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Simulation of Delayed Neutrons Using MCNP

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Abstract-Accurate modeling of the delayed neutron response in a fission process has been a desired capability for MCNP™[1]. After a year of data library and code development, a delayed neutron feature has now been incorporated into the latest version of MCNP, 4C. In this work, a validation of the integrated delayed neutron model is performed by comparisons to an analytic solution and experimental results.

Introduction

A new feature has been added to MCNP version 4C that more accurately represents the secondary production of neutrons from the fission process. Previous versions of MCNP used total nubar (ν_{total}), where available, to determine the number of secondary neutrons produced from fission. However, all of these neutrons are created as prompt neutrons in time and energy. The information with regards to the delayed neutrons, such as precursor group, energy spectra, and time of emission are not available in current data libraries, and, up until now, not handled in the MCNP code itself. Therefore, to produce a more accurate simulation of the secondary production of neutrons from fission, a data library containing the relevant delayed neutron information was constructed[2] and a delayed neutron sampling treatment was added to the MCNP code.[3,4] In this document the delayed neutron treatment in MCNP4C is tested against an analytical solution, as well as compared to experimental results for the purpose of demonstrating the validity of this new delayed neutron feature.

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Analytic Solution

The impulse response (or Green's function), which may be used for representing the response to an arbitrary time-dependent source,[5] is shown in equation 1. This function is used to test the delayed neutron treatment integrated into MCNP4C by modeling a sphere of ^{235}U having a radius of 8.5407 cm. For this calculation a neutron source with a fission-emission spectrum was placed at the center of the sphere. Fifty million source neutrons were generated with all source neutrons born at time = 0 seconds (i.e. a pulse neutron experiment). The neutron flux was tallied over the sphere as a function of time, the results of which, along with the analytic solution, are shown in Figure 1.

$$i(t) = l\sum_{j=1}^{m+1} \frac{e^{\omega_j t}}{l + \sum_i \frac{\beta_i \lambda_i}{(\omega_j + \lambda_i)^2}} \quad , \qquad (1)$$

where; $i(t)$ = relative power as a function of time,
t = time,
l = neutron generation time(1/sec),
ω_j = roots of inhour equation (7 for 6-group model) (1/sec),
β_i = delayed neutron fraction for group i, and
λ_i = decay constant for group i (1/sec).

The characteristics of the ^{235}U system used to solve the inhour equation are as follows. A neutron generation time (l) of 4.8434e-9 seconds determined from the slope of the prompt drop shown in Figure 1, a k_{eff} of 0.99873 $\pm$ 0.00003 determined from the KCODE routine in MCNP, a β_{eff} of 6.4182e-3 determined from MCNP using the algorithm described in Reference 6, and a reactivity ($\rho=(1-1/k_{eff})/\beta_{eff}$) of -0.1981.

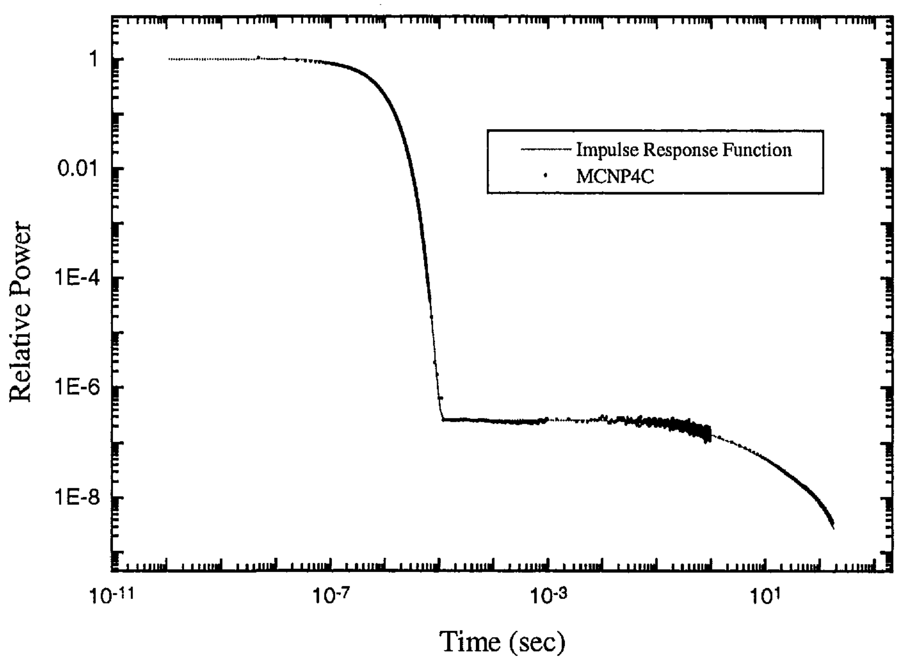

Figure 1 – The impulse response function and the MCNP simulation vs. time.

From Figure 1 it can be seen that the impulse response function and the MCNP4C calculation are in very good agreement.

Experimental Solution

Another verification of the delayed neutron treatment for MCNP4C was performed in which a source jerk experiment has been modeled. The Winco Slab Tank source jerk experiment[7] consists of two cylindrical disks of uranyl-nitrate that can be separated at various distances, see Figure 2. This experiment included a Californium source placed between the two cylinders, at the center of the apparatus, marked as '+' in Figure 2. The neutron source remained at the center of the apparatus until source equilibrium was achieved. At this point, the source was removed and the neutron intensity was measured as a function of time. The MCNP input model of this experiment used a Californium neutron source located at the center of the apparatus turned off after the flux, tallied over the entire volume of the two cylinders, had reached equilibrium.

