QUALIFICATION OF BERYLLIUM POISONING MODELS FOR MARIA AND JULES HOROWITZ RESEARCH REACTORS

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ABSTRACT

This study is the continuation of ongoing beryllium poisoning research with applications to JHR in France and MARIA reactor in Poland. In order to develop more precise poisons' distribution modeling in beryllium blocks reference Monte-Carlo calculations have been done. Depletion calculations have been performed in an infinite lattice of few elements, held in the lattice physics code APOLLO2. have been benchmarked against TRIPOLI-4®, SERPENT2 and MCNP5 codes. This calculation scheme is the basis for further study, based on a complementary experimental validation phase (BENICE) currently being designed for conduction in the MARIA reactor.

Preliminary studies demonstrated that all safety requirements will be satisfied. Comparison of calculated values with the experimental irradiations and measurements of beryllium in the MARIA reactor is a key step to verify both precision and correctness of developed model. The forthcoming experiment will be the most significant evaluation of the predictions made so far and will allow a fine tuned validation of the developed depletion models.

1. Introduction

Beryllium properties allow its utilization as moderator and reflector in Material Testing research Reactors (MTR), where they are commonly used (in MARIA as both moderator and reflector, and in JHR, as reflector).. However during reactor operation, several isotopes of high absorption cross section are being generated, causing reactivity decrease. Moreover, some of them decay during reactors off-work period into some other isotopes having high absorption cross section. These isotopes are He^3, H^3 and Li^6 .

Thus the production of the mentioned isotopes is triggered by neutrons of certain energies, the reactions of interest here are presented in detail on figure 1.

The isotopes having the biggest impact on the reactivity change in thermal research reactors are lithium ($\sigma_{Li\text{-}6\ a} \approx 940$b) and helium ($\sigma_{He\text{-}3\ a} \approx 5300$b), being generated on-work periods, while tritium is mainly generated during off-work periods.

High concentrations of these isotopes affect not only flux and power distribution, but as accumulating gazeous isotopes, also physical and mechanical properties of the material.

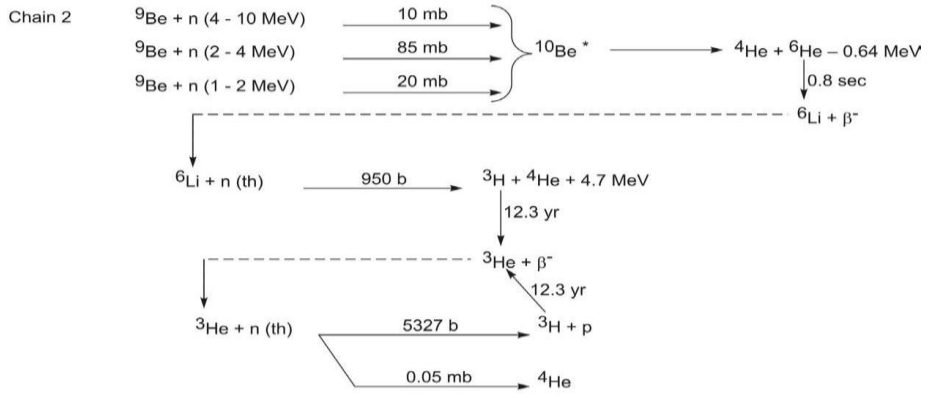

Fig. 1 (n,α) triggered reaction in beryllium taken into account in calculations.

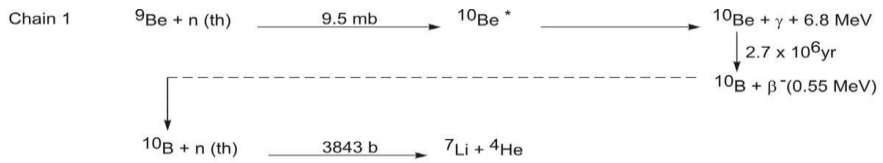

Fig. 2 (n,γ) triggered reaction in beryllium taken into account in calculations.

