# SELF SHIELDING TREATMENT TO PERFORM CELL CALCULATION FOR SEED FUEL IN THORIUM/URANIUM PWR USING DRAGON CODE

Ahmed Amin ABD El-HAMEED[1], Mohammed NAGY[2], Hanaa ABOU-GABAL[3]
[1] BSc in Nuclear Engineering and MSc degree researcher, Nuclear Engineering Department, Alexandria University, Egypt
[2] Professor Emirates, Nuclear Engineering Department, Alexandria University, Egypt,
[3] Professor and Head, Nuclear Engineering Department, Alexandria University, Egypt

***Abstract -*** Time and precision of the results are the most important factors in any code used for nuclear calculations. Despite the high accuracy of Monte Carlo codes, MCNP and Serpent, in many cases their relatively long computational time causes difficulties in using any of them as the main calculation code. Monte Carlo codes are used mainly to benchmark the results. Ideally, the flux distribution in the lattice would be determined by solving the transport equation in the exact geometry of the lattice using continuous energy cross sections, the way a Monte Carlo code might. But because of time constraints, the calculation scheme within a lattice physics code is intended to reduce the overall computation time without sacrificing too much accuracy. The nuclear deterministic codes have limited precision due to the approximations made to solve the multi-group transport equation. Self –shielding treatment is responsible for the biggest error in any deterministic code; it is an algorithm that produces an average cross-section defined over the complete energy domain of the neutrons in a nuclear reactor.

The inaccuracy in deterministic codes is increased dramatically in the resonance period, where a small change in the neutron's energy may lead to a great change in the corresponding microscopic cross section. There are mainly two resonance self-shielding models commonly applied: models based on equivalence and dilution and models based on subgroup approach. The fundamental problem with any self-shielding method is that it treats any isotope as there are no other isotopes with resonance present in the reactor. The most practical way to solve this problem is to use multi-energy groups (50-200) that are chosen in a way that allows us to use all major resonances without self-shielding. This paper offers the best use of a known International Program (DRAGON Code) for nuclear reactor fresh cell calculations by using different methodologies, hypothesis and different Nuclear Data Library, with comparison of other well-known international codes, particularly MCNP5 code and WIMS-D5 Nuclear Data. Similar self-shielding methodologies were tested in other studies. However, all of these studies are concentrated on the low enriched nuclear fuel (3%) which is the standard fuel enrichment of most of nowadays nuclear power reactors. However, there are new designs which propose the use of higher enriched fuel for economical and safety purposes. In higher enriched fuel the effect of resonance interference of uranium isotopes is expected to change. Hence, it is important to test the validity of these self-shielding methodologies and hypothesis in such case. One of these new designs introduces the idea of developing thorium/uranium fueled PWRs. This design has been characterized by its good economics, wide safety margins, minimal waste burden and high proliferation resistance. In such design whole assembly seed and blanket are used, where individual seed and blanket regions each occupy one full size PWR assembly.

The seed fuel pin in this design has enrichment close to those used in research reactors (about 20%). In this paper, we selected the fresh seed fuel pin, which is used in thorium/uranium reactors, to be our physical model. Then, we performed cell calculations by solving 172 energy group transport equation using the deterministic DRAGON code. Two types of self-shielding models (equivalence and dilution models and subgroup models) are used. The data libraries used are WIMS-D5 and DRAGON libraries. To obtain the accuracy of the self –shielding treatments, the results are compared with the result obtained from the stochastic MCNP5 code. We also tested the sensitivity of the results to a specific change in self-shielding method implemented, for example the effect of applying Livolant-Jeanpierre Normalization scheme and Rimman Integration improvement on the

equivalence and dilution method, and the effect of using Ribbon extended approach on sub-group method.

***Keywords*** **-** Nuclear Cell Calculation; Self-Shielding Treatment; Precision of Nuclear Deterministic Codes, DRAGON, MCNP5.

## I. INTRODUCTION

Nuclear fuel cell calculation is one of the most complicated steps of neutron transport problems in the reactor core [1]. Broadly speaking, neutron physics problems need to be solved at three levels of precision with respect to the energy variable:

- In the epithermal domain, the many resonances of heavy nuclei such as uranium 238 typically require a few tens of thousands of energy groups in order to properly describe each of the resonances. It is not essential to handle the heterogeneities very precisely at this level; in fact, by using equivalence we can relate the real geometry to a homogeneous geometry.

