Advances in Nuclear Science and Technology

VOLUME 11

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Series Editors

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Preface

The present volume in our annual review series reviews a wide range of developments, giving a broad interpretation to the "technology" of our title. Starting at the beginning, Science, we have the review of basic nuclear physics data of Walker and Weaver for reactor kinetics, particularly, therefore, delayed neutron data. In the search for better and better accuracy, it is being realized that this involves the closest scrutiny of fundamental data, given to us here from the Birmingham school. Associated with this review of data is the review from Italy by Professor Pacilio and his co-workers of the theory of reactor kinetics in the stochastic form, and a valuable compilation of the theory underlying a wide range of practical techniques.

Tending more to technology come the papers by Jervis, reviewing the application of digital computers to the control of large nuclear power stations as developed in both the United Kingdom and Canada, Pickman's review of the design of fuels for heavy water reactors, and the account by Ishikawa and Inabe of the new Japanese Research Reactor Program, itself initially directed largely to fuel element studies.

The balance of the volume is made up of more philosophical contributions to the practicalities of nuclear power. Much has been said about plutonium (indeed, much that did not deserve to be said) in the last two years; Puechl's review of its role in relation to breeding and recycling is a welcome, sane, summary of the problems and advantages. Similarly, the paper by McNelly on the usefulness of Nuclear Power Parks comes at an opportune time in the development of national programs that seek to retain and exploit the advantages of nuclear power in acceptable ways. All those who are engaged in such applications have a responsibility to see that the education and training of professional scientists and engineers has the highest technical and ethical content, so it is not inappropriate to see the review of computer-assisted learning for nuclear engineers by Smith included in this volume as well.

The year of the Carter initiative has passed, and it is not for us to give premature judgment. Clearly, much that has been aired needed saying, if only to expose it to critical examination. Not all the initiatives that came from the U.S.A. appear permanent, either within the U.S.A. itself, or by adoption elsewhere in the world. Clearly, the major European countries led by France and Germany see the valuable benefits that nuclear programs can bring to countries with little indigenous coal left and no indigenous oil; the United Kingdom has no pressing reason for an early decision (for which she is no doubt thankful) but the present review of the Japanese research reactor program given here must point up the difficult economic circumstances of that country and the strong pressure there will be to seek the benefits of nuclear power.

Some of the issues concerning fast breeders and the recycling of plutonium have been dealt with here within this and previous volumes of the series; other issues raise broader questions of public policy than can be analyzed here. Yet it is important to remember that sound analysis, judgment, and selection of a course of action in any nation requires acknowledgment of physical and technical facts, rather than prejudices and speculation. We may hope that in this volume, therefore, we and our authors contribute to such a balance.

E. J. Henley

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Contents

Nuclear Physics Data for Reactor Kinetics
J. Walker and D. R. Weaver

I. Principles of Reactor Kinetics and Data Requirements 1
II. Evaluated Nuclear Data Libraries 4
III. Fission-Product Nuclear Data 17
IV. Data on Delayed Neutrons 25
V. Data on Heavy Elements 39
Acknowledgments 47
References 48

The Analysis of Reactor Noise: Measuring Statistical Fluctuations in Nuclear Systems
N. Pacilio, A. Colombino, R. Mosiello, F. Norelli, and V. M. Jorio

I. Introduction 67
II. Theory 75
III. Models for Interpreting Experiments 90
IV. Epilogue 120
Acknowledgments 127
References 127

On-Line Computers in Nuclear Power Plants - A Review
M. W. Jervis

Abstract 135
I. Introduction 135
II. Computer Functions and Application Classes 142
III. Functions 145

IV. Applications Classes 186
V. Computer Hardware and Software 188
VI. Examples of Systems in Operation 197
VII. Ergonomic Factors and Operator Acceptance 203
VIII. Reliability 206
IX. Obsolescence and Designing for Replaceability 207
X. Project Management 210
XI. Program and Data Security 211
XII. Licensing of Initial Design and Modification During Life 212
XIII. Future Development 213
XIV. Concluding Remarks 215
Acknowledgments 217
References 217

Fuel for the SGHWR
D. O. Pickman, J. H. Gittus, and K. M. Rose

I. Introduction 233
II. Fuel Design 235
III. Operational Experience 246
IV. Performance Models: The Seer-Sleuth Computer Code 259
V. Behavior in Accident Conditions 265
VI. Transient Modeling 273
VII. Future Development 280
References 281

The Nuclear Safety Research Reactor (NSRR) in Japan
M. Ishikawa and T. Inabe

I. Introduction 285
II. NSRR Facility and Experimental Capability 289
III. Experimental Research Program 304
IV. Some Results of the Fuel Failure Experiments 315
V. Future Plans 332
References 333