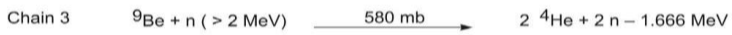

Fig. 3 (n,2n) triggered reaction in beryllium taken into account in calculations.

2. Problem description

The problem of beryllium poisoning has been observed in reactors utilizing this material in core structures. Depending on the energies and fluxes, some effects such as swelling can be more present than the others

As MARIA reactor, located and operated in National Centre for Nuclear Research in Poland, is a high flux, pool type, water and beryllium moderated material testing reactor (MTR) with the average thermal flux of $4 \cdot 10^{14}$n/cm^2·s. Fuel channels, situated in a matrix containing beryllium blocks are reflected by aluminium covered graphite blocks (figure 4.). Currently the reactor is operated using AREVA-CERCA 19.75% U-235 mass enriched (MC-5) fuel.

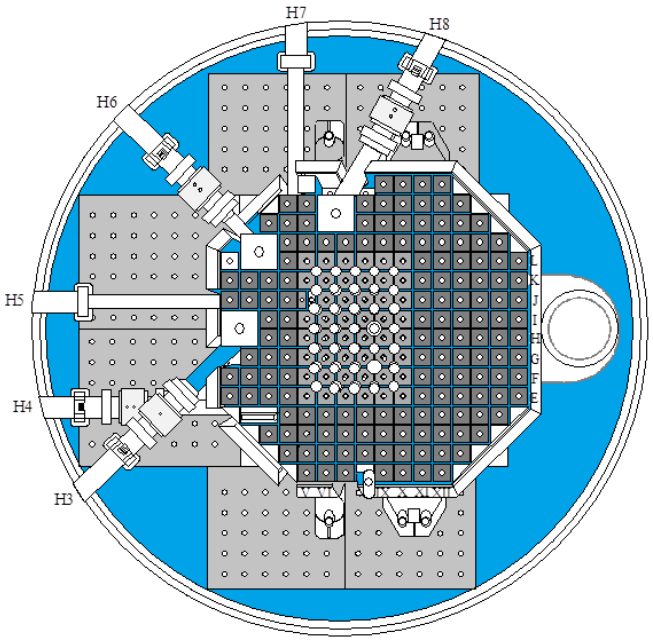

Fig. 4 Horizontal cross section of the reactor

The basic operation needs are well satisfied by currently used models and calculations, however these models lack the poisons spatial distribution development in exploited beryllium.

The typical MARIA reactor cycles consists of on and off work weekly periods, varying during the year by the total power, number fuel elements and matrix configuration – according to current exploitation needs.

This paper presents follow-up works on the implementation and qualification of the beryllium depletion scheme neutronics calculations as well as improvement of currently used calculation scheme in MARIA core calculations.
As it has been proven in previous calculations[13], initial approach to the problem in a lattice 2D transport code APOLLO2 seems to be correct, however the resulting concentration of the isotopes of interest did not match nor SERPENT2 neither MCNP Monte Carlo results.
As a consequence, calculations in TRIPOLI4 Monte Carlo code were made as an additional evaluation of the concentration discrepancies. The next step was the correction of the APOLLO2 scheme, based on single assembly in the infinite lattice model.

3. Codes, methods and models

For the purpose of this study, initial assumption for all schemes was to use fresh material compositions, where depletion calculations took into account 20 operating cycles, the power normalized to 1MW per element and the reflexive boundary conditions. Due to the specific geometry of the beryllium block (as on figure 5.), the equivalent of block's mid height geometry was chosen for calculations The representation of the calculation cell is given on figure 6.