- Handling the spectrum requires about a hundred energy groups; this can be done at the assembly constituting the elementary “mesh” of the core of a nuclear reactor, but it requires “microscopic” heterogeneity to be taken into account; this means on the scale of the fuel element or assembly.

- In practice, the multiplication factor of a core and the power distribution in it can be calculated to a few energy groups only, e.g. two groups for the usual calculations for pressurized water reactor cores. This calculation takes “macroscopic” heterogeneity into account, i.e. the differences between assemblies and axial variations. These three types of calculation must be performed one after the other: when calculations have been carried out at a given level, the mean values in space (homogenization) and energy (condensation) must be taken in order to prepare for the calculation at the next level. [8]

The first step in building lattice physics code is to create an energy group-dependent cross section library. This is the first approximation – which we assume we can accurately represent cross sections over a range of energies using constant values as opposed to point-wise data.

The number of energy groups needed and the location of the group boundaries are determined by the lattice physics code’s range of application. For example, analysis on mixed-oxide fuel requires different energy detail than does analysis on Uranium-oxide fuel; analysis on fast reactors requires different energy detail than does analysis on thermal reactors; analysis on light water reactors requires different energy detail than does analysis on heavy water reactors, and so on. From the early WIMS days through the1980s, the cross section libraries associated with production-level lattice physics codes were limited to 100 energy groups or fewer.

The original WIMS library contained 69 energy groups and many early lattice physics codes replicated that structure. In the early 1990s, HELIOS was released with a master library containing 190 energy groups, although this group structure was rarely used for analysis. Instead, a smaller library with 35 groups was routinely utilized. Today, it is common for libraries to contain hundreds of energy groups. [10] There are two generations of physical codes used to solve neutron transport problems. A few first generation lattice codes based of the four-factor formula are still in production use today.

The second generation lattice codes features a consistent multi-group (between 50 and 400 groups) representation of the neutron energies.
The main components of a typical second generation lattice code are the following:

- Library access and temperature interpolation.

- Resonance self-shielding calculation.

- Main flux calculation.

- Homogenization and condensation of the reaction rates

- SPH factor calculation.

- Isotopic depletion calculation; figure (1) illustrates the main components of a second generation lattice code [9]

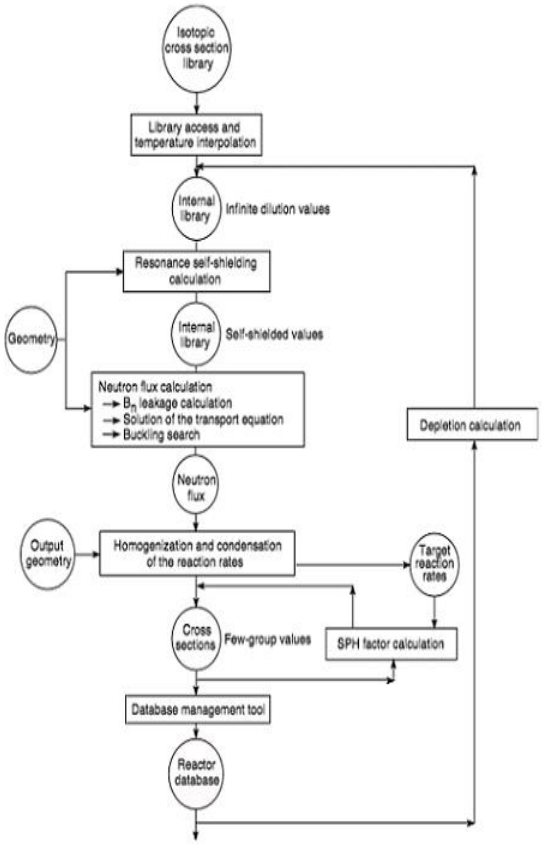


Fig .1. The main components of a second generation lattice code

For realistic systems with complex geometry and detailed energy dependency, transport equation can only be solved by using numerical methods implemented in the lattice codes. Those calculations go through several levels before the final solution is obtained. Each level of the calculation has its own characteristics, and any simplification at any step can lead to a poor final result. [2] In this paper, we focus on the self-shielding step as it is responsible of the biggest inaccuracy in any lattice code used to solve the multi-energy transport equation.