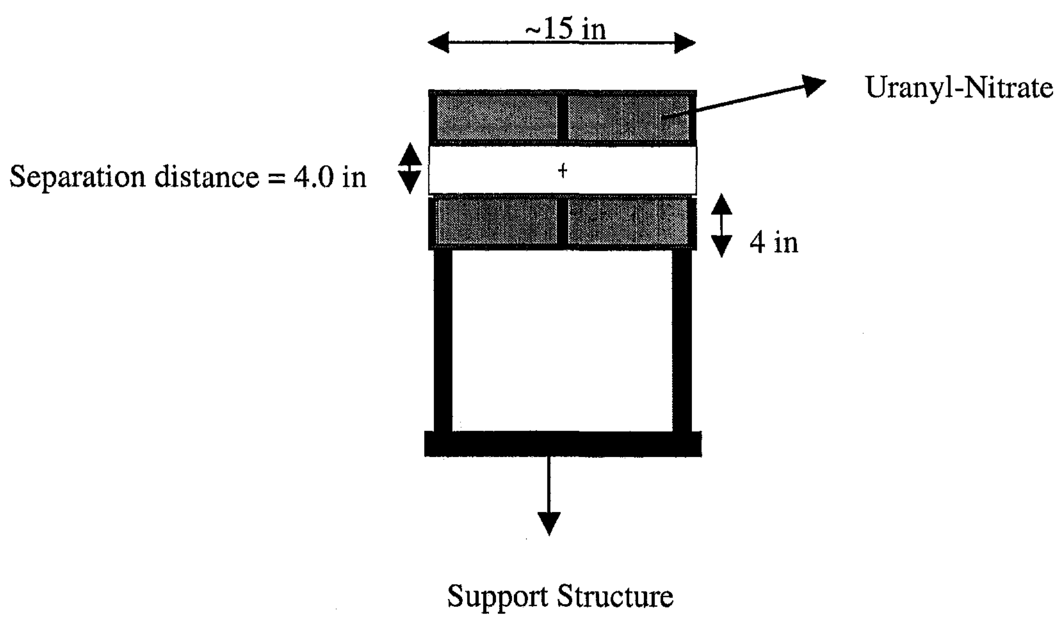

Figure 2 – The Winco Slab Tank experimental geometry modeled in MCNP

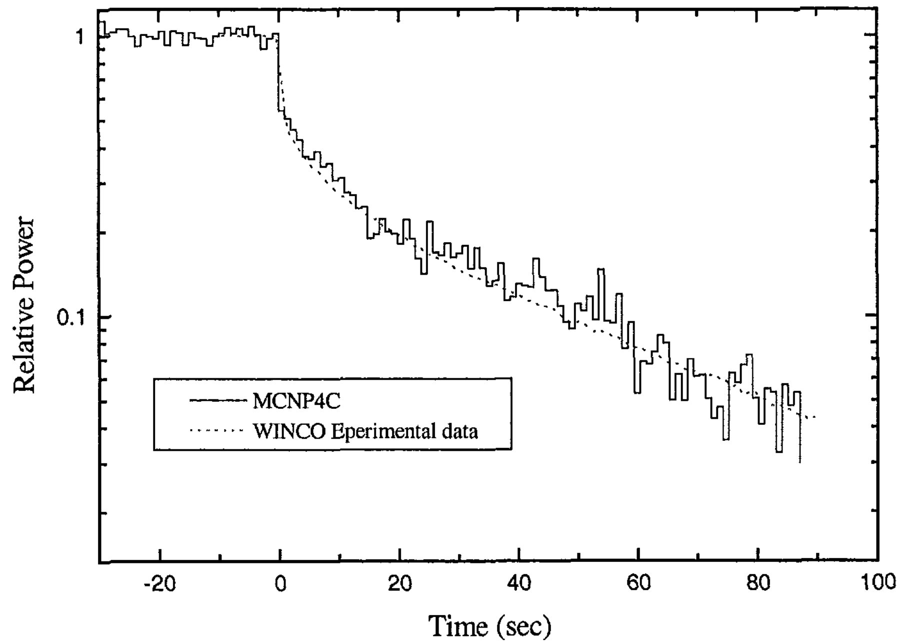

Figure 3 – Experimental and calculated delayed neutron response for the WINCO slab tank.

The results of the MCNP calculation using the delayed neutron treatment were averaged over the last 10 seconds prior to removal of the source, where equilibrium was achieved, and normalized to 1.0. This data is shown with the experimental data in Figure 3. The agreements between the experimental and calculated results are in excellent agreement.

Summary

In previously released versions of MCNP, up to and including MCNP4B, the secondary production of neutrons from the fission process has not been appropriately represented due to the code treating all neutrons created in the fission process as prompt neutrons in time and energy. This deficiency has been corrected by adding delayed data to the MCNP data libraries and modifying the MCNP code to sample delayed neutron time of emission and energy.

For the purpose of demonstrating the accuracy of this delayed neutron implementation within MCNP4C, an analytic solution that includes delayed neutrons has been compared to an MCNP calculated result. The agreement of the result to the analytic solution is very good. In addition, a delayed neutron source jerk experiment, the WINCO Slab Tank experiment, has been modeled in MCNP, the results of which are also in good agreement with those measured.

References

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