PHYSOR 2004 -The Physics of Fuel Cycles and Advanced Nuclear Systems: Global Developments
Chicago, Illinois, April 25-29, 2004, on CD-ROM, American Nuclear Society, Lagrange Park, IL. (2004)

The Nuclear Heating Calculation Scheme For Material Testing in the Future Jules Horowitz Reactor

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The calculation of nuclear heating is an important issue for the design of the future material testing reactor called "Jules Horowitz Reactor" (JHR) and its test device.

In order to ensure a heat gradient below 5°C across the material testing sample, the deposited energy, mainly due to photon interaction, has to be calculated in various irradiation devices with an accuracy better than 15%.

To achieve this aim, an innovative heating calculation scheme has been carried out at CEA. The classical scheme chaining deterministic and stochastic code is modified. First a heterogeneous gamma source calculation is performed at assembly level using the deterministic code APOLLO2, followed by a Monte Carlo gamma transport calculation in the whole core using the TRIPOLI4 code. The calculated gamma sources at the assembly level are applied in the whole core simulation using a weighting based on power distribution obtained from the neutronic core calculation.

These gamma sources at the assembly level can be used in various core configurations at the Monte Carlo gamma transport step without repeating the complete photonic calculation.

This new scheme has been motivated by the special geometry of the JHR assembly. Moreover, the neutronic code scheme is fully validated whereas previous schemes determine gamma source on homogeneized core resolution without intermediate neutronic validation.

This new heating scheme has been validated by comparison with a 3D Monte Carlo calculation in coupled neutron-gamma mode, which gives a result without any assumption (except concerning the nuclear data used), but with a run-time consuming 4 to 5 times greater to reach the same precision.

The discrepancies between the developed scheme and the reference calculation for a simple experimental device (single CHOUCA) are below 3%, which is close to the Monte Carlo statistical uncertainty.

A series of mock-up experiments in the EOLE research reactor is going to furnish a set of gamma dose measurements, allowing to consolidate the validation.

KEYWORDS: gamma, heating, MTR, JHR

1. Introduction

The simulation of materials damage under irradiation is a key point in the definition of the

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next reactor generation and the usual “cook and look” process has become insufficient. In order to achieve these objectives, a new reactor called "Jules Horowitz Reactor" intended for the studies of radiation damage to materials and is currently under design at CEA. The experimental control of sample irradiation conditions requires knowledge and understanding of nuclear heating due to both neutron and photon flux. The heat gradient has to remain lower than 5°C across the sample. That is why a calculation scheme of nuclear heating is being developed by CEA’s Nuclear Energy Division and is going to be validated by means of small-scale mock-up experiments in the EOLE research reactor. These features are included in the HORUS 3D code package which is being used for the design studies of the Jules Horowitz Reactor.

2. General Background

2.1 JHR description

The “Jules Horowitz Reactor” is the future European Material Testing Reactor [1]. The design studies are split into different phases. Preliminary design studies allowed to define the main options of the core and its elements, and to characterize a reference configuration. The Detailed Design Studies will optimise the JHR performances by testing modifications from the reference configuration.

Its assembly would be composed of 18 cylindrical fuel plates maintained by 3 stiffeners (Fig. 1). The central cavity can host either aluminium filler, a hafnium control rod or an irradiation test device.

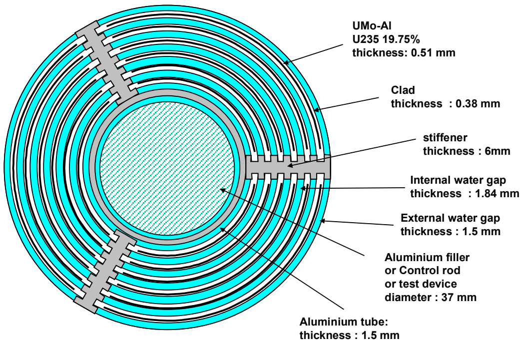

Figure 1 : Cross-section of the JHR assembly

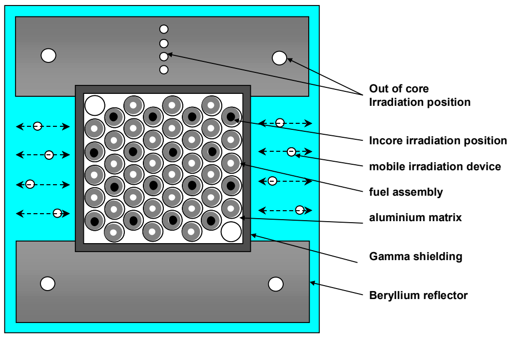

Figure 2 : Cross-section of the JHR core

The JHR will be cooled and moderated by light water. The preliminary core consists of 46 assemblies (Fig. 2), arranged in a hexagonal lattice inside a rectangular aluminium matrix. It is boarded on two sides by a beryllium (Be) reflector. The other two sides are left free (water) in order to introduce mobile irradiation devices.

