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Monte Carlo analysis of the propagation of fusion neutrons in a high enriched uranium system

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ABSTRACT

Monte Carlo methods were used to calculate experimental observables related to the propagation of pulses of fusion neutrons in a compact and highly enriched (90%) subcritical system. These observables are the amplitude and phase of the Fourier transform of the detection rate of a ^{3}He moving detector, they correspond to the propagation of neutron waves excited by a sinusoidal neutron source. The MCNP code was used to model in great details all the heterogeneities of the experimental set up allowing in particular to have a good model of the neutron leakage in the direction perpendicular to the propagation.

The very good results of the comparison with the experimental results contrast with previous comparisons with diffusion and transport theory models. The Monte Carlo modeling allows a full analysis of neutron wave experiment in space, time and energy allowing to define asymptotic regions where global complex wave vector exists. We propose to use the extensive literature of neutron wave experiments for further benchmark of MCNP.

1. Introduction

A considerable amount of experimental data exists in the literature involving the measurement in space and time of the propagation of neutrons pulses in subcritical systems (a very active field in the 1960s and 1970s, see for example Perez and Uhrig (1967)). In most of the cases the source neutrons are produced by the modulation in time (for example as a pulse) of the deuterium–tritium fusion reaction which produces 14 MeV neutrons. In order to reduce the space transients due to the highly penetrating 14 MeV neutrons they are in general moderated and thermalized by an array of materials at one of the borders of the multiplicative system. The Fourier transform of the detector responses as function of the distance to the source produces two set of data, the amplitude and phase of the neutron wave response of the system to a sinusoidal source. Amplitude and phase, functions of two variables, frequency and detector position, show opposite sensitivities to uncertainties (see for example, Difilippo (1977)) in neutron cross sections.

In contrast with the simplifying hypothesis of diffusion and transport models, these experiments can be simulated with extreme accuracy by the Monte Carlo code MCNP (X-5 Monte Carlo Team, 2003) which not only can describe extremely well all the complications of the geometry but also can compute the time transients. Extensive checks of MCNP (Brown, Mosteller, Sood, 2003) calculations involves comparisons with critical configurations, considering the availability of many good neutron wave experimental data plus the particular sensitivity of the observables we propose to add these experiments to the benchmark of MCNP and its cross section data base.

We contribute to this idea with the MCNP analysis of a rare neutron wave experiment performed in a very small 90% uranium enriched MTR (Material Test Reactor) subcritical system. Due to President Eisenhower *Atoms for Peace* policy (Eisenhower, 1953) research reactors were worldwide built with very high enriched uranium, later in the 1980s this policy was changed in favor of much less enriched fuel; therefore the adjective "rare" to the experiment analyzed here. A bombastic (but descriptive) title for this work would then be *Monte Carlo Analysis of the Propagation of Fusion Neutrons in a Weapon Grade Enriched Uranium System*.

2. Diffusion and transport theories description of neutron wave experiments

To the author's knowledge no previous Monte Carlo analysis of neutron wave experiments exists. The common practice was to use diffusion and transport theories to simplifying models in order to compare calculations with experiments. Section 3 describes the extreme detailed models available with the Monte Carlo method so to put this work in perspective we summarize here the indicated simplifications of previous models.

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2.1. Diffusion theory description of neutron wave experiments

In principle diffusion theory allows the calculation of neutron wave experiments with a lot of details in the description of the source, the media and the detector system. But this was not the method of practice in the literature, instead authors preferred to use an eigenvalue approach summarized below.

Assuming a homogeneous (or homogenizable) and uniform system which is bare in the directions (x, y) perpendicular to the direction of propagation z, we look for solutions to the neutron flux in energy group i, ϕ_i, which can be written as a neutron wave:

$$\phi_i = \Gamma_i \Phi_i(x,y) e^{-\rho z} e^{j\omega t} \quad (1)$$

where the complex wave vector is $\rho(\omega) = \alpha(\omega) + i\xi(\omega)$. Using the standard technique to solve the equations of conservation we first introduce Eq. (1) in the source free media in order to find the eigenvalues ρ if we assume that despite heterogeneities the transverse flux satisfies approximately $\nabla^2\Phi_i = -B_i^2\Phi_i$, the balance equations for each energy group are:

$$\left[(1-\beta)\chi_i(\nu\Sigma_f)_i + D_i\rho^2 - \Sigma_{ri} - D_i B_i^2 - \frac{j\omega}{v}\right]\Gamma_i + \sum_{l\neq i}[\Sigma_{l\to i} + (1-\beta)\chi_i(\nu\Sigma_f)_l]\Gamma_l = 0 \quad (2)$$

with the customary meaning for the symbols. Equating to zero the determinant of the system of Eq. (2) allow the calculation of the eigenvalues ρ. For each value of the transverse buckling for group i B_i^2 we have N values of ρ where N is the number of energy groups. Additionally, higher modes buckling exist because the source has a non-fundamental mode distribution in the (x, y) directions, these higher modes produce an additional multiplicity of ρ's. The flux is then a sum of exponential terms whose amplitudes satisfy Eq. (2) and the distribution in (x, y) of the source neutrons. Positioning the source to generate few higher (x, y) modes, measuring far from it and putting the detector at the zero of the first harmonics the experimenter tries then to measure a pure exponential decay produced by the smallest α. Therefore the diffusion theory was used in this context as a one-dimensional eigenvalue problem to find the smallest decay constant; experimenters tried then to have conditions under which this approximation is reasonable. To show experimentally that the measured amplitude of the neutron wave, for example, is a pure exponential is, in general, difficult. One common method is to use thermalization equipment between the high energy neutron source and the media under study and to make a Cd filter difference.

