Letters to the Editor

Comment on "Reduction of 'Calculational' Uncertainties Due to Approximate Fission-Source Matrices"

In a recent Note, Salmi et al.[1] reported on the effect of a single fission-neutron-independent spectrum versus the exact full fission matrix. Unless the authors modified the ANISN code[2] for their calculations, it would appear that their results are not quite exact.

Most S_n transport codes[2-5] use an approximate total-source matrix $\widetilde{\mathbf{TS}}$ ($\tau_{gg'}$ of Ref. 1) of the form

$$\widetilde{\mathbf{TS}} = S + \frac{1}{\tilde{k}} \tilde{\chi} \widetilde{\nu\Sigma}_f \ , \tag{1}$$

where

$\mathbf{S}$ = full scatter matrix ($\tau^s_{gg'}$ of Ref. 1)

$\tilde{\boldsymbol{\chi}}$ = diagonal matrix of a single fission spectrum ($\bar{f}_g$ of Ref. 1)

$\widetilde{\boldsymbol{\nu\Sigma}}_f$ = full matrix of fission production cross sections, all rows identical ($\nu\sigma^f_g$ of Ref. 1)

$\tilde{k}$ = k_{eff} eigenvalue for this approximate fission matrix.

The rigorous total-source matrix $\mathbf{TS}$ should be

$$\mathbf{TS} = \mathbf{S} + \frac{1}{k} \boldsymbol{\chi\nu\Sigma}_f \ , \tag{2}$$

where

$\boldsymbol{\chi\nu\Sigma}_f$ = full fission matrix ($\tau^f_{gg'}$ of Ref. 1)

k = exact k_{eff} eigenvalue.

The authors describe the use of a modified scatter matrix $\mathbf{S}^*$ to generate a total source matrix $\mathbf{TS}^*$ of the form

$$\mathbf{TS}^* = \mathbf{S}^* + \frac{1}{k^*} \tilde{\boldsymbol{\chi}} \widetilde{\boldsymbol{\nu\Sigma}}_f \ , \tag{3a}$$

where

$$\mathbf{S}^* = \mathbf{TS} - \tilde{\boldsymbol{\chi}} \widetilde{\boldsymbol{\nu\Sigma}}_f \ (\tilde{\tau}^s_{gg'} \text{ of Ref. 1}) \tag{3b}$$

and k^* is their k_{eff} eigenvalue.

Insertion of Eq. (2) into Eq. (3b) shows that their $\mathbf{TS}^* = \mathbf{TS}$ only if $k^* = 1$. To be rigorously correct, their scatter matrix *should* be

$$\mathbf{S}^* = \mathbf{TS} - \frac{1}{k} \tilde{\boldsymbol{\chi}} \widetilde{\boldsymbol{\nu\Sigma}}_f \ , \tag{3c}$$

which means their modified scatter matrix cannot be precomputed, and the ANISN code cannot be used without modification. Clearly, the differences between Eqs. (3b) and (3c) are small for systems near critical.

The authors also state, in effect, that $\widetilde{\mathbf{TS}} = \mathbf{TS}$ for eigenvalue problems other than k_{eff} problems. Comparing Eqs. (1) and (2), the full matrix $\boldsymbol{\chi\nu\Sigma}_f$ *cannot always*, in general, be factored into the form $\tilde{\boldsymbol{\chi}} \widetilde{\boldsymbol{\nu\Sigma}}_f$, a diagonal matrix times a full matrix with identical rows.

Furthermore, no general S_n transport code, to my knowledge, can use only the total scatter matrix $\mathbf{TS}$. These codes usually compute k as an intermediate eigenvalue with which to search for the primary eigenvalue. Thus, the fact that $\widetilde{\mathbf{TS}} \neq \mathbf{TS}$ *is* important for other types of eigenvalue problems as well.

Finally, the effect described by the authors was reported by Hill et al.[6] in 1973. The Los Alamos National Laboratory ONETRAN/TIMEX codes have, in fact, an option to correctly treat the fission matrix without the $\tilde{\boldsymbol{\chi}} \widetilde{\boldsymbol{\nu\Sigma}}_f$ approximation or resort to a modified scatter matrix.

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[1] U. SALMI, J. J. WAGSCHAL, A. YAARI, and Y. YEIVIN, *Nucl. Sci. Eng.*, **84**, 298 (1983).

[2] W. W. ENGLE, Jr. "ANISN, A One-Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering," K-1693, Oak Ridge Gaseous Diffusion Plant (1967).

[3] T. R. HILL, "ONETRAN: A Discrete Ordinates Finite Element Code for the Solution of the One-Dimensional Multigroup Transport Equation," LA-5990-MS, Los Alamos National Laboratory (1975).

[4] T. R. HILL and W. H. REED, "TIMEX: A Time-Dependent Explicit Discrete Ordinates Program for the Solution of the Multigroup Transport Equations with Delayed Neutrons," LA-6201-MS, Los Alamos National Laboratory (1976).

[5] R. D. O'DELL, F. W. BRINKLEY, Jr., and D. R. MARR, "User's Manual for ONEDANT: A Code Package for One-Dimensional, Diffusion-Accelerated, Neutral-Particle Transport," LA-9184-M, Los Alamos National Laboratory (1982).

[6] T. R. HILL, K. D. LATHROP, and G. HANSEN, *Trans. Am. Nucl. Soc.*, **16**, 329 (1973).