Two kind of experimental device are more particularly used inside the core:

- Single CHOUCA which are modelized by substituting the central aluminium cavity of a standard fuel assembly (Fig. 1) by a cylinder containing experimental device cooled by NaK.
- Clustered CHOUCA that consist of the substitution of the whole standard fuel assembly by three cylinders containing three samples cooled by NaK. (Fig. 3)

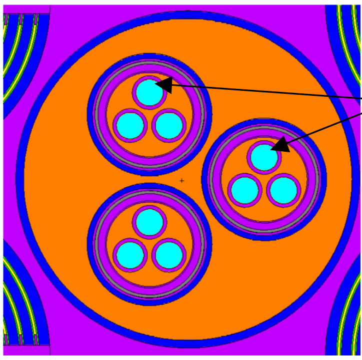

Figure 3 : Clustered CHOUCA among standard fuel assemblies

The gamma heating calculation in sample is validated on various core configurations with these experimental devices.

An adapted and consistent neutronics/thermal-hydraulics code package, named HORUS3D (HOrowitz Reactor simulation Unified System) is developed in order to fulfil the needs of the Detailed Design Studies [2].

2.2 Objectives of the calculation scheme

Nuclear heating of samples has two sources:

- It can be due to photons, coming from fission, capture and inelastic scattering that transfer their energy to electrons by means of pair creation, photo-electric and compton interaction.
- It can also be due to neutrons which transfer their energy after collision by nucleus recoil or by slowing down of charged particles emitted by the nucleus (α,p). However, this contribution represents only about 10% of the total heating.

For that reason, the main part of the work deals with the assessment of photon heating.

In order to minimize the heat gradient inside the sample, gamma heating has to be known with an accuracy of at least 15%.

In addition, the HORUS 3D photonic code package has to fulfil high flexibility requirement to fit easily every kind of core configuration and to allow fast and accurate calculations.

To reach this goal, emphasis has been put on the constitution of a set of libraries at assembly level, including geometry data, gamma source, gamma response functions and homogenization choices. This innovative approach allows to calculate the heating in the whole core simply by picking suitable data in the various libraries without carrying out the whole core calculation.

3. Computational scheme description

After a brief presentation of the neutronics calculation scheme, we are going to detail the photonic part.

3.1 Neutronic calculation scheme

The assembly and core neutronics calculation is a preliminary step before the photonic calculation.

The JHR assembly neutronics route is performed with the APOLLO2 [3] code and its 172-group library. The complex geometry (Fig. 1) necessitates the use of the exact 2D collision probability method. The optimized slowing-down model with a Doppler cross-section broadening for the first resonance, developed for PWR UOX fuels, can be used. The cross sections are homogenized on assembly level and collapsed on an adapted 6-group energy mesh.

The 3D core neutronics route uses the new finite elements developed in the CRONOS2 code [4] . The whole core is split into 145,000 prisms and the CRONOS2 code allows to reconstruct power and 6-group flux distribution inside all of them.

A fully validation against reference Monte Carlo calculation has demonstrated the great confidence of this scheme. The error on the calculated keff is lower than 200 pcm ans the assembly power map is determined within 2 % [7].

3.2 Photonic calculation scheme

The complete photon calculation scheme is shown on Figure 4.