A self-shielding model is required in any lattice code in order to take into account the resonant behavior of the cross sections. [3] Self- Shielding is an algorithm that produces an average cross-section defined over the complete energy domain of the neutrons in a nuclear reactor. The microscopic self-shielded cross section for any reaction ρ in group g ( σ_(ρ,g)^(˜ )) which is defined in equation (1) as:

$$\tilde{\sigma}_{\rho,g}=\mu_g \frac{\int_{U_{g-1}}^{U_g} du\ \sigma_\rho\ \Phi(u)}{\int_{U_{g-1}}^{U_g} du\ \Phi(u)} \quad (1)$$

where u is the lethargy (= ln(E0/E)), ug is the upper lethargy limit of group g, μg is the SPH factor obtained from the multi-group equivalence procedure, φ(u) is the neutron flux inside the resonances and σρ(u) is the microscopic cross section for nuclear reaction ρ. [3],[5] The development of methodologies used to treat resonance self-shielding with respect to energy was improved from assuming that the resonant material was purely absorbent to the improvement made by applying the approach used by Michel Livolant and his doctoral student Françoise Jeanpierre in the late 1960s.

This improvement allowed for scattering and slowing down by the resonant material, which affected self-shielding factors, heterogeneous-homogeneous equivalence, Dancoff effect and Doppler Effect. This theory was introduced at that time in the first version of APOLLO. With the new developments introduced in Version 2 of this code, there was renewed interest in the theory of resonant absorption of neutrons, because the inadequacies of Livolant and Jeanpierre's original theory had been clearly identified. [8]

Two resonance self-shielding shielding models are commonly applied [2]:

- Model based on equivalence and dilution:

This approach is based on the rational expansion of fuel to fuel collision probabilities, either in closed or open cell (or assembly). For infinite and homogeneous problems, each self-shielded cross section of each resonant isotope is tabulated against the dilution parameter.

For heterogeneous problems, a heterogeneous resonant situation is replaced with a linear combination of homogeneous resonant problems. In its simplest form, this technique reduced to the use of Bell and Dancoff factors. This kind of model is implemented in the deterministic code WIMS-D5. The extension of this model has been proposed by Stamm'ler and Abbate (PHOENIX code) and later by Hébert and Marleu (DRAGON code, SHI module) [11].

It is known as a generalized Stamm'ler model (GSM). To achieve better accuracy with the GSM model, two additional improvements are implemented in the Dragon code:

- Use of the Nordheim distributed self -shielding effects in a fuel rod (Level=1 in DRAGON code)

- The Nordheim distributed self-shielding model and use of the Riemann integration method (Level=2).

## II. MATERIALS AND METHOD

### A. *Physical Model (Case study)*

To perform cell calculations, we selected our physical to be the initial composition of the seed fuel pin used Thorium/Uranium fuel. This design uses Whole assembly Seed and Blanket, where individual seed and blanket regions each occupy one full size PWR assembly. Figure 2 illustrates SBU and WASB Assembly Configuration & Pin Cell Models of Seed and Blanket. The initial composition and the design parameters are illustrated in tables 1, 2 respectively.[4]

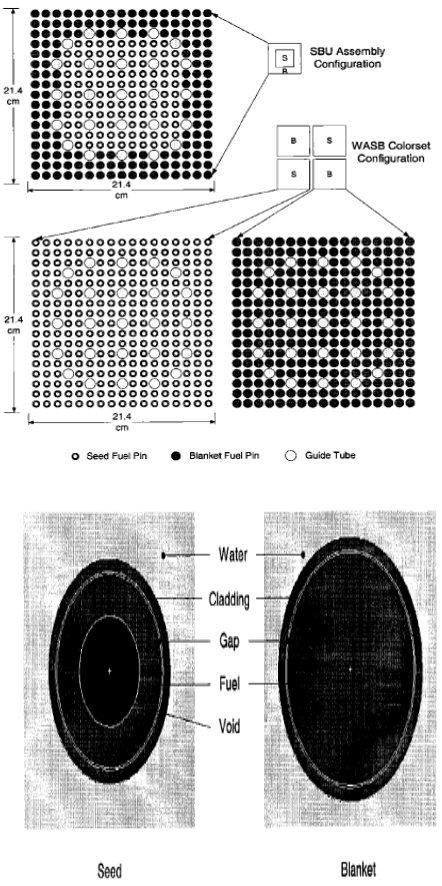


Fig .2. SBU and WASB Assembly Configuration & Pin Cell Models of Seed and Blanket