Practical Usage of Plutonium in Power Reactor Systems
K. H. Puechl

I. Popular Notions about Plutonium 335
II. Historical Perspective and Review of Plutonium Properties 339
III. Plutonium Recycle Alternatives 348
IV. Current Status and Conclusions 392
References 393

Computer Assisted Learning in Nuclear Engineering
P. R. Smith

I. Introduction 399
II. Educational Roles of the Digital Computer 400
III. Claims for CAL 403
IV. Computers in Nuclear Engineering Education 406
V. The Role of Graphics in Computer-Assisted Learning 409
VI. Current CAL Projects 420
VII. A Typical CAL Program 438
VIII. Transferring CAL Materials 447
IX. Implementation 451
X. Evaluation 456
XI. Future Prospects for CAL in Nuclear Engineering 462
XII. Summary and Conclusions 464
References 465

Nuclear Energy Centers
M. J. McNelly

I. Introduction 469
II. Fuel Cycle and Facility Integration 489
III. Modular Construction 532
IV. Heat Rejection 550
V. Conclusions 556
References 558

Index 561

Contents of Volume 10

Optimal Control Applications in Nuclear Reactor Design and Operation, *W. B. Terney and D. C. Wade*

Extrapolation Lengths in Pulsed Neutron Diffusion Measurements, *N. G. Sjöstrand*

Thermodynamic Developments, *R. V. Hesketh*

Kinetics of Nuclear System: Solution Methods for the Space-Time Dependent Neutron Diffusion Equation, *W. Werner*

Review of Existing Codes for Loss-of-Coolant Accident Analysis, *Stanislav Fabic*

Controlled Fusion and Reactors of the Tokamak Type, *Robert W. Conn*

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NUCLEAR PHYSICS DATA FOR REACTOR KINETICS

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I. PRINCIPLES OF REACTOR KINETICS AND DATA REQUIREMENTS

The type of information required for reactor kinetics calculations can be seen by inspection of a neutron balance equation, even the following simple one without independent (external) sources of neutrons.

$$\frac{dN(t)}{dt} = \frac{\rho-\beta_{eff}}{\Lambda} N(t) + \sum_{i=1}^{I} \lambda_i C_i(t) \qquad (1)$$

where N(t) is the number of neutrons, Λ is their reproduction time (closely, the neutron lifetime), ρ is the reactivity of the system, β_{eff} is a factor (usually assumed to be constant but which actually changes with time to some extent) which allows for the fraction of neutrons being delayed and for the difference in importance between them and the prompt neutrons, and $\Sigma\lambda_i C_i(t)$ gives the neutrons from the decay of precursors formed earlier. The summation in the last term is carried out over the $\underline{I}$ (usually 6) groups, each with a characteristic decay constant, that are used to give a mathematical representation of the decay of the delayed neutrons after a fission burst.

The last term makes immediately obvious the need for data on delayed-neutron precursors that are well known as a special group of nuclides in the whole range of fission products, but they are not the only ones of importance to reactor kinetics; neutron absorbers such as ^{135}Xe may have a marked effect on ρ, and therefore have to be included. Contained in ρ are also the effects of changes in the heavy elements in the reactor fuels. These changes and the growth of some of the fission product absorbers influence reactor behavior on a much longer time scale than do delayed neutrons or the short-lived absorbers, but we have included them in this review.

The impact of nuclear properties can be seen in more detail from the multigroup form of the neutron balance equation in which the whole spectrum of neutrons is divided into a number of energy groups, with a balance equation such as equation (2) applying to each; information on the delayed-neutron precursors is provided by equation (3).

$$
\begin{aligned}
\frac{1}{v^g}\frac{\partial \phi^g(\underline{r},t)}{\partial t} &= \nabla\cdot D^g(\underline{r},t)\,\nabla\phi^g(\underline{r},t) \\
&- \Sigma_a^g(\underline{r},t)\,\phi^g(\underline{r},t) \\
&+ \sum_{g'=g+1}^{G} \Sigma_s^{g'\to g}(\underline{r},t)\,\phi^{g'}(\underline{r},t) \\
&+ \sum_{m=1}^{M} (1-\beta_m)\,\chi_m^g \sum_{g'=1}^{G} \nu_m^{g'}\,\Sigma_{f_m}^{g'}(\underline{r},t)\,\phi^{g'}(\underline{r},t) \\
&+ \sum_{m=1}^{M}\sum_{i=1}^{I} \chi_m^{i,g}\,\lambda_m^i\, C_m^i(\underline{r},t) \qquad (2)
\end{aligned}
$$

$$
\frac{dC_m^i(\underline{r},t)}{dt} = \beta_m^i \sum_{g'=1}^{G} \nu_m^{g'}\,\Sigma_{f_m}^{g'}(\underline{r},t)\,\phi^{g'}(\underline{r},t) - \lambda_m^i\, C_m^i(\underline{r},t) \qquad (3)
$$