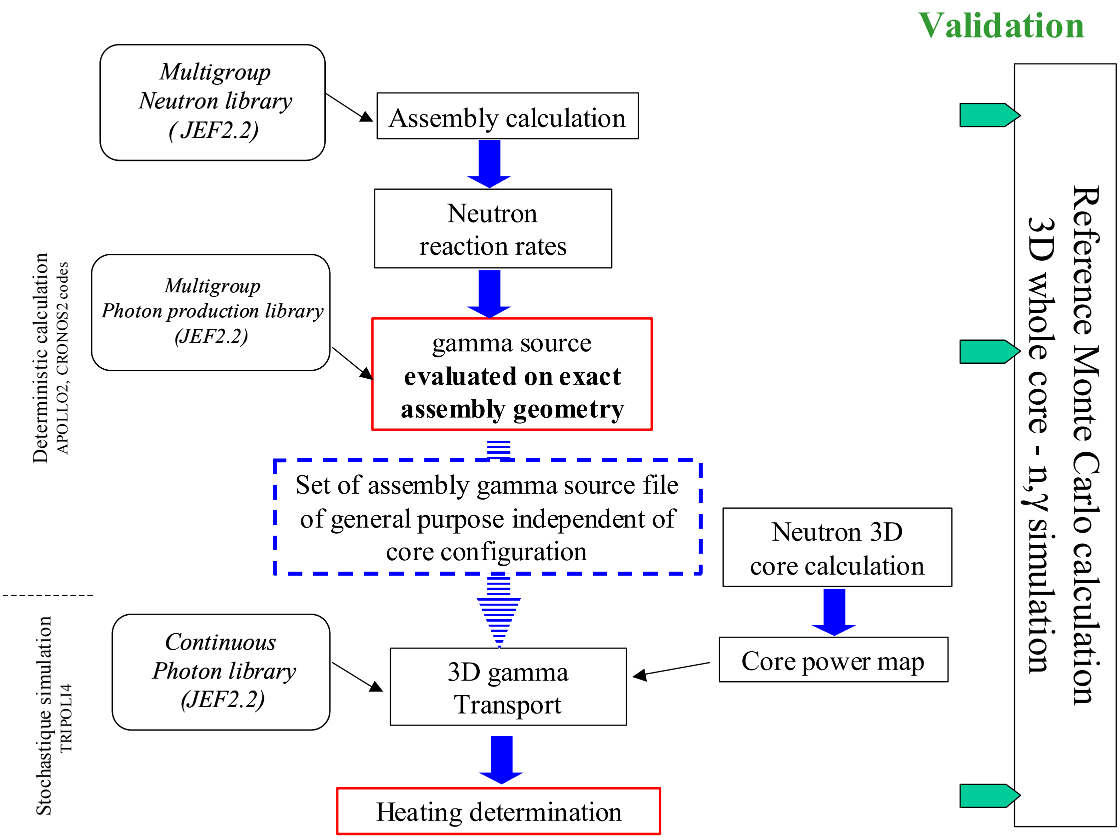

Figure 4 : The photon calculation scheme

The photon calculation scheme so defined is divided in several steps. Its point of departure is the calculation of the neutron flux in fuel and irradiation device assemblies with the help of the deterministic code APOLLO2 [3], following the standard calculation route.

A new set of procedures has been implemented in APOLLO2 [3] to allow the evaluation of the heterogeneous gamma source distribution in the complete assembly geometry. The originality of this method is the use of the exact assembly geometry up to the gamma source calculation level. There are no neutronic spatial homogenization or segmentation assumptions.

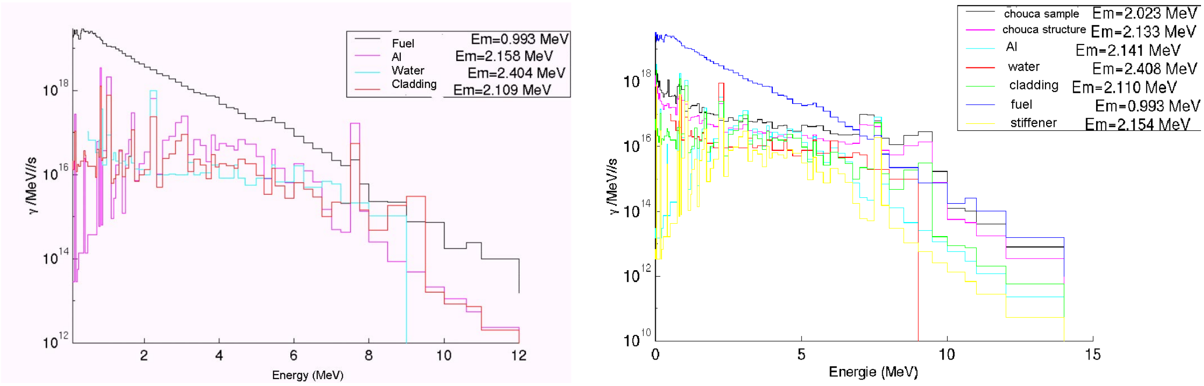

Figure 5 : Example of gamma source spectra in fuel (left) and single CHOUCA (right) assembly with average energy

An example of the obtained gamma spectra inside a standard fuel and single CHOUCA assembly is shown on Figure 5.