Table 1. Initial Isotopic Composition

| | | Nuclide | Weight Percent w/o | Number Density 1/ cm$^3$ |
|---|---|---|---|---|
| Fuel | Seed | U-234 | 0.1410 | 3.73911E+19 |
| | | U-235 | 17.6300 | 4.65393E+21 |
| | | U-238 | 70.3790 | 1.83438E+22 |
| | | O-16 | 11.8500 | 4.59793E+22 |
| | Blanket | Th-232 | 79.0955 | 1.92206E+22 |
| | | U-234 | 0.0071 | 1.71054E+18 |
| | | U-235 | 0.8814 | 2.11441E+20 |
| | | U-238 | 7.9326 | 1.87893E+21 |
| | | O-16 | 12.0834 | 4.26071E+22 |
| Coolant, $H_2O$ | | H-1 | 11.19 | 6.66295E+22 |
| | | O-16 | 88.81 | 3.33339E+22 |
| Cladding Zircaloy-4 | | O | 0.125 | 3.08281E+20 |
| | | Cr | 0.100 | 7.58663E+19 |
| | | Fe | 0.210 | 1.48338E+20 |
| | | Zr | 98.115 | 4.24275E+22 |
| | | Sn | 1.450 | 4.81835E+20 |

Table 2. Pin Cells design Parameters

| Parameter | Value | |
|---|---|---|
| | Seed | Blanket |
| Fuel Pellet Radius, cm | 0.22-0.385 | 0.4646 |
| Cladding Inner Radius, cm | 0.3932 | 0.4728 |
| Pin Pitch, cm | 1.26 | 1.26 |
| Fuel Density, g/cm$^3$ | 10.3024 | 9.3624 |
| Coolant Density, g/cm$^3$ | 0.977 | |
| Cladding Density, g/cm$^3$ | 6.55 | |
| Power Density, kW/liter | 129.58 | 79.42 |

### B. *Self-shielding calculations*

The neutronic calculations were carried out using DRAGON code with DRAGON library (or WIMS-5D library) and two self-shielding modules (SHI for equivalence and dilution method, USS for sub-group method) to calculate Kinf. The results are then benchmarked using the well-known Monte Carlo MCNP5 code. Hence, various self-shielding modules and data libraries are tested. Also, the sensitivity to various self-shielding deterministic approximations are tested.

### C. *Nuclear Codes Used for Analysis*

Two different classes of codes are available in respect to solving transport equation: stochastic and deterministic models. In our study we used stochastic models in MCNP, which are the most accurate but

also the most time consuming, as well as the deterministic code DRAGON Version 4. The nuclear codes used for cell calculations are:

- MCNP5 [6]

It is a general-purpose Monte Carlo N–Particle code that can be used for neutron, photon, electron, or coupled neutron/photon/electron transport, including the capability to calculate Eigen values for critical systems. The code treats an arbitrary three-dimensional configuration of materials in geometric cells bounded by first- and second-degree surfaces and fourth-degree elliptical tori. It was tested on several criticality benchmarks, so it is verified to be a reliable and accurate code.

Therefore, in this paper obtained value kinf of fresh seed fuel is set to be the reference value of all deterministic calculations, because it has point wise consentaneous energy data that eliminates self-shielding issue of multi-groups.

- DRAGON [7], [2]

The computer code DRAGON contains a collection of models, which can simulate the neutronic behavior of a unit cell or a fuel assembly in a nuclear reactor. It includes all of the functions that characterize a lattice cell code, namely: the interpolation of microscopic cross sections which are supplied by means of standard libraries; resonance self-shielding calculations in multidimensional geometries; multi-group and multidimensional neutron flux calculations which can take into account neutron leakage; transport or transport-diffusion equivalence calculations as well as editing of condensed and homogenized nuclear properties for reactor calculations, and finally isotopic depletion calculations.

The code DRAGON contains a multi-group iterator conceived to control a number of different algorithms for the solution of the neutron transport equation. Each of these algorithms is presented in the form of a one group solution procedure where the contributions from other energy groups are included in a source term. The current version of DRAGON contains many such algorithms.