In these equations, m represents one of M fissioning nuclides, g represents one of G energy groups, and i represents one of I delayed-neutron groups, and:

$\phi^g(\underline{r},t_\ell)$	is the flux in energy group g at the point $\underline{r}$ and at time t
D^g	is the group diffusion constant
Σ_a^g	is the cross-section for removal from group g, and includes absorption and scattering
$\Sigma_s^{g'\to g}$	is the cross-section for scattering from group g' to group g (down scattering assumed)
$\Sigma_{f_m}^{g'}$	is the fission cross-section of nuclide m for neutrons in group g'
$\nu_m^{g'}$	is the number of neutrons for each of these fissions
$\beta_i (= \sum_i^I \beta_m^i)$	is the fraction of neutrons emitted by fission of the nuclide m which are delayed (β_m^i is the delayed neutron fraction emitted from nuclide m into delayed neutron group i)
χ_m^g	is the fraction of fission neutrons that enter group g
$\chi_m^{i,g}$	is the fraction of delayed neutrons that enter group g
C_m^i	is the concentration of precursor i from fissioning nuclide m

The roles of the various cross-sections are clear from the equations, as is the need for information on the energies of delayed neutrons (and of the prompt ones, for that matter) as well as on their yields and decay constants. The actual selection of the necessary nuclear data obviously becomes very complex and extensive, and because of this, the main objective of this paper has been more to direct readers to the sources of data rather than to attempt to present the data themselves. This is the reasoning behind the inclusion of comments on the various computer libraries that are now available; in addition to the libraries, the compilation of neutron cross-sections by the Brookhaven National Laboratory (BNL325) (*1,2*) and the CINDA bibliography(*3*) are important sources of information. Actual data have been included when necessary to illustrate main features, when they could be kept within acceptable bounds.

It is important to realize that full reactor kinetics calculations embrace much more than nuclear information and, indeed, more than the properties of a reactor itself; changes in coolant temperature, for example, depend not only on flux changes in the reactor but also on the equipment outside the reactor core. Changes in temperature within the core alter densities, and consequently, macroscopic cross-sections, as well as neutron spectra, and thus influence reaction rates. The speed with which a temperature change affects reactivity depends, of course, on where it occurs, and on the materials and structure of a reactor. When additionally, it is realized that equations (2) and (3) must be integrated over the whole of a reactor and thus also introduce its structural features, it becomes clear that kinetic calculations in their entirety must be related to specific systems. Nothing on these matters has been included in this paper, which concentrates on the explicitly nuclear aspects of the data for reactor kinetics.

II. EVALUATED NUCLEAR DATA LIBRARIES

Few, if any, kinetics calculations have the ability or the necessity to use all the data now available in the several libraries. They are usually performed with multigroup data sets which, in the case of complex neutronics-hydraulics calculations, may be limited to very few groups indeed for the sake of computer storage. However, in order to generate the flux-averaged cross-section sets, it is necessary to use the more detailed information in the libraries. A frequent situation that may arise is that the library data, which are fairly generally available, are collapsed into group format and then adjusted to fit the results of integral measurements. Reactor manufacturers often regard these adjusted group data sets as proprietary information (Poncelet et al. (*4*)) so that these are not available outside the company. As in addition, the group data sets are so many and various, it is not possible, as mentioned, to contemplate a complete review of data that are the immediate input to any kinetics program. It is more sensible to step back from the group data to the basic libraries that are more widely available and not limited to particular reactor systems; Pearlstein has discussed the development of the libraries in a previous paper in this series (*5*). The whole question of whether the private data set used by a reactor vendor to calculate core characteristics should be available to a customer is discussed in the paper by Poncelet et al. on "Proprietary, Standard and Government-supported Nuclear Data Bases" (*4*). Table I is a list of the nuclides for which neutron cross-sections are given in the five most frequently used libraries.