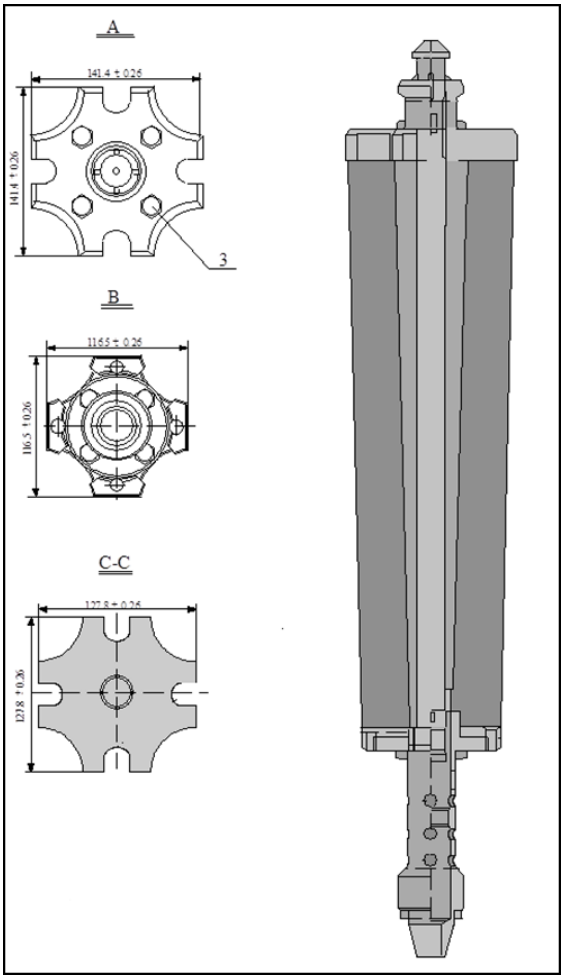

Fig.5 Beryllium block.

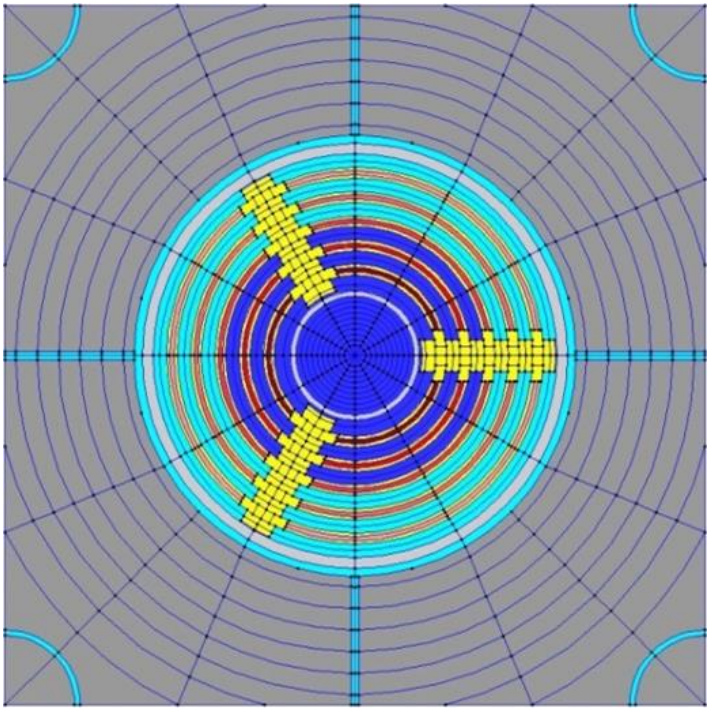

Fig.6 Cross section of calculation cell.

3.1. Monte Carlo –SERPENT2 and TRIPOLI4.10

The Monte Carlo simulations are based on the random numbers and evaluated nuclear data libraries. The modelling is fully heterogeneous, with 3D geometry and continuous energy description.

SERPENT2 provides the depletion calculations without the need to externally define the decay chain of depleted materials [4]. The reactions are available defined in the code. TRIPOLI-4® Monte Carlo however requires the definition of depletion chains for each of depleted material. The daughter and mother isotope are connected with each other by reaction type

3.2. APOLLO2

The calculations were made using two continuous energy neutron Monte Carlo neutron transport codes: SERPENT2 [2] and TRIPOLI-4® version 10 **Error! Reference source not found.**, as well as the APOLLO2.8.3. deterministic transport code, which is routinely used for 2-D LWR lattice physics calculations (APOLLO-2.8 MOC/CEA2005 & SHEM281-group code package **Error! Reference source not found.**) These codes were used in combination with neutron data library derived from the same JEFF-3.1.1 nuclear data file.