The importance of the fuel's fission photons, with an average energy of about 1 MeV, is obvious. Its source rate is several orders of magnitude higher than photon production of the other materials in the standard assembly (capture, inelastic scattering) . Above 7 MeV, the different spectra come closer but the source intensity is negligible in comparison to that at lower energy. Notice the 7.5 MeV peak due to $^{27}Al(n,\gamma)$.

Using these spectra, we can define a homogenization choice for the gamma source on the basis of gamma mean free path within the assembly (Fig 6).

Independently of the neutron energy and reaction which cause the gamma production, there is no significant differences of photon flux if reactions occur closer than about one photon mean free path. At the average gamma fission energy (1 MeV), the gamma mean free path is of about 3 cm, of the order of the assembly size (Fig 7).

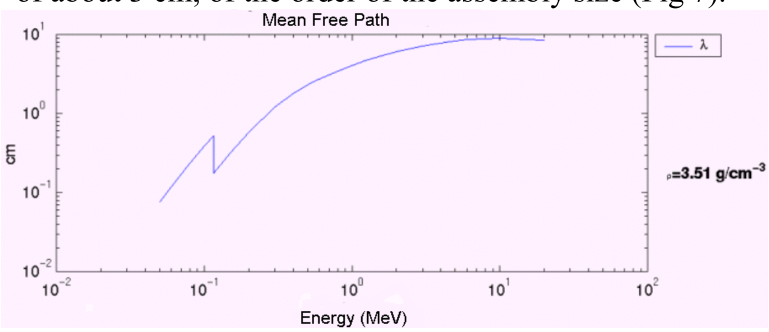

Figure 6 : Gamma mean free path in standard fuel assembly

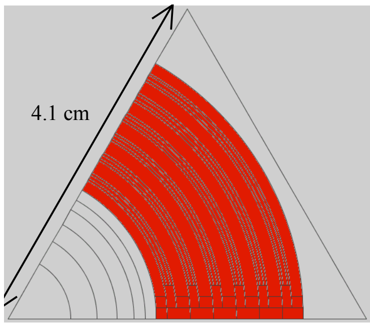

Figure 7 : Example of γ-source homogenization in assembly fuel area (red)

Similar arguments allow defining analogous homogenization choices for the other assemblies types.

At the end of this first step, a set of gamma sources for each assembly type has been created. They are homogenized according to mean free path in various assembly material, normalized to unit power and ready to be combined into a given core configuration with experimental devices.

Specific procedures allow the insertion of these photon sources in an input deck for the

TRIPOLI4 3D Monte Carlo code, used in photon transport mode. The source is spatially weighted by the 2D power distribution coming from a core calculation using the CRONOS2 deterministic code. The Monte Carlo transport of photons of about 1 MeV inside the reactor is a fast process, compared to neutron transport.

The importance of fission in gamma production justifies the use of the core power distribution obtained from 2D whole core neutron calculation to weight assemblies gamma sources before inserting them.

The axial source shape is modelized as identical to axial power shape, which is considered to be the same for all assemblies.

We can modify the experimental device arrangement and repeat the gamma transport inside the core without starting again the assembly source calculation. The only need is the 2D power distribution per assembly.

Heating can be evaluated using adapted gamma response functions in miscellaneous materials including carbon, stainless steel and aluminum.

The photonic procedures use a set of libraries describing gamma sources, assembly geometry, assembly homogenization choices and heating response functions which can be combined in order to deal easily with various core configurations.

3.3 Validation on a configuration with 8 single CHOUCA devices

The calculation scheme has been validated by comparison to a direct calculation using the TRIPOLI4 [5] Monte Carlo code in coupled neutron-gamma mode. This calculation allows to obtain heating reference values without any hypotheses about spatial homogenization, energy groups division or self-shielding but to the detriment of computer time which is far longer (about 4 to 5 times). The two calculations use JEF2.2 nuclear data.

The scheme has been validated using different arrangements of the irradiation devices inside the core. In the following, we give the case of a core equipped with eight single CHOUCA devices. Sample testing devices are inserted in place of the central aluminium cylinder of the standard fuel assembly. We calculate the gamma heating in the sample part of the experimental device using a stainless steel gamma response function.

