TRANSACTIONS

OF THE
AMERICAN NUCLEAR SOCIETY
1974 ANNUAL MEETING

Bellevue Stratford Hotel — June 23-27, 1974 — Philadelphia, Pennsylvania

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TECHNICAL SESSIONS

Monday, June 24, 9:30 AM

Fast Reactor Safety: Mechanical Damage Analysis 196
*Should Nuclear Engineers Register as Professional Engineers? 29
*Reactor Operator Retraining 282
*Plant Design Standardization 237
*In-Plant Radiation Exposure Experience-I . 355
*Nuclear Power for Terrestrial and Underwater Applications and Nuclear Propulsion 1
*Computational Models in CTR Plasma Simulation 130
Cross Sections for Radiation Transport . . 362
Measurement and Calculation of Reactor Parameters 309
Analytical Applications of Isotopes and Radiation 61

Monday, June 24, 2:00 PM

Fast Reactor Safety: Local Subassembly Fault Conditions 202
Educational Programs and Resources for the Classroom and the Public 29
Sodium Technology 101
*Construction Status of Nuclear Power Plants 241
*In-Plant Radiation Exposure Experience-II . 361
*Low-Cost Nuclear Power System Alternatives for Aerospace Applications-I 6
Reactor Theory and Applications 316
*Comparison of First Year Power Reactor Performance with Prediction 286
Reactor Siting and Decommissioning 242
Radiation Sources and Detection 67

Monday, June 24, 7:30 PM

*Operational Quality Assurance 292

Tuesday, June 25, 9:00 AM

Fast Reactor Safety: Hypothetical Subassembly Accident Experiment and Analysis 209
*Power Generation System Planning and Optimization 171
*Nuclear Criticality Safety Especially for Mixed Oxides 160
*Fuel Performance Experience in Power Reactors 247
*Nuclear Decontamination and Low-Level Wastes 77
*Low-Cost Nuclear Power System Alternatives for Aerospace Applications-II . . . 13
*Track Etch Applications-I 82
Reactor Operations: Power 294
*High-Fluence Effects on Stainless Steels in Fast Reactors 106
Radiation Transport (Experimental) 369

Tuesday, June 25, 2:30 PM

*A Standard of Missile Protection 216
Fuel Cycle Analysis, Management, and Economics 177
*History of the Development of LWR Fuels and Fuel Elements 111
*Reactor Physics Informal 325
*Experience with Standards and Guides 252
Controlled Nuclear Fusion-I 18
Track Etch Applications-II 87
Environmental Releases and Consequences . 42
Methods for Solution of Engineering Problems 132
Shielding and Radiation Transport Applications 377

Tuesday, June 25, 7:30 PM

*Nuclear Energy Options for the Future 325

Wednesday, June 26, 9:00 AM

Fast Reactor Safety: Fast Reactor Thermal Interactions/Miscellaneous Safety 220
*Nuclear Accountability and Safeguards 184
*Non-Reactor Neutron Activation Analysis . . 91
*Fort St. Vrain Startup Experience 257
*Nuclear Steam Generator Materials Technology 115
*Dosimetry for Internal Emitters-I 384
Reactor Operations: Test and Research . . 301
Environmental Surveillance and Analytical Techniques-I 49
*Industrial Needs and Academic Research Fuel Cycle 35
Nuclear Data for Reactor Design 327

Wednesday, June 26, 2:00 PM

*Public Hearings and the Public Interest (Does One Serve the Other?) 227
Waste Management 190
*In-Vivo Activation Analysis 96
*Meeting "As Low As Practicable" for Water Reactor Effluents 261
*Fuel Densification 120
*Dosimetry for Internal Emitters-II 389
*Nuclear Engineering Education and Socio-Economic Problems 39
Environmental Surveillance and Analytical Techniques-II 54
Transport Theory Methods 142
*Implication for Reactor Design of Uncertainties in Data and Methods 335

Thursday, June 27, 9:00 AM

Water Reactor Safety 232
Stress and Structure Analysis 262
Controlled Nuclear Fusion-II 22
Fast Reactor Safety Engineering 267
Fuels Technology 124
Reactor Engineering 274
Radiation Transport (Analytical) 394
Methods for Solution of Physics Problems . 151
Reactivity Surveillance in FFTF 343
*Status of ENDF/B for Thermal Reactors . . 351

