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F. C. Maienschein, Director

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CONTENTS

	<u>Page</u>
Introduction	xv
1. MEASUREMENTS OF NEUTRON CROSS SECTIONS AND RELATED QUANTITIES	
1.0. Introduction	1-3
1.1. Review of Cross-Section Data Important to the Uranium-Plutonium Fuel Cycle in Thermal Reactors	1-5
L. W. Weston	
1.2. An Assessment of the "Thermal Normalization Technique" for Measurement of Neutron Cross Sections vs Energy	1-5
R. W. Peelle and G. de Saussure	
1.3. Neutron Capture and Fission Cross Sections of Plutonium-241	1-6
L. W. Weston and J. H. Todd	
1.4. Multilevel Resonance Parameters of Plutonium-241	1-6
L. W. Weston and J. H. Todd	
1.5. Neutron Capture Cross Section of Plutonium-240	1-6
L. W. Weston and J. H. Todd	
1.6. Measurements of the Average Number of Prompt Neutrons Emitted per Fission of ^{239}Pu and ^{235}U	1-7
R. Gwin, R. Spencer, R. Ingle, J. H. Todd, and H. Weaver	
1.7. Measurement of the Uranium-238 Subthreshold Fission Cross Section for Incident Neutron Energies Between 0.6 and 100 keV	1-7
F. C. Difilippo, R. B. Perez, G. de Saussure, D. K. Olsen, and R. W. Ingle	
1.8. The Neutron-Induced ^{238}U Subthreshold Fission Cross Section for Neutron Energies Between 5 eV and 3.5 MeV	1-7
R. B. Perez, F. C. Difilippo, G. de Saussure, and R. W. Ingle	
1.9. Measurement of the Uranium-238 to Uranium-235 Fission Cross-Section Ratio for Neutron Energies Between 0.1 and 25 MeV	1-10
F. C. Difilippo, R. B. Perez, G. de Saussure, D. K. Olsen, and R. W. Ingle	
1.10. Statistical Tests for the Detection of Intermediate Structure: Application to the Structure of the ^{238}U Neutron Capture Cross Section Between 5 keV and 0.1 MeV	1-10
R. B. Perez, G. de Saussure, R. L. Macklin, and J. Halperin	
1.11. Measurement and Resonance Analysis of 150-m Neutron Transmissions Through Uranium-238	1-10
D. K. Olsen, G. de Saussure, R. B