cm³ for periods longer than 100 years. The most important isotopes, from the radioactivity contribution point of view, are Fe55 and Cr51 during the first month. Until the 10th year, almost all the radioactivity is caused by Fe55; afterwards the contribution of C14 sharply increases, representing more than 75% of the total activity after 100 years.

Activated material classification [3] does not represent a problem for the 9Cr-1Mo martensitic steel vessel. Indeed, regarding the limits for recycling within the nuclear industry, the activated material meets both remote recycling (the dose rate must be kept below 10 mSv/h within 50 year cooling) and hands on recycling criteria (the contact dose rate can not exceed 10µSv/h after 100 year cooling).

However, the steel vessel activated material can not be disposed as Shallow Land Burial. The important amount of Nb (0.08%) in the steel vessel generates Nb94 isotopes in the activated material, main responsible of keeping the Waste Disposal Rating (WDR) always greater than one.

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