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On Exact Solutions of the Neutron Slowing Down Equation in Resonant Absorbing Media: L. Finkelstein and M. Shatz

A broad class of resonant absorbing cross section functions was found for which there are exact and analytically simple solutions of the neutron moderation equation. Some of these solutions were compared with the corresponding numerical solutions, obtained by the methods of Rowlands[1], Nordheim[2] and Finkelstein[3]. While the results of the first two methods diverged progressively from the analytical solution after passing the resonance peak (even for extremely small steps, $c/\Delta \sim 13000$, the results obtained by the last method were close to the analytical curve over the entire range of calculation, especially for heavy nuclei. In this case the steps of the numerical integration could be made large ($c/\Delta = 16$) almost without influencing the accuracy.

References:

1. ROWLANDS, G., J. Nucl. Eng., 13, 14 (1960)
2. NORDHEIM, L.W., in Nuclear Reactor Theory, Proc. Symp. App. Math., Rhode Is, Vol.2, 1961, p.65
3. FINKELSTEIN, L., Nucl. Sci. Eng., 32, 241 (1968)

Theory of a One-Dimensional Bare Reactor Treated by the Lie Series Method(1): H. Lemanska, Y. Ilamed and S. Yiftah

The one-dimensional G-group diffusion equation for the bare reactor was solved using the Lie series technique. The G-vector flux at any point of the reactor is given by a G x G transfer matrix multiplied by the flux at the origin, and a generalized concept of buckling is introduced. The physical meaning of the largest eigenvalue of this transfer matrix was considered.

A method was worked out for calculating the critical size of a bare reactor. The calculated critical radii for several assemblies having spherical geometry are given in Table I, where the results obtained previously by the diffusion code are also given for comparison. The composition of the ZPR-III assemblies was taken from Ref. 2 and the cross sections were taken from the YOM tables[3]. The results

obtained by the two methods are practically the same. The Lie method is simpler in calculation because the matrix whose eigenvalue has to be found is only of order G, while in the diffusion it is of order G x I, where I is the number of division points of the radius.

TABLE I

Critical Radii obtained by the Lie method and the classical difference equation method

Assembly (core only)	Critical radius (cm)	
	Lie transfer matrix	Diffusion code
ZPR-III/10	35.42	35.43
ZPR-III/11	40.43	40.8
ZPR-III/2A	40.17	40.30
ZPR-III/6F	38.13	38.12
ZPR-III/9A	37.12	37.15
ZPR-III/16	40.64	40.62
ZPR-III/12	39.87	39.89
ZPR-III/17	39.72	39.75
ZPR-III/14	40.8	40.9

References:

1. LEMANSKA, M., ILAMED, Y. and YIFTAH, S., IA-1187 (1967)
2. LONG, J.K., LOEWENSTEIN, W.B., BRENGAN, C.E., BORENSON, G.S., KIRU, F.S., OKRENT, D., RICE, R.E. and THALGOTT, F.S., Proc. of the 2nd Int. Conf. on the Peaceful Uses of Atomic Energy, Vol. 12, Geneva, 1958, p.119
3. YIFTAH, S., OKRENT, D. and MOLDAUER, P.A., Fast Reactor Cross Sections. London, Pergamon Press, 1960

Theory of a One-Dimensional Reflected Reactor Treated by the Lie Series Method: M. Lemanska, Y. Ilamed and S. Yiftah

The one-dimensional G-group diffusion equation for the reflected reactor was solved using the Lie series technique. As in the case of the bare reactor[1] (see previous article) the G-vector flux at any point of the reactor is given by a G x G transfer matrix multiplied by the flux at the origin. When the point of interest belongs to the

reflector zone this matrix is the product of the transfer matrices for the core zone and for the reflector zone.

The critical equations for the reflected slab and spherical reactor were obtained according to the multigroup method. The analogy between these equations and the ones obtained by the monoenergetic theory have been considered.

Reference:

1. LEHANSKA, M., ILAMED, Y. and YIFTAH, S., IA-1187 (1958)

HYBSHOV - A Program for Solving the Few-Group Space-Time Dependent Diffusion Equation with a Hybrid Computer(1): D. Saphier* and S. Yiftah

A program for solving the space-time dependent diffusion equation with a hybrid computer has been set up for the E.A.I. - 8900 hybrid system. The digital part of the program is written in FORTRAN IV, and the control and data transfer subroutines are in the machine language of the E.A.I. - 8400 digital computer.

The diffusion equation is solved for a slab reactor with 44 mesh-points, 4 energy groups and 4 groups of delayed neutrons. The number of regions is 7. In each region different isotopic composition is assumed. The code provides for a maximum number of 10 isotopes in a mixture.