*Special Session

Since $\partial\theta/\partial t = K\psi$ for the adiabatic reactor, the flux can be obtained from Eq. (13) as

$$\psi^0(x, t) = \frac{p^2/K \cos\left(\frac{\pi}{2} x\right)}{\frac{8\alpha}{3\pi} - pg\left[\exp(pt) + \left(\frac{4\alpha}{3\pi}\right)^2 \frac{\exp(-pt)}{pg}\right]} \quad , \qquad (15)$$

yielding

$$\lim_{t \to \infty} \psi^0(x, t) = 0 \qquad (16)$$

regardless of the change in λ. This intrinsic shutdown mechanism of the adiabatic reactor also has been shown by Kastenberg.[6]

Much information essential for the safety analysis of fast reactors can be obtained from the solutions obtained above. In spite of the intrinsic shutdown mechanism of the adiabatic reactor, its flux reaches a maximum several decades higher than that of the Newtonian reactor before dropping off. For the Newtonian feedback, the gas-cooled reactor reaches an asymptotic flux several magnitudes below that of the sodium-cooled reactor. Other information such as the rate of flux increase and the time at which the maximum flux occurs can be readily obtained and compared for various reactors.

1. L. V. KANTOROVICH and G. P. AKILOV, *Functional Analysis in Normed Spaces*, Chap. 18, The MacMillan Co., New York (1964).
2. HERBERT B. KELLER, *Numerical Methods for Two-Point Boundary-Value Problems*, Blaisdell Publishing Co., Waltham, Mass. (1968).
3. DONG H. NGUYEN, *Nucl. Sci. Eng.*, **52**, 292 (1973).
4. W. K. ERGEN, *Nucl. Safety*, **8**, 30 (1966).
5. A. N. TIKHONOV and A. A. SAMARSKII, *Equation of Mathematical Physics*, MacMillan Company, New York (1963).
6. WILLIAM E. KASTENBERG and PAUL L. CHAMBRÉ, *Nucl. Sci. Eng.*, 31, 67 (1968).

8. A Summation-Exponent Analysis for Space-Dependent Reactor Transients, *W. J. Garland, J. Vlachopoulos, A. A. Harms (McMaster Univ)*

Hansen et al.[1,2] have recently demonstrated the effectiveness of two methods of extracting the dominant exponential time dependence in space-dependent reactor transient calculations. In their first analysis,[1] the g'th-group neutron flux in the space-time multigroup diffusion formulation is written as

$$\phi_g(\mathbf{r}, t) = \psi_g(\mathbf{r} \cdot t) \exp(\alpha_g t) \quad ; \qquad (1)$$

while in a subsequent analysis,[2] the special case of Eq. (1),

$$\phi_g(\mathbf{r}, t) = \psi_g(\mathbf{r} \cdot t) \exp(\alpha t) \quad , \qquad (2)$$

is employed.

We have studied these two approaches and find that a summation-exponent-type representation can have a significant effect in the reduction of computer calculation time and/or increasing numerical accuracy. We consider the g'th-group neutron flux in the form

$$\phi_g(\mathbf{r}, t) = \sum_{l=1}^{G} \psi_{g,l}(\mathbf{r} \cdot t) \exp(\alpha_l t) = \psi_g(\mathbf{r}, t) \sum_{l=1}^{G} f_{g,l} \exp(\alpha_l t) \quad , \qquad (3)$$

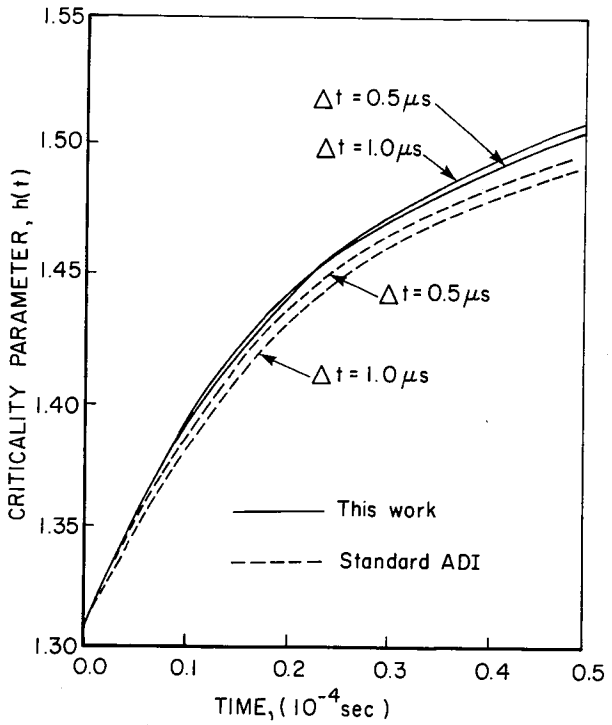

Fig. 1. Criticality parameter, h(t), as a function of time and time steps for the case of constant power productivity.