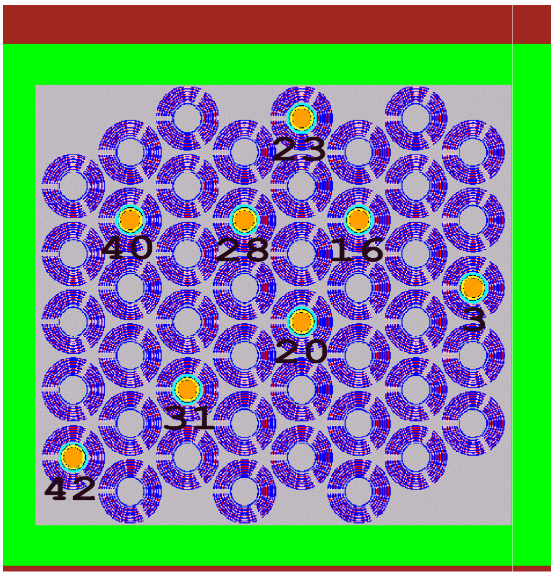

Figure 8 : Geometric arrangement with Single CHOUCA

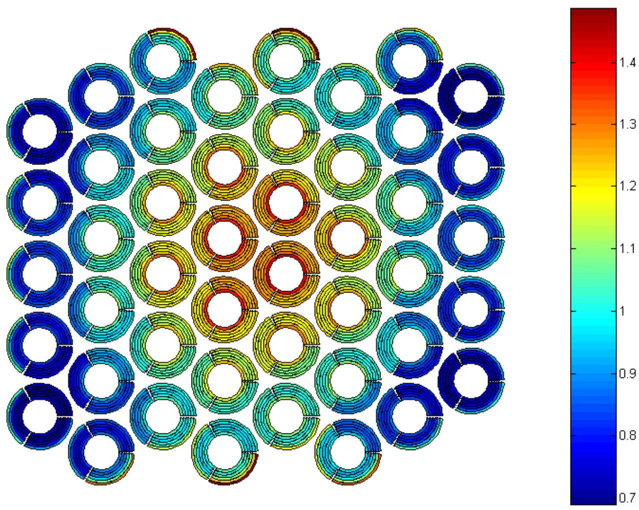

Figure 9 : Fine power distribution per plate

	Reference TRIPOLI4 n,γ		APOLLO2+TRIPOLI4 γ		Discrepancies
Index	Heating (W/g)	σ (%)	Heating (W/g)	σ (%)	
3	4.36	0.43	4.42	0.93	1.4%
16	6.42	0.35	6.49	0.76	1.1%
20	7.48	0.32	7.46	0.73	-0.3%
23	4.72	0.4	5.09	0.88	7.8%
28	6.89	0.34	6.96	0.73	1.0%
31	6.49	0.34	6.46	0.78	-0.5%
40	5.28	0.38	5.25	0.81	-0.6%
42	3.45	0.48	3.61	1.07	4.6%

Standard deviation	3.3%

Table 1 : Heating discrepancies with TRIPOLI4 coupled (n,γ) calculation

Gamma heating calculations were carried out in the configuration depicted in Figure 8. The results and the differences compared to the reference calculation are given in Table 1. The standard deviation is about 3%. The smallest differences are observed for assemblies situated near the center of the core (16, 20, 28, 31, 40). For devices situated towards the core periphery, in particular near the reflector, (23, 42), the APOLLO2-TRIPOLI4 calculation over-estimates the heating from 5 to 8%. This result can be put in relation to the fine power distribution (Figure 9). The power distribution of the core equipped with 8 single CHOUCA devices remains very similar to that of the standard core. The assemblies located within the core (16, 20, 28, 31,40) are correctly described by the APOLLO2 calculation of source in infinite medium, featurin a power gradient decreasing from the inside to the outside. Assemblies 23 and 42, however, are located near the Be reflectors, their outermost plates close to the reflector present a significant increase in power of almost 20% compared to the other plates. The emission of fission photons follows the same gradient, which is reversed compared to the other assemblies. Taking into account the mean free path of the fission photons in the order of 3 cm for an average energy of about 1 MeV, the internal fuels plates forms a gamma shielding. Consequently, the APOLLO2 calculation of the source term in the assembly center is over-estimated compared to the TRIPOLI4 calculation of reference.

The heating calculation bias near reflector could be improved by using fine power distribution for gamma source weighting instead of assembly power for the specified single CHOUCA assembly, but for this category of core arrangement, the mean standard deviation remains widely negligible in comparison with nuclear data uncertainties detailed in the following paragraph.