In the program the energy and the space variable are discretized while the solution with respect to time is continuous. After the definition of the critical system and the initial conditions, the equations for each region are integrated separately by the analog part. In each region there are 6 meshpoints. At any meshpoint 8 equations are integrated - 4 for prompt neutrons and 4 for delayed neutrons - hence a total of 48 equations are integrated at the same time by the

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analog part. Calculations in one region determine the left boundary conditions for the next region and the right boundary condition for the next iteration of the previous region.

The assumptions made in developing the equations to be solved are described in Ref. 1, as well as the limitations and range of applicability of the program. A detailed description of the code as well as the various options are presented. The problems (hardware and software) arising in hybrid computation are discussed. It is shown that the price per solved problem with the hybrid computer may be reduced by a factor of 3 to 10 as compared with a similar solution with a digital computer.

Reference:

1. SAPHIER, D. and YIFTAH, S., IA-1191 (in press)

SHOVAV - A Program for Solving the Few-Group Space-Time Dependent Diffusion Equation with Temperature Feedback(1): D. Saphier and S. Yiftah

A digital computer program for solving the space-time dependent diffusion equation with temperature feedback has been set up for the Philco TRANSAC S-2000 computer. The program is written in FORTRAN II and may benrun on other computers after minor changes are introduced.

The diffusion equation is solved for a slab reactor with a maximum of 11 regions and 100 meshpoints. There are 4 energy groups and up to 6 delayed neutron groups. The code provides for a maximum number of 10 isotopes in a mixture.

The mean values of fuel and coolant temperature at each meshpoint are calculated from the power production distribution in the reactor. The influence of temperature and density changes on the reactor dynamics is taken into account via the mixing and shielding factors, which cause t the cross sections to change with temperature and density. Thus the Doppler and sodium coefficients are fully included in the calculations.

The assumptions made in developing the equations to be solved, as well as the limitations and range of applicability of the program are discussed in Ref. 1. The options available to the user and a full description of the code and its structure are given. A sample problem showing the input-output facilities is presented.

Reference:

1. SAPHIER, D. and YIFTAH, S.

Periodic Solutions of the Transport Equation in a Reactor Lattice Cell with a Fuel Element of Finite Radius: I. Aviram

The research on this topic reported previously[1] has been continued.

The matrix of the system of equations for the determination of the expansion coefficients $C(k,\ell,m)$ can be arranged according to the indices ℓ,m,k (in this order) so that a block tri-diagonal partitioning is obtained. The proposed Point-Successive-Over-Relaxation iterative method for solving the system failed for the following reasons. The theory of Ostrovsky and Young defines the necessary conditions for the convergence of the SOR method for a symmetric, positive definite matrix. The theory also permits the choice of the optimum relaxation factor when the matrix fulfils certain ordering properties called "consistent ordering". In our case the matrix is neither symmetric nor positive definite. It is, however, consistently ordered when its block tri-diagonal partitioning is taken into consideration. Preliminary experiments with the Point-SOR method showed that values of the relaxation factor exist for which the method converges. It is not possible, however, to determine the value of the relaxation factor for a given matrix beforehand, nor is it possible to correct it automatically during the process. The method was therefore abandoned as being of no practical value.

The Block-Successive-Over-Relaxation method (which takes advantage of the consistent ordering) also failed, because the spectral radius of the corresponding Jacobi iteration matrix turned out experimentally

to be larger than unity in all cases examined. Gaussian elimination can also not be used effectively in this case because of the size of the matrices, which introduces considerable difficulties in program organization.

The only feasible method of solution appears to be the method of Conjugate Gradients. Although quite laborious (it requires six times as many multiplications as Gaussian elimination for a full matrix), it can easily take advantage of the sparseness of the matrix. Our matrix posses a band structure i.e., non-zero elements appear only in a central diagonal band of well defined width. The band itself contains very many zeros (about 60% of the elements included), and it is this feature that gives the C.G. method a clear superiority over Gaussian elimination.

Two programs, TATRAN-6C and TATRAN-4, are being written for the soltion of the problem in the case of the hexagonal centered and square lattices respectively. The index ℓ can be as large 5 (equivalent to the P5 approximation), while the indices k_1 and k_2 (components of $\underline{k}$ in the Fourier plane) can go up to 7. Full advantage is taken of the geometric symmetry properties of cells in order to obtainaa maximum reduction of the number of equations. For example, the hexagonalncell requires the solution of a matrix of dimensions (420 x 420) with about 10% non-zero elements. The whole matrix can be stored in the fast memory of a 32K compuger. The problem is programmed for the GOLEM computer of the Weizmann Institute of Science, Rechovoth (75 bits per word, 8 μsec per multiplication). Run time for a typical problem is less than 20 min. The main features of these programs are:

a) Up to 4 annualr regions centered at the axis of the cell can be included, plus a moderator region which fills the remaining volume of the cell.

b) Refdection conditions are satisfied automatically on the actual

boundary of the cell.

c) Anisotropic scattering up to order 5 in all materials can be included with no additional computational effort.