The SYBIL option which solves the integral transport equation using the collision probability method for simple one dimensional (1–D) geometries (either plane, cylindrical or spherical) and the interface current method for 2–D Cartesian or hexagonal assemblies. The EXCELL option solves the integral transport equation using the collision probability method for general 2–D geometries and for three-dimensional (3–D) assemblies. The MCCG option solves the integro-differential transport equation using the long characteristics method for general 2–D and 3–D geometries. The execution of DRAGON is controlled by the generalized GAN driver. [2] Version 4 is a new distribution of the reactor physics computer code sat GAN .Its components are:

DRAGR module in NJOY, GAN lib tools, and Modules (calculation operators) of the following codes:

- Dragon: lattice code
- Trivac: reactor (full core) code Donjon: simulation of reactor operation
- Optex: reactor design optimization.[9]

*D. Nuclear Data Libraries used*

In order to obtain the final solution, isotopic cross section libraries are required. We used the following libraries:

- 172 energy group WIMS-D5 library format based on ENDF/B-VII.0 evaluated nuclear data library (WIMS-D5)
- 172group DRAGON-lib library format based on ENDF/B-VII.0 evaluated nuclear data library (DRAGON 4)
- A compact ENDF ACE format in MCNP5 based on ENDF/B-VII.0

*E. Deterministic Methods used*

In this paper, we study sixteen deterministic methods; each method is given an identification number from 1 to 16. All these methods are implemented using the same lattice code-DRAGON Version4- and calculated for the same number of energy groups (172), but they have different tracking modules and data libraries( DRAGON-lib/ WIMS-D5-lib).

These sixteen deterministic methods are:

- One dimensional self-shielding calculations are made using the equivalence and dilution method to obtain the microscopic cross-sections of isotopes with resonance present in the material without using neither Norhdeim distribution nor Rimman integration improvements on the equivalence and dilution method (SHI level=0 in DRAGON code). Also, Livolant and Jeanpierre normalization scheme (which modifies the self-shielded averaged neutron fluxes in heterogeneous geometries) is not used (NOLJ module in DRAGON code). The data library used is 172 energy groups WIMS-D5 library.

- One dimensional self-shielding calculations are made using the equivalence and dilution method with Norhdeim distribution improvement (SHI Level=1 module). But without Livolant and Jeanpierre normalization scheme (NOLJ). The data library used is 172 energy groupsWIMS-D5 library.

- One dimensional self-shielding calculations are made using the equivalence and dilution method using both Norhdeim distribution and Riemann Integration improvements (SHI Level=2 module). But Livolant and Jeanpierre normalization scheme (NOLJ module) is not used. The data library used is 172 energy groups WIMS-D5 library.

- One dimensional self-shielding calculations are made using the equivalence and dilution method with resonance present in the material without using any improvements on the equivalence and dilution method (SHI Level=0 module). Livolant and Jeanpierre normalization scheme (LJ module) is used. The data library used is 172 energy groups WIMS-D5 library.

- One dimensional self-shielding calculations are made using the equivalence and dilution method with using Norhdeim distribution improvement (SHI Level =1 module). Livolant and Jeanpierre normalization scheme (LJ module) is used. The data library used is 172 energy groups WIMS-D5 data library.

- One dimensional self-shielding calculations are made using the equivalence and dilution method using Norhdeim distribution and Rimman integration improvements (SHI Level=2 module). Livolant and Jeanpierre normalization scheme (LJ module) is used. The data library used is 172 energy groups-DRAGON library.

- One dimensional self-shielding calculations are made using the equivalence and dilution method without any improvements on the equivalence and dilution method (SHI Level=0 module). Livolant and Jeanpierre normalization scheme is not used (NOLJ module). The data library used is 172 energy groups- DRAGON library.

- One dimensional self-shielding calculations are made using the equivalence and dilution method Norhdeim distribution improvement (SHI Level=1 module). Livolant and Jeanpierre normalization scheme is not used (NOLJ module). The data library used is 172 energy group DRAGON library.

- One dimensional self-shielding calculations are made using the equivalence and dilution method with using Norhdeim distribution and Rimman integration improvements (SHI Level=2 module). Livolant and Jeanpierre normalization scheme is not used (NOLJ module). The data library used is 172 energy groups DRAGON library.

- One dimensional self-shielding calculations are made using the equivalence and dilution method without using any improvements on the equivalence and dilution method (SHI Level=0 module) with the Livolant and Jeanpierre normalization scheme (LJ module). The data library used is 172 energy groups DRAGON library.

- One dimensional self-shielding calculations are made using the equivalence and dilution method using Norhdeim distribution improvement (SHI Level=1 module). Livolant and Jeanpierre normalization scheme (LJ) is used. The data library used is 172 energy groups DRAGON library.

- One dimensional self-shielding calculations are made using the equivalence and dilution method using Norhdeim distribution and Rimman integration improvements Livolant and Jeanpierre normalization scheme (LJ) is used. The data

library used 172 energy groups- DRAGON library.