where we have specified G energy groups and, hence, G expansion coefficients and exponents. A fundamental and most significant consequence of this representation is that—depending on the relative "stiffness" of the diffusion equations—a most effective means of incorporating rapid changes in the system description is possible.

As an illustrative example, we consider a CANDU-BLW lattice cell modeled by a two-group neutron diffusion representation.[3] The initial flux distributions were assumed to be flat. The two-group diffusion equations were solved by the alternating direction implicit (ADI) method[4] using the transformation of Eq. (3) and, for comparison purposes, without the transformation. As a case of interest, we simulated the control situation in which the criticality parameter,

$$h(t) = \epsilon \eta f(t) \quad , \qquad (4)$$

was adjusted to maintain a constant power productivity, defined by

$$\int_{\text{fuel}} h(t)\, \phi_{th}(\mathbf{r}, t)\, d\mathbf{r} = \text{constant} \quad . \qquad (5)$$

The result of this analysis is shown in Fig. 1. Of interest in this particular case is the observation that the summation-exponent representation, Eq. (3), provides consistently better numerical accuracy than the standard ADI method. The effective computation time saving was on the order of a factor of 3 to 4.

In addition to several calculational tests designed to study the effectiveness of the summation-exponent solution approach, we have also investigated the property of stability and convergence; these properties are met, in general, to the same extent as those applicable to the use of Eqs. (1) and (2). Finally, we have examined some criteria that can be used to provide qualitative conclusions of the effectiveness of this solution formalism based on the neutronic properties of the system.

1. A. L. WIGHT, K. F. HANSEN, and D. R. FERGUSON, *Nucl. Sci. Eng.*, **44**, 239 (1971).

2. D. R. FERGUSON and K. F. HANSEN, *Nucl. Sci. Eng.*, **51**, 189 (1973).

3. M. SCRINIVASAN and K. J. SERDULA, AECL-3513, Chalk River Nuclear Laboratories (1969).

4. B. CARNEHAN, H. A. LUTHER, and J. O. WILKES, *Applied Numerical Methods*, Wiley and Sons, Inc., New York (1969).

9. A Higher Order Relationship Between Static Power Tilts and Eigenvalue Separation,

W. D. Beckner, R. A. Rydin (U of Va)

An important parameter in the analysis of spatial kinetics problems is the difference between the two largest eigenvalues of the static reactor balance equation.[1] A method of accurately measuring eigenvalue separation would be useful to confirm calculational models used extensively in reactor design. Wade and Rydin[2] have proposed measuring eigenvalue separation through a static flux tilt technique. This method uses the fact that, to first order, eigenvalue separation is inversely proportional to the flux tilt per unit of perturbing reactivity.

A higher order perturbation theory relationship has been developed here. Previous work has shown that the perturbed flux can be expanded as a linear combination of the lambda modes, with the expansion coefficients given by

$$A_j = \frac{\lambda_0}{\lambda_0 - \lambda_j} \frac{\langle \Psi_j^*, (\delta M - \delta L)\phi \rangle}{\left\langle \Psi_j^*, \frac{M_0}{\lambda_j} \Psi_j \right\rangle} , \quad (1)$$

where δM and δL are the perturbations introduced in the production and destruction operators. A first-order approximation to these coefficients was obtained by assuming that the perturbed flux, ϕ, is proportional to the fundamental mode, Ψ_0. This derivation also requires that the net reactivity introduced by the perturbation be zero.[2]

In an actual reactor, a tilt-inducing perturbation results in a nonzero reactivity, and a compensating change in reactor properties must accompany this perturbation to achieve criticality; e.g., a uniform change in soluble poison. The total perturbation can then be represented as the sum of the initial perturbation and the compensating perturbation. Now the coefficients become

$$A_j = \frac{\lambda_0}{\lambda_0 - \lambda_j} \frac{C_j \rho}{\left(1 + \rho \frac{\lambda_j \lambda_0}{\lambda_0 - \lambda_j}\right)} , \quad (2)$$

where

$$C_j = \frac{\rho_j}{\rho} = \frac{\langle \Psi_j^*, (\delta M - \delta L)\phi \rangle}{\langle \Psi_0^*, (\delta M - \delta L)\phi \rangle} , \quad (3)$$

and where ρ is the change in k_{eff} induced by the initial perturbation. For small reactivities compared to the eigenvalue separation this form reduces to the original representation.

