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Delayed Neutron Fraction Using MCNP4B

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Determination of the Effective Delayed Neutron Fraction Using MCNP4B

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Introduction

The effective delayed neutron fraction is defined as

$$\beta_{\text{eff}} = \tilde{\gamma} \beta_o,$$

(1)

where $\tilde{\gamma}$ is the average effectiveness factor for delayed neutrons and $\beta_o$ is the fundamental delayed neutron fraction. The average effectiveness factor, $\tilde{\gamma}$, has been traditionally calculated using the following equation derived from perturbation theory.\(^1\)

$$\tilde{\gamma} = \frac{\int \psi \chi_d \tilde{\nu}_f \Sigma_f \Phi dE d\Omega dr}{(1 - \beta_o) \int \psi \chi_p \tilde{\nu}_f \Sigma_f \Phi dE d\Omega dr + \int \psi \chi_d \tilde{\nu}_f \Sigma_f \Phi dE d\Omega dr},$$

(2)

where $\Phi$ is the angular flux, $\psi$ is the adjoint flux, $\Sigma_f$ is the macroscopic fission cross section, $\tilde{\nu}_f$ is the average prompt neutron spectrum, $\chi_d$ is the normalized average delayed neutron spectrum, $\chi_p$ is the normalized prompt neutron spectrum, and $\Omega$, $E$, and $r$ represent angle, energy, and position, respectively. The numerator in Eq. (2) is the total neutron production rate evaluated using the average delayed neutron spectrum, whereas, the denominator is the total neutron production rate evaluated using the total neutron spectrum. To evaluate this equation, one must be able to calculate the adjoint flux. In most Monte Carlo codes, this option is usually not available. Hence, Monte Carlo codes have not been routinely used in the past to perform effective delayed neutron calculations. However, current developments have shown that it is possible to calculate the effective delayed neutron fraction without any knowledge of the adjoint flux.\(^2\) Using this new theory, we have now implemented a new capability into MCNP4B which allows for the estimation of $\beta_{\text{eff}}$ in multiplying systems.

Beta-Effective Calculation 1 February 2, 1999
Theory

As shown by Spriggs et al., \( \bar{\gamma} \) can also be calculated from

\[
\beta_{\text{eff}} = \bar{\gamma} \beta_o = \frac{k_d}{k_t},
\]

where \( k_d \) is the \( k \)-eigenvalue obtained using only the average delayed neutron spectrum and \( k_t \) is the \( k \)-eigenvalue obtained using the total (i.e., prompt plus delayed neutron) spectrum. Using this equation, MCNP4B\(^3\) has been modified to perform these two \( k \)-eigenvalue solutions during a single run.

Previous versions of MCNP model the secondary neutrons from fission using only prompt neutrons, both in time and energy. In modifying MCNP4B to allow for more accurate secondary neutron production, delayed neutron data contained in ENDF/B-VI,\(^4\) such as delayed neutron yield, \( \nu_d \), delayed neutron decay constants, \( \lambda_i \), delayed neutron abundances, \( a_i \), and delayed neutron group spectra, \( \chi_i \), has been incorporated into a \( k \)-eigenvalue solution. During this \( k \)-eigenvalue solution, the active cycles are split into sub-runs in which one \( k \)-eigenvalue solution is obtained using only the six-group delayed-neutron spectra, and a second \( k \)-eigenvalue solution is obtained using the total neutron spectrum. In accordance to Eq. (3), the ratio of these two eigenvalues yields the average effectiveness factor, \( \bar{\gamma} \). To complete the calculation, the fundamental delayed neutron fraction, \( \beta_o \), is obtained by maintaining a tally of both prompt and delayed neutrons born from all fissioning isotopes in the system.

Numerical Example

The \( \beta_{\text{eff}} \) capability in MCNP4B is now in the testing phase. As an initial test, \( \beta_{\text{eff}} \) has been calculated for the WINCO Slab Tank Experiment, which was performed at the Los Alamos Critical Assemblies Facility. The slab tank assembly consisted of two cylindrical tanks 39.37 cm in diameter and approximately 15 cm in height. Each tank was filled with uranyl-nitrate solution containing 405 gm/cc of highly enriched uranium. At delayed critical, the tanks were separated by an air gap approximately 9.9 cm in height. Using continuous cross sections in the MCNP4B solution, the average effectiveness factor for this configuration was estimated to be 1.25.

This result was compared to a deterministic solution using DANTSYS,\(^5\) the 16-group Hansen-Roach\(^6\) cross sections, and a recent calculation of delayed neutron spectra for thermal fission of \(^{235}\text{U}\).\(^7\) This resulted in an effectiveness factor of 1.26. This slight difference is believed to be the result of multi-group cross sections used in the DANT solution versus continuous cross sections used in the MCNP4B solution.

Conclusions

The capability to calculate effective delayed neutron fractions has now been implemented
into MCNP4B and is in the testing phase. This option should prove to be most useful for multiply
pering systems which are not easily modeled using deterministic codes.
References