- Two dimensional Self-shielding calculations are made using the subgroup with 172 energy groups WIMS-D LIB calculations with SUBG method where the physical Probability Tables are computed using the RMS approach (root-mean-square).

- Two dimensional Self-shielding calculations are made using the subgroup with 172 energy groups WIMS-D LIB calculations with PTSL method where mathematical probability tables and slowing down correlated weight matrices can be computed in selected energy groups using the Ribbon extended approach.

- Two dimensional Self-shielding calculations are made using the subgroup with 172 energy groups DRAGON calculations with SUBG method where the physical probability tables are computed using the RMS approach (root-mean-square).

- Two dimensional self-shielding calculations are made using the subgroup with 172 energy groups DRAGON library calculations with PTSL method where mathematical probability tables and slowing down correlated weight matrices can be computed in selected energy groups using the Ribbon extended approach.

These sixteen deterministic methods are summarized in table 3 as follows:

Table 3. Sixteen self-shielding methodologies which are applied and tested

| Method Number | Code Used | Tracking Module | Data Library |
|---|---|---|---|
| 1 | DRAGON | SYBIL+SHI (L=0) NOLJ | 172gWIMS-D |
| 2 | DRAGON | SYBIL+SHI (L=1) NOLJ | 172g WIMS-D |
| 3 | DRAGON | SYBIL+SHI (L=2) NOLJ | 172gWIMS-D |
| 4 | DRAGON | SYBIL+SHI (L=0) LJ | 172gWIMS-D |
| 5 | DRAGON | SYBIL+SHI (L=1) LJ | 172g WIMS-D |
| 6 | DRAGON | SYBIL+SHI (L=2) LJ | 172g WIMS-D |
| 7 | DRAGON | SYBIL+SHI (L=0)NOLJ | 172g DRAGON |
| 8 | DRAGON | SYBIL+SHI (L=1) NOLJ | 172g DRAGON |
| 9 | DRAGON | SYBIL+SHI (L=2) NOLJ | 172g DRA GON |
| 10 | DRAGON | SYBIL+SHI (L=0) LJ | 172g DRAGON |
| 11 | DRAGON | SYBIL+SHI (L=1) LJ | 172g DRAGON |
| 12 | DRAGON | SYBIL+SHI (L=2) LJ | 172g DRA GON |
| 13 | DRAGON | USS-SUBG | 172gWIMS-D |
| 14 | DRAGON | USS-PTSL | 172gWIMS-D |
| 15 | DRAGON | USS-SUBG | 172g DRAGON |
| 16 | DRAGON | USS-PTSL | 172g DRAGON |

## III. RESULTS AND DISCUSSION

### *A. MCNP5 Result of $K_{inf}$*

For our physical Model (fresh seed cell used in thorium/uranium PWR) using MNCP 5 the resultant value of Kinf is 1.66440 with an estimated standard deviation of 0.00061. Figure (2) illustrates the MCNP5 simulation of fresh seed fuel cell drawn by Vised program, the design parameters are as indicated in tables 1 and 2.

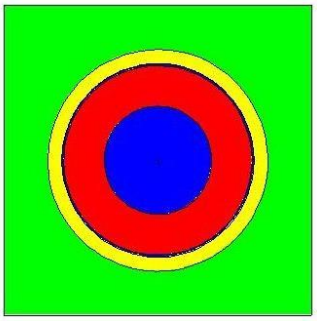

Fig .3. MCNP5 seed cell simulation

In this paper, sixteen deterministic self-shielding methods are tested by comparing the resultant Kinf with the reference value obtained by MCNP5 code. The resultant values of Kinf for different modules in DRAGON code compared with the reference MCNP value of Kinf illustrated in figure 3, show that of the sixteen tested deterministic methods the best are methods of numbers 11, 12, 15, and 16. In these methods, high accuracy was that found using the equivalence and dilution method with Norhdeim distribution and Rimman integration improvements and Livolant and Jeanpierre normalization scheme with 172 energy groups DRAGONLIB. (Method number 12)The absolute error in this case is 0.000662 which is approximately the standard deviation of the Monte Carlo result of Kinf. Also, using subgroup method with SUBG with 172 energy groups DRAGON-library is also a very sufficient self-shielding treatment as its absolute error is only 0.001039.

### *B. Sensitivity of Kinf to various deterministic modules and data libraries*

Using DRAGON code the sensitivity of the obtained value of Kinf to the various modules and data libraries is studied to identify the impact of each of these modules on the efficiency of the self-shielding treatment used.