Reference:

1. AVIRAM, I., in IA-1168 (1967) p.2

Neutron Distribution in Bulk Media: E. Greenspan

Measurements of position-dependent angular distribution of neutrons in a depleted uranium assembly revealed that the distribution in this assembly is highly anistropic[1]. Furthermore, the anistropy was found to be a strong function of the neutron energy, with non-monotonic behaviour. Similar measurements conducted on an iron assembly yielded neutron spectra whose fine structure showed a strong correlation with the energy dependence of the iron total cross section[2]. Studies are being carried out aimed at understanding the physical phenomena involved in neutron deep-penetration problems and at assessing the adequacy of computational techniques and cross section data for the prediction of the neutron distribution in experimental assemblies. Particular attention is being paid to the investigation of the anisotropy in the neutron distribution and of the way in which large interference minima in the cross sections affect the neutron spectra.

References:

1. GREENSPAN, E., MALAVIYA, B.K., KAUSHAL, H.N., GAERTTNER, E.R. and DAITCH, P.B., Proc. of the Second Conf. on Neutron Cross Sections and Technology, Washington, D.C. 1968
2. MALAVIYA, B.K., GREENSPAN, E., GAERTTNER, E.R. and MALLEN, A., Proc. of the Second Conf. on Neutron Cross Sections and Technology, Washington, D.C., March 1968

Classification of Methods for Measuring Neutron Importance Functions in Nuclear Reactors(I): E. Greenspan

The various methods for measuring the neutron importance function distribution in nuclear reactors have been formulated and classified.

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The classification is based on the theoretical interpretation of the measured distribution. The measurement techniques are divided into two categories: source methods and perturbation methods. The source methods are further subdivided into measurements in just critical reactors, comprising the Pulsed Source, the Linear Power Rise, the Extrapolation, and the Source Jerk methods, and into measurements in slightly subcritical reactors, including the Source Multiplication and the Pseudo-Reactivity methods. Experimental measurements of importance function distributions reported in the past are categorized according to the classification scheme.

The motive behind this project is not only the scientific interest in fundamental reactor theory but also the growing need for accurate knowledge of importance function distributions for the analysis and design of nuclear reactors, especially fast ones. It is hoped that this work will help in establishing standard techniques for the measurement of importance function distributions. These techniques could supplement the well established methods for the measurement of neutron fluxes used extensively for experimental verification of theoretical predictions.

Reference:

1. GREENSPAN, E. and CADY, K.B., Nucl. Sci. and Eng. (in press)

Sodium Void and Doppler Effects in Large Fast Power Reactors[1]:
Yiftah, S., Gitter, L. and Ilamed, Y.

The results obtained in previous computations for 27 plutonium assemblies were analyzed. The following aspects were considered: reactivity as a function of sodium leakage; sodium leakage as a function of volume at normal temperature; threshold and stationary volume for variable initial sodium content at normal temperature; temperature effects.

It was concluded that oxide assemblies permit the largest reactors with combined non-positive reactivity effect, and that changes in the isotopic composition of the fuel do not affect this safety factor adversely.

Reference:

1. YIFTAH, S., GITTER, L. and ILAMED, Y., IA-1177, 1968

Evaluation of Nuclear Data for the Higher Plutonium Isotopes: Caner, M. and S. Yiftah

A revision of the evaluation[1] of data for Pu-240, 241, 242 is in progress, incorporating experimental data published since the last evaluation. Particular emphasis is being put on the theoretical calculation of neutron cross sections based on the optical model of nuclear reactions and on the Hauser-Feshbach-Moldauer theory. These calculations are being done with the codes ABACUS-2[2] and NEAREX[3].

References:

1. YIFTAH, S., SCHMIDT, J.J., CANER, M. and SEGEV, M., in Fast Reactor Physics, Vol. 1. Vienna, IAEA, 1968, p.123. Also IA-1152 (1967)
2. AUERBACH, E.H., preprint BNL 6562 (revised 1964)
3. MOLDAUER, P.A., ENGELBRECHT, C.A. and DUFFY, G.J., SNL-6978 (1964)

Comparison of Parameters of Nuclear Systems as Calculated Using Various Nuclear Data Files: Y. Gur and S. Yiftah

The aim of this work is to study the sensitivity of the calculated parameters of representative nuclear systems to different sets of basic nuclear data, by recalculating each system using the same method and techniques but a different basic nuclear data file in each calculation.

Multigroup cross section sets based on the ENDF/B Nuclear Data File are now in preparation. The other files to be tested are the U.K. Nuclear Data File and the Keddak file.

A Steam Generator Model for Dual Purpose Nuclear Power Plant Simulation[1]: D. Saphier, J.R. Wolberg and S. Yiftah

A mathematical model for simulating a natural circulation steam generator has been developed. This work was performed as part of an overall analysis of the dynamic behavior of a dual purpose nuclear plant.

The model reflects the compromises and simplifying assumptions