A. ENDF - Evaluated Nuclear Data File

Probably the most extensive and widely-used file is the ENDF/B file maintained by the National Neutron Cross-Section Center (NNCSC) at the Brookhaven National Laboratory. It has passed through several versions since 1968, the current one being ENDF/B IV which was released in 1974. Descriptions of data formats and procedures can be found in Garber et al. (*6*); this paper, the latest of a series, brings together the descriptions of neutron and photon formats. Two different evaluated data libraries are kept at the NNCSC; the ENDF/A library consists of evaluations of data that may not be complete in that only limited energies and cross-sections are covered; also, more than one evaluation of the same cross-section for the same nuclide may appear in the A library. The B library, on the other hand, is complete, and a single evaluation is presented for each cross-section of a particular material. The task of selecting the data sets to be contained in the B library is undertaken by the Cross-section Evaluation Working Group, a body composed of representatives of the U. S. national laboratories, the reactor vendors and other interested parties. From time to time they may recommend a change in the data set which represents the cross-section for a particular material; the recommendation may arise from users indicating the need for an improved representation of a cross-section, or it may arise from new and significant experimental results, either from

TABLE I

LIST OF NUCLIDES FOR WHICH NEUTRON CROSS-SECTIONS ARE AVAILABLE IN LIBRARIES

		ENDF/B IV	ENDL	UKNDL	KEDAK 3	SOKRATOR
1	H	1, 2, 3	1, 2, 3	1 in H_2O 2 in D_2O 3	1, 1 in H_2 1in H_2O, 2	2, 3
2	He	3, 4	3, 4	3, 4	3, 4	3, 4
3	Li	6, 7	6, 7	6, 7		6, 7
4	Be	9	7, 9	9		9
5	B	10, 11	10, 11	10, 11		natural
6	C	12	12	natural	12	12
7	N	14	14	natural	natural	14
8	O	16	16	natural	16	16
9	F	natural	19	19		19
10	Ne					
11	Na	23	23	23	23	23
12	Mg	natural	natural, 24			24
13	Al	27	27	27	27	27
14	Si	natural	natural	natural		28
15	P		31			31
16	S	32	32			32
17	Cl	natural	natural	natural	natural 35, 37	

TABLE I. (cont'd)

		ENDF/B IV	ENDL	UKNDL	KEDAK 3	SOKRATOR
18	Ar		natural			
19	K	natural	natural	natural		39
20	Ca	natural	natural	natural		natural
21	Sc	45	45			
22	Ti	natural	natural	natural		48
		46, 47, 48				
23	V	natural	51	natural		51
24	Cr	natural	natural	natural	natural	52
					50, 52	
					53, 54	
25	Mn	55	55			natural
26	Fe	natural	natural	natural[a]	natural	natural
		54, 56, 58	54, 58		54, 56	56
					57, 58	
27	Co	59	59			59
28	Ni	natural	natural	natural	natural	59
		58, 60	58		58, 60, 61,	
					62, 64	
29	Cu	natural	natural	natural		65
		63, 65	63	63, 65		
30	Zn		64			64
31	Ga		natural	natural		

TABLE I. (cont'd)

		ENDF/B IV	ENDL	UKNDL	KEDAK 3	SOKRATOR
32	Ge	72, 73 74, 76				
33	As	75				
34	Se	76, 77, 78, 80, 82				
35	Br	78, 81				
36	Kr	78, 80, 82, 83, 84, 85, 86				
37	Rb	85, 86, 87				
38	Sr	86, 87, 88, 89, 90				
39	Y	89, 90, 91				89
40	Zr	90, 91, 92, 93, 94, 95, 96	natural 90	natural		91
41	Nb	93, 94, 95	93	93		93
42	Mo	natural 94, 95, 96, 97, 98, 99, 100	natural	natural	natural 92, 94, 95, 96, 97, 98, 100	96
43	Tc	99				

TABLE I. (cont'd)

		ENDF/B IV	ENDL	UKNDL	KEDAK 3	SOKRATOR
44	Ru	99, 100, 101, 102, 103, 104, 105, 106				
45	Rh	103, 105				
46	Pd	104, 105, 106, 107, 108, 110				
47	Ag	107, 109, 111	107, 109	107, 109		
48	Cd	natural, 108, 110, 111, 112, 113, 114, 115m, 116	natural, 114	natural, 113	natural,	
49	In	113, 115	115			
50	Sn	115, 116, 117, 118, 119, 120, 122, 123, 124, 125, 126	natural			119

TABLE I. (cont'd)

		ENDF/B IV	ENDL	UKNDL	KEDAK 3	SOKRATOR
51	Sb	121, 123, 124, 125, 126				
52	Te	122, 123, 124, 125, 126, 127m, 129m, 130 132				
53	I	127, 129, 130, 131, 135				
54	Xe	124, 126, 128, 129, 130, 131, 132, 133, 134, 135, 136		135		
55	Cs	133, 134, 135, 136, 137				
56	Ba	134, 135, 136, 137, 138, 140	138			