3.4 Evaluation and improvement of uncertainties

We estimate that there are large uncertainties on gamma heating of approximately 30% (2σ). They are mainly due to the lack of knowledge about gamma emission yields and spectra (about 20%), and secondarily from unsatisfactory modelling of the gamma response function. The calculation scheme biases previously evaluated are negligible with respect to these errors. However, they stay twice greater than the precision target for sample irradiation of 15%.

In order to reduce these errors and to validate experimentally this calculation scheme, a set of experiments called ADAPh are going to be carried out in the core of the research reactor EOLE in Cadarache during 2004.

EOLE is an experimental reactor of very low power, dedicated to neutronics studies of moderated lattices, in particular those of industrial pressurized water reactors (PWR). Measurements of integrated gamma doses in various places, more or less off-centre of the reactor core are in progress by means of thermoluminescent dosimeters (TLD). The use of TLD is made necessary because of the high dose rate, the high neutron flux and the small size assigned to the detector system. Various types of TLD are going to be tested (CaF_2:Mn, ^{7}LiF:Mg,Ti, Al_2O_3:Na) to select the best adapted one (Figure 12). Activation or saturation phenomena have to be avoided in order to permit measurement interpretation. The contribution of fast and particularly thermal neutrons in thermoluminescent signal has to be corrected by the evaluation of neutron spectra flux.

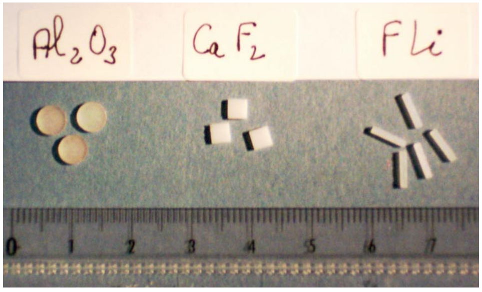

Figure 12 : close vue of a set of different TLD

A total number of about 300 TLD are used. An important part of measurement uncertainty comes from the dispersion of TLD efficiency. In order to reduce this error, an individual sensitivity coefficient is measured for each TLD into a ^{60}Co beam. This factor is combined with the efficiency curve obtained for some TLD representative of a set of TLD of the same type. They are calibrated into ^{60}Co beam for several kerma air values.

Two additionnal correction factors have to be applied to measurement :

- The dose measured must be transfered from the kerma air calibrated detector to specific TLD material.
- Cavity phenomena: Due to the millimeter size of TLD (Figure 12), the measured dose represents the deposited energy within the TLD and coming from surrounding material by electron flux. Electronic equilibrium condition is not fulfiled (input electron flux is not equal to output electron flux) because of the atomic number difference between TLD material and surrounding material, and so a correction factor must be applied to photon kerma calculation.

These factors are obtained from standard dosimetry models and MCNP calculations, they are of the order of some tens of percent. Table2 shows a comparison of dose calculation with various hypothesis for ^{7}LiF into ^{60}Co calibration beam. The use of kerma air calculation leads to overestimate dose in ^{7}LiF by about 25% whereas the calculated value in air corresponds to the measureated value.

Kerma air measurement	Kerma air calculation	Kerma ^{7}LiF calculation	Calculation of Deposited Energy in ^{7}LiF surrounded by plexiglas
214	212	193	167

Table 2 : Comparison of dose (mGy/h) measured and calculated in the ^{60}Co calibration setup

The data measured in EOLE reactor are going to be compared with those given by the Monte Carlo code in coupled neutron-gamma mode calculation. They will be used to provide global factors to adjust gamma production yields of main reactions in order to reduce heating uncertainties, the precise methodology is in progress. First analyzed results are expected at the middle of 2004.

4. Conclusion

In this paper, we describe the part of HORUS 3D code package intended to carry out gamma heating calculation in the JHR core.

It is based on the construction of photon source libraries on assembly level, using the new photonic modules available in APOLLO2. The main specificity of these procedures with regard to previous ones is to carry out the gamma source calculation on the exact assembly geometry.

The same source libraries can be used in various core arrangements, and the only need is to carry out the photon transport inside the core.

These sources are inserted in the core description of a gamma transport code TRIPOLI4 by weighting them with the assembly power distribution coming from neutronic calculation.

This innovative scheme is validated by comparison with coupled Monte Carlo neutron-gamma heating calculation. The difference is of about 3%, but this scheme executes about 4 to 5 times faster.

The heating calculation of experimental device inside reflector is in progress, although it is approximately 10 times weaker than inside the core.

The most important error comes from gamma production nuclear data, it is estimated to about 30%. A set of experimental dose measurement will be carried out during 2004 in order to reduce this uncertainty.

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