The experiment analyzed in this work failed to be descript with this diffusion model (Difilippo, 1975). As we are going to see this was because (1) the transverse leakage is very high compared to the removal cross section of high energy neutrons so departures of the transverse flux from simple cosine shape are very important and (2) the efficiency of the thermalization system was not good enough for the length of the fuel elements so energy transients were unavoidable.

2.2. Transport theory description of neutron wave experiments

Similar to the previous one-dimensional eigenvalue approach multigroup transport calculations (Travelli, 1967) were developed to analyze neutron wave phenomena. Therefore the same limitations appear: (1) acceptable approximations for the transverse leakage and (2) a good thermalized source in order to secure a pure exponential behavior for the amplitude of the wave.

An improvement in the analysis has been the transport code TASK (Dodds, Robinson and Buhl, 1972) which computes the amplitude and phase of the reaction rate of the material of a detector in the system driven by a sinusoidal neutron source. In this way details of the system as, for example, boundary effects in the direction of the propagation and not ideal thermal source (a limitation for the eigenvalue approach), can be included to compare direct observables: amplitudes and phases as function of detector position and frequency.

TASK is a one-dimensional code so one of the two indicated limitations still remains: the need of acceptable approximations for the transverse leakage.

The experiment analyzed in this work was calculated with TASK (Difilippo, 1977) with improved results with respect to the diffusion eigenvalue calculations. There were anyway discrepancies that can be attributed to the high transverse leakage included approximately as an extra removal cross section within the limitations of the code.

3. Description and modeling of the measured system

Here we give an overview of the measured system and its modeling in order to describe completely the proposed benchmark. Because there is an almost one to one correspondence between the real system (heterogeneities are included explicitly in the model) and the MCNP model we show pictures produced by the geometry module of the MCNP system to show the details of the experiment setup.

The subcritical assembly was built with sixteen boxes of MTR fuel in a four by four array. The dimensions of the fuel box are 7.62 cm (flat to flat boundaries distance along axis x), 8.04 cm (curved to curved boundaries distance along axis y) and 75.5 cm (axis z) and it has nineteen fuel plates curved in the (x, y) plane (radius of curvature 14 cm). Each fuel plate contains 8.53 g of U 90% enriched mixed with 51.53 g of Al; the U-Al mixture has a thickness of 0.52 mm and its extension in the z coordinates is 61.5 cm, the projection of the curved U-Al mix in the x direction is 6 cm. The U-Al mixture is sandwiched between two Al plates of 0.4 mm thickness and a z extension of 65.5 cm, the resulting fuel plate is then 1.32 mm thick and 65.5 cm length (the upper and lower 2 cm without fuel, plates 1 and 19 are 75.5 cm in length), the separation between plates is 2.89 mm which contains the water coolant–moderator, the unit cell of this system is therefore 4.21 mm in the y direction. With a perfect ensemble of boxes the distance between the two contiguous plates of different boxes would be 3.30 mm instead of 2.89 mm The 19 plates of the box are held in place by their insertion in two lateral Al walls 75.5 cm in length. The (x, y) cross section of the walls is a parallelograms with a base of 80.4 mm (axis y), a thickness of 5.2 mm (axis x) and an angle of 14° with axis x which follows the curvature of the plates. The sixteen boxes are contained in a 3 mm wall Al tank with Cd wrapped in the inner side, the distance between the black boundary conditions are a = (30.97 + –0.20)cm (x direction) and b = (32.71 + –0.20) cm (y direction). One special plate was used to allow the positioning of a very thin Al tube for the ^{3}He detector, the detector was located at the 0 of the first harmonics in x and y.

The (x, y) cross section of the model of the fuel box is shown in Fig. 1, the rectangle that contains the box is 76.2 mm wide with a height of 81.02 mm which goes from the horizontal tangent to the top plate to the line connecting the insertion points of the lowest plate, water gaps beside each side are included to model the non-ideal mechanical ensemble of the 16 boxes. The model is quite realistic except that the separation between the two contiguous plates of different boxes is 3.92 mm instead of 3.30 mm, approximation compatible with some undefinitions due to the non-ideal ensemble.

Views of the system are shown in Figs. 2 and 3. Fig. 2 corresponds to an (x, y) cross section of the assembled 16 boxes of Fig. 1, the water thickness between boxes is 1 mm in the x direction and 0.6 mm in the y direction, any small interstices between