The previous work also assumed that the perturbed flux was similar to the original flux when calculating the correction factor, C_j. For small perturbations, C_j is approximately equal to C_j^0, where

$$C_j^K = \frac{1}{\lambda_K} \frac{\langle \Psi_j^*, (\delta M - \delta L)\Psi_K \rangle}{\langle \Psi_0^*, (\delta M - \delta L)\Psi_0 \rangle} . \quad (4)$$

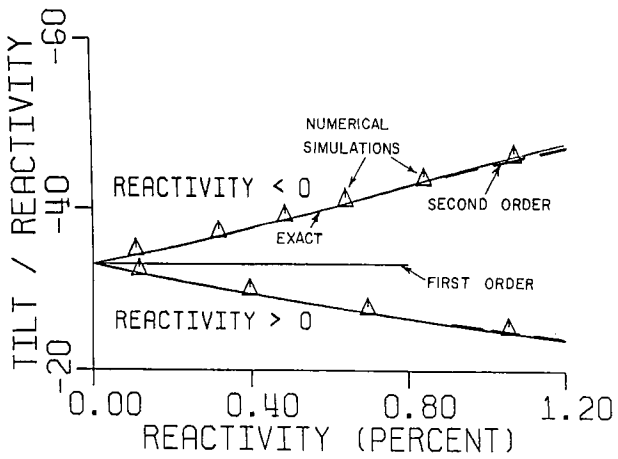

Fig. 1.

For perturbations that significantly affect the flux shape, this approximation is not valid. By expanding the perturbed flux in Eq. (3) using the expansion coefficients given in Eq. (2), the following representation for the correction factors is found:

$$C_j = \frac{C_j^0 + \rho \sum_{k=1} \frac{\lambda_0 \lambda_k}{\lambda_0 - \lambda_k} \frac{C_k C_j^k}{\left(1 + \rho \frac{\lambda_0 \lambda_k}{\lambda_0 - \lambda_k}\right)}}{1 + \rho \sum_{k=1} \frac{\lambda_0 \lambda_k}{\lambda_0 - \lambda_k} \frac{C_k C_0^k}{\left(1 + \rho \frac{\lambda_0 \lambda_k}{\lambda_0 - \lambda_k}\right)}} . \quad (5)$$

A second-order approximation is obtained by substituting the first-order approximation that $C_j = C_j^0$ into Eq. (5). Higher order approximations are found by continuing this process. It has been found that this iteration converges very rapidly for realistic perturbations.

By substituting these two changes into the original theory, the tilt induced by a perturbation for a given eigenvalue separation can be calculated. The figure shows the predicted power tilt calculated by the original first-order theory compared to second-order theory and an "exact" prediction found by taking the iteration on the C_j's to as many terms as necessary to get the desired accuracy. Data from computer simulations of power tilts on one-dimensional models are included in the figure. Agreement with the higher order theory is within 2% for several types of perturbations over a wide range of perturbation magnitudes.

In a practical situation, the seemingly complicated theory presented here simplifies greatly. Since the correction of the C_j terms is small, Eq. (5) can be truncated with little loss of accuracy. This new theory also suggests practical experimental procedures that would provide for more accurate interpretation of data by making the effects of higher harmonic modes less important than in the first-order theory; this analysis has been successfully applied to the SHA data.[3]

1. W. M. STACEY, Jr., *Space-Time Nuclear Reactor Kinetics*, Chap. 5, Academic Press, New York (1969).

2. D. C. WADE and R. A. RYDIN, "An Experimentally Measurable Relationship Between Asymptotic Flux Tilts and Eigenvalue Separation," in *Dynamics of Nuclear Systems*, p. 335, D. L. HETRICK, Ed., The University of Arizona Press, Tucson, Arizona (1972).

3. R. A. RYDIN, J. A. BURKE, W. E. MOORE, and K. W. SEEMANN, *Nucl. Sci. Eng.*, **46**, 179 (1971).