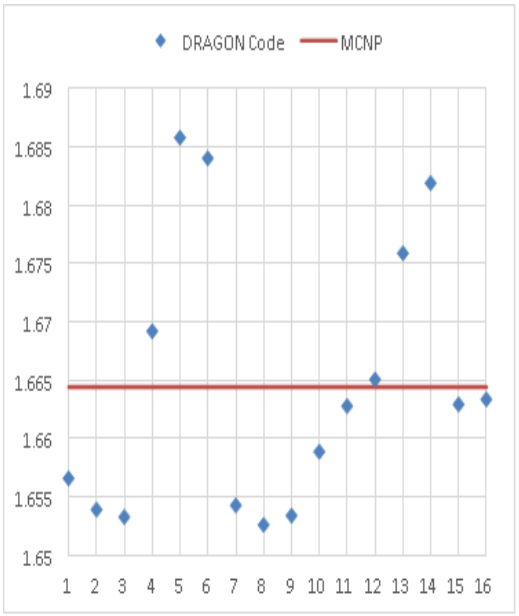


Fig .4. $K_{inf}$ as obtained by the different 16 self-shielding methodologies implemented in DRAGON code Vs. the reference value obtained by MCNP5 code.

### *C. Effect of changing data libs. on SHI module*

The resultant values of Kinf in case of using equivalence and dilution method for various deterministic options are LJ/NOLJ, L=0 / L=1 / L=2. These calculations are performed using DRAGON-LIB/ WIMS-LIB is illustrated in Fig 4. It shows that there is advantage of using DRAGON-LIB with the equivalence and dilution method of self-shielding treatment.

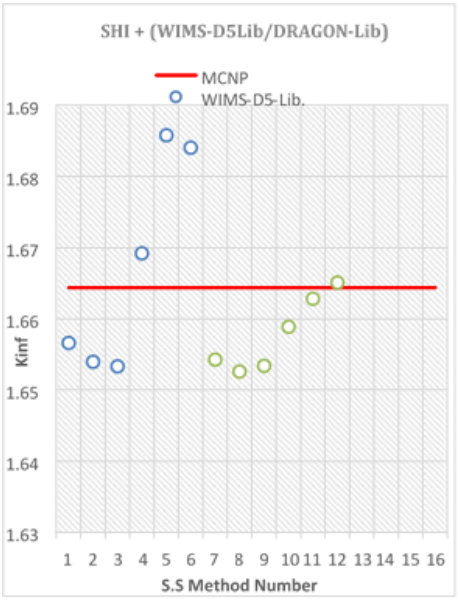


Fig .5. $K_{inf}$ using SHI module for WIMS-D5 Lib implemented in DRAGON code Vs. The reference value obtained by MCNP5 code

The results show that a high accuracy is obtained using DRAGONLIB especially with the LJ module.

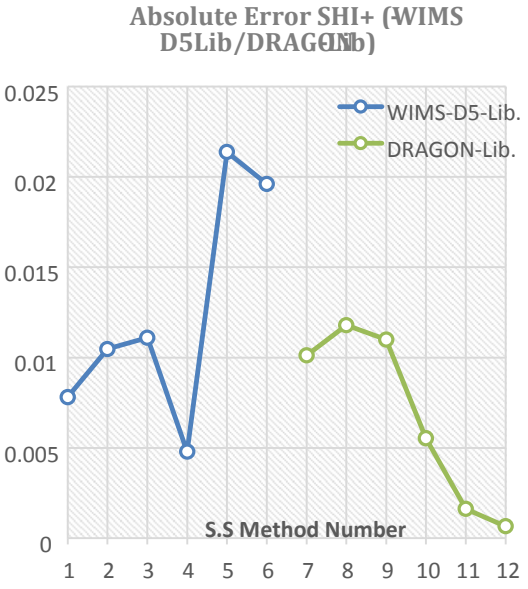


Fig .6. Absolute error in Kinf using SHI module for different Libraries

The absolute error in Kinf for SHI with DRAGON library and WIMS-D5 library (figure 5) may be divided into two periods:

- Period contains points (1, 2, 3, 4) for WIMS-D5 library and points (7, 8, 9, 10) for DRAGON library.

- Period contains points (5, 6) for WIMS-5D-library and points (7,8) in DRAGON library.