The APOLLO2 model involves the following four successive calculation steps:

1. The calculation of cross-section resonance self-shielding relies on a 1D geometry, it uses the collision probability method (2D exact Pij) in conjunction with the SHEM energy mesh (281 groups) **Error! Reference source not found.**.

2. The cross-sections are collapsed into XX energy groups using the neutron flux calculated in 281 energy groups with the Method of Characteristics (MOC), on the 2D exact geometry.

3. Based on the previously calculated XX energy group cross-sections, the neutron transport equation is solved on the whole core geometry using the Method Of Characteristics.

4. The fuel depletion is calculated from the neutron flux obtained in step 3.

This calculation scheme is summarized in the flowchart below.

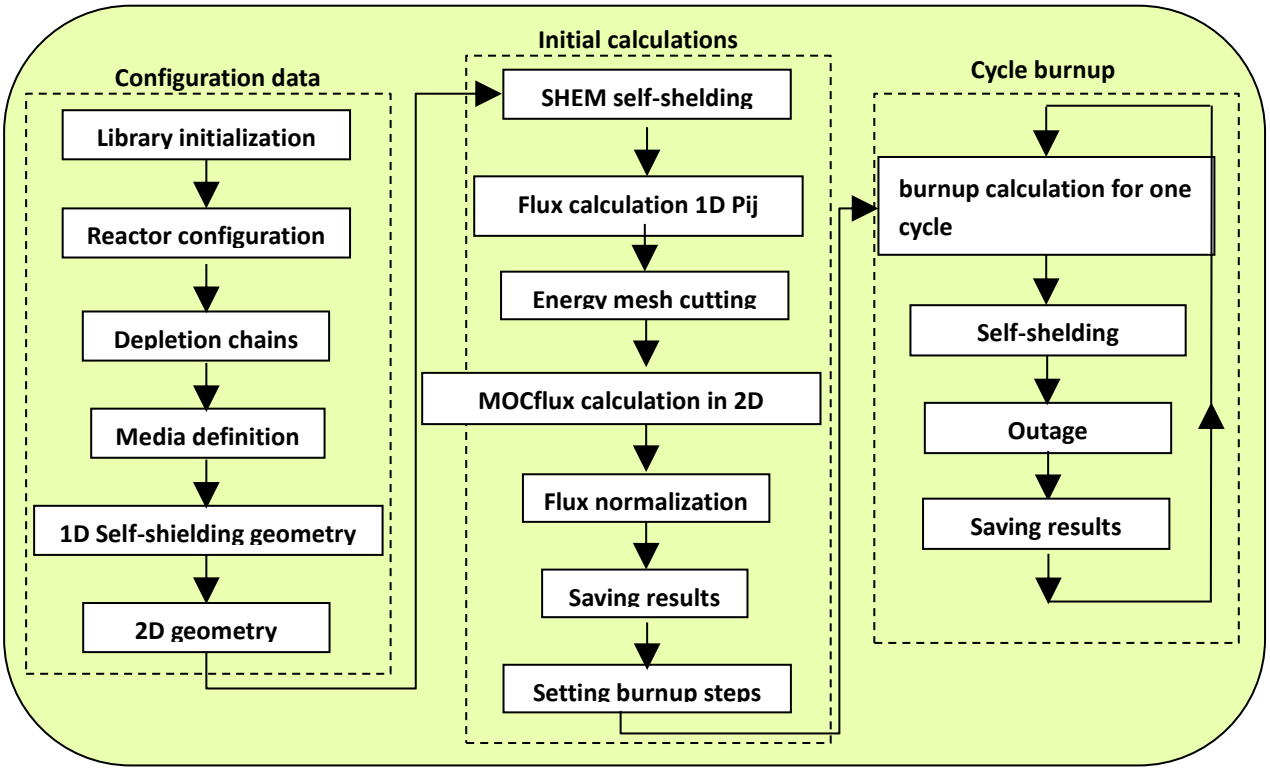

Fig. 8. APOLLO2 two-level calculation scheme

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                                &REACTION 'CREA-H3'  &SECTION   3  2
                                &REACTION 'CREA-P'   &SECTION   0  1
                                &REACTION 'NALPHA'   &CAPTURE   3  2
   TISO.'LI6'      &NALPHA      &PERE TISO.'BE9RAP'
                   &N2N         &PERE TISO.'LI7'
   TISO.'H3'       &CREA-H3     &PERE TISO.'LI6'
                   &CREA-P      &PERE TISO.'HE3'
   TISO.'HE3'      &BETA &PERE TISO.'H3'
   TISO.'BE10'     &NGAMM       &PERE TISO.'BE9'
   TISO.'HE4'      &N2N         &PERE TISO.'BE9'
                   &CREA-H3     &PERE TISO.'LI6'
   TISO.'LI7'      &NGAMM       &PERE TISO.'LI6'
```

Fig. 9. Beryllium depletion chain definition for APOLLO2 calculations

4. Results

The results of the APOLLO2 calculations with updated depletion chain procedure were compared with TRIPOLI4 and SERPENT2 results. On the figures 10 the effective

multiplication factor change for burnup calculations is presented, with the average standard deviation for the Monte Carlo result, marked. Red dashed line is a reference od the power change, representing calculated cycles. The exact discrepancy between results in pcm is given on figure 11, where the results of corrected scheme in APOLLO2 were compared with previous APOLLO2 scheme, as well as with SERPENT2 and TRIPOLI4. The presented values are related to different values of power.

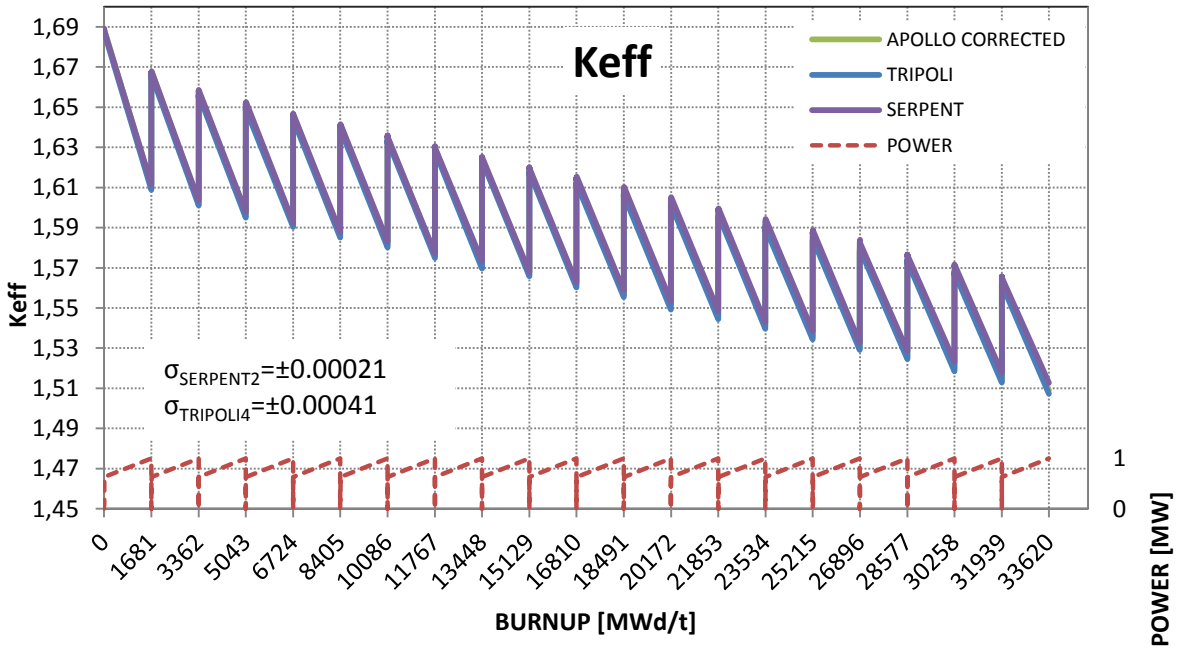

Fig. 10. Multiplication factor comparison.

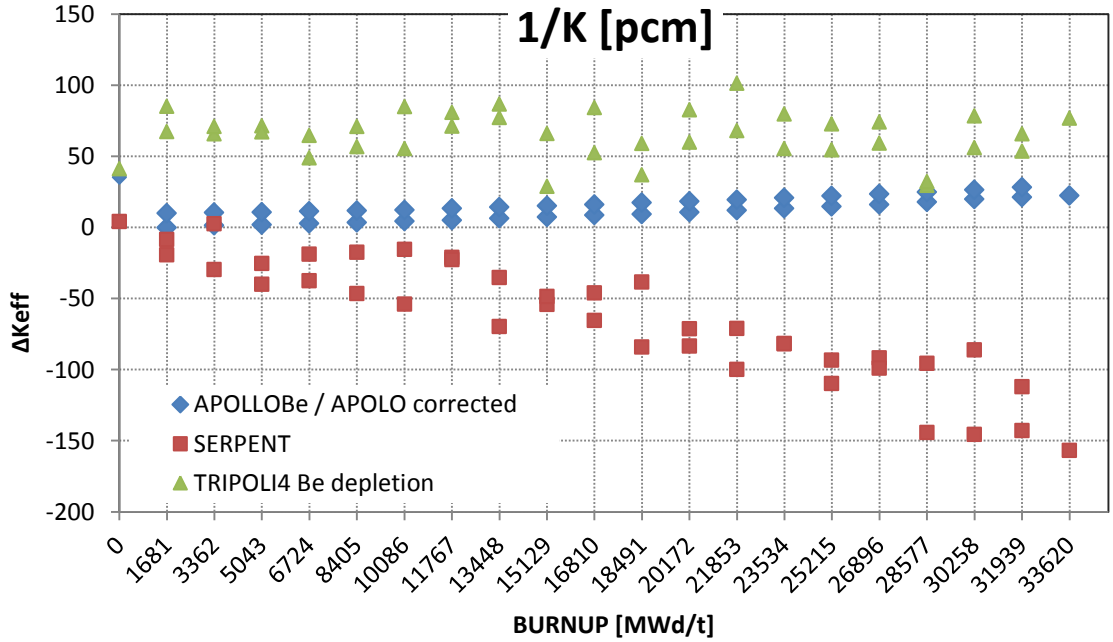

Fig. 11. Multiplication factor discrepancies – calculations with the corrected APOLLO2 depletion chain as a reference.

Additionally the analysis of the power distribution in a fuel element has been provided. As an example on the figure 12, comparison between TRIPOLI4 and APOLLO2 results, which shows that the normalized power in each fuel layer (each layer consists of 3 plates of the same radius) in both codes match acceptably well. The last layer presents the highest, yet acceptable discrepancy.