It is observed in Fig.5 that the effect of changing Data-lib in the first period is trivial, but the effect of changing Data-lib in the second period has a significant effect. Hence, it can be concluded that for equivalence and dilution and without Livolant and Jeanpierre normalization scheme (SHI + NOLJ), changing between WIMS-D5 and DRAGON libraries has a trivial effect. However, the results also show that DRAGON library is more suitable with Livolant and Jeanpierre normalization scheme (SHI +LJ) than WIMS-D5 library as higher accuracy is obtained.

### *D. Effect of changing data libraries on USS module*

The use of DRAGON-LIB and WIMS-LIB for self-shielding using sub-group method and SUBG / PTSL modules (Fig. 6) illustrates that the result of Kinf using DRAGON library results in higher accuracy than WIMS-SD library.

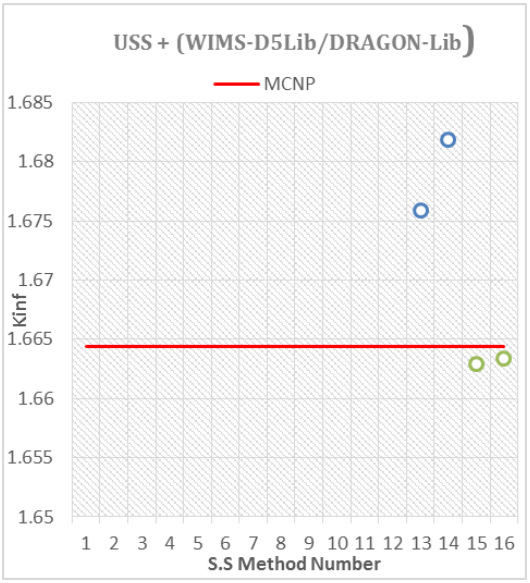


Fig .7. Kinf results of using USS module for WIMS-D5/DRAGON data libraries compared with reference value obtained by MCNP5

### *E. Effect of Norhdeim Distribution and Rimman Integration Improvements*

In figure 7, the effect of Norhdeim distribution alone (level=1), normalization distribution and Riemann Integration improvements (level=2) on improving the results of a deterministic module uses equivalence and dilution method with LJ normalization scheme. The effect was tested by comparing the resultant value of Kinf in each case with the reference value obtained using the Monte Carlo MCNP5 code.

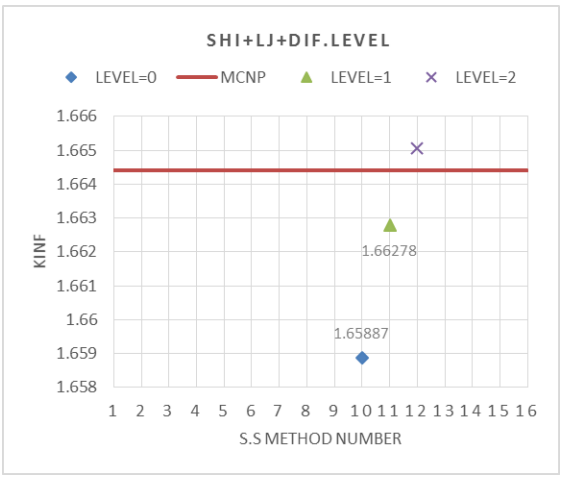


Fig .8. $Ki_{nf}$ using SHI module for different improvement levels compared with the reference value obtained by MCNP5

The results show that the improvements are sufficient to obtain higher accuracy. Where level=0 stands for the case in which neither Norhdeim distribution nor Riemann integrations are used.

## IV. CONCLUSION

For our standard fresh seed fuel, the following items are concluded:

- For equivalence and dilution self –shielding method without applying Livolant and Jeanpierre normalization scheme (SHI + NOLJ), changing between WIMS-D5 and DRAGON data libraries has a trivial effect. However, the results also show that DRAGON library is more suitable if Livolant and Jeanpierre normalization scheme is used (SHI +LJ) than WIMS-D5 library, as higher accuracy is obtained.

- Using equivalence and dilution method with Norhdeim distribution and Rimman integration improvements and Livolant and Jeanpierre

normalization scheme is a very sufficient deterministic method to deal with resonant self-shielding problem for fresh fuel – even with a relatively high U-235 concentration, as the absolute error is only 0.000662 .

- Using sub-group method with PTSL or SUBG with 172 energy groups DRAGON library method shows high accuracy for fresh fuel.

- Norhdeim distribution and Rimman integration improvements are sufficient to improve the results id DRAGON library is used.

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