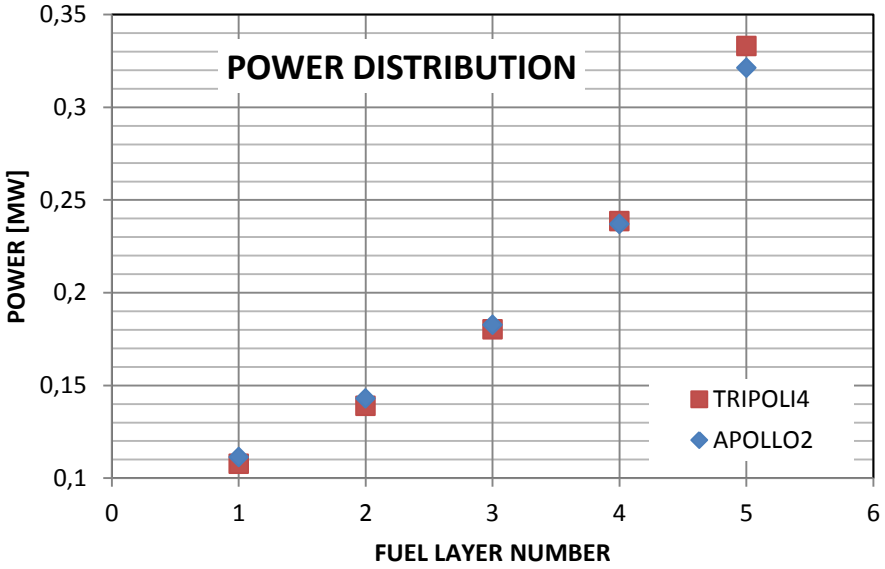

Fig. 12. Power distribution in fuel layers APOLLO2 vs TRIPOLI4.

As the correction of previous calculation scheme concerned mainly the depletion chain definition in the APOLLO2, the biggest impact on the results can be observed in the concentrations of beryllium poisons, accumulating during burnup. For Li^6 there is almost no difference in the isotope's concentration during burnup calculations for APOLLO2 both schemes (green dashed line for a previously used depletion chain, orange solid for a corrected chain). Concentrations of He^3 and H^3 however are differ a lot. The previous scheme for beryllium depletion (green dashed line) gave isotope concentrations close to zero. The corrected chain definition is much closer to the expected from Monte Carlo results. Small difference is observed between the APOLLO2 results and both SERPENT2 and TRIPOLI4 as it can be seen on figures 13-15. Average standard deviation for Monte Carlo calculations results is given on each figure.

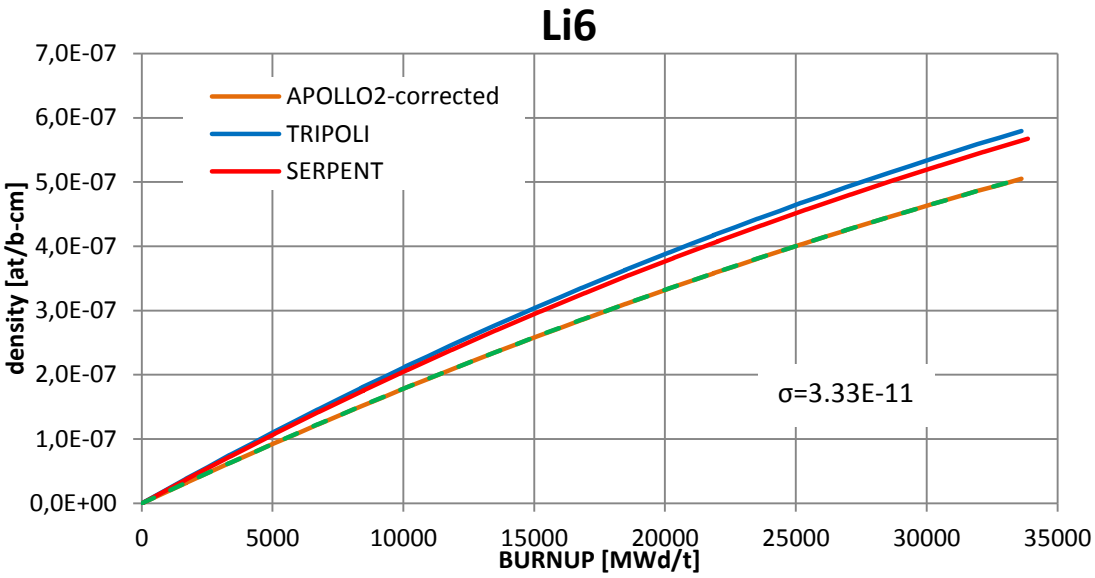

Fig. 13. Li^6 concentration build-up .

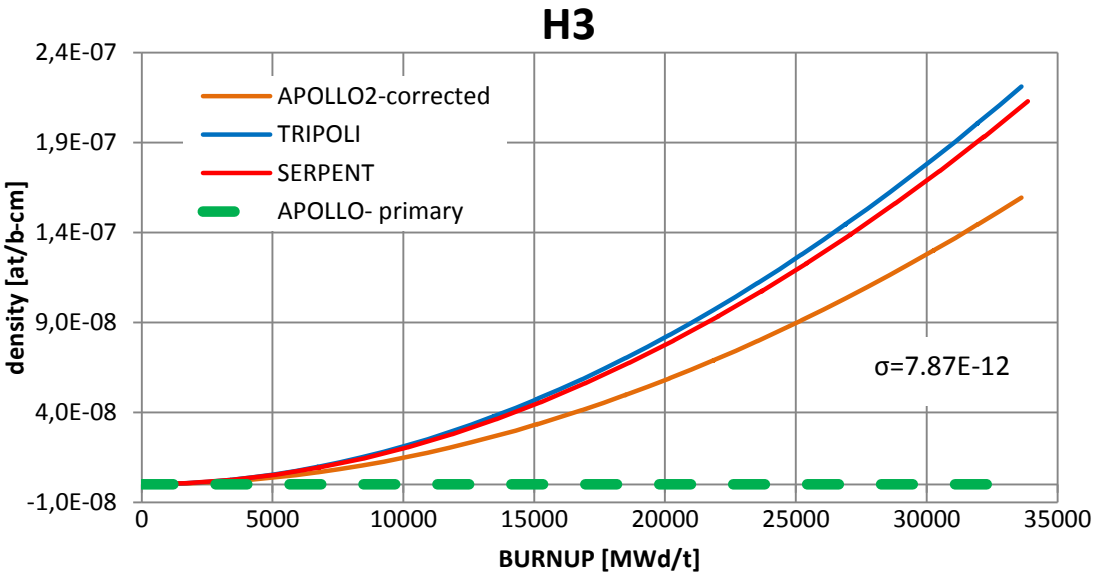

Fig. 14. H^3 concentration build-up .

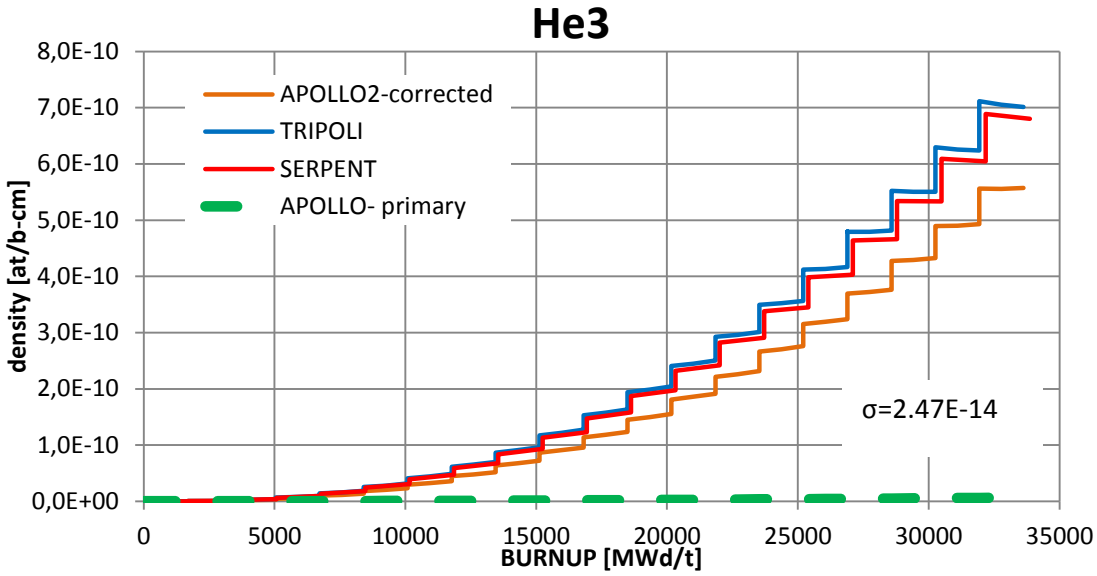

Fig. 15. He^3 concentration build-up .

5. Conclusion

The conducted studies present improvement in correct prediction of Li^6, He^3 and H^3 build-up in an irradiated beryllium. It was shown that detailed modelling of all elements impacts information obtained from simulation and the possibility of correct future measurement results, as the part of the evaluation of this calculation scheme is the experimental campaign. Further analysis assume performing the experiment firstly with fresh and as a next step with poisoned beryllium. The calculations and studies carried so far, showed the relevance of performing direct measurements.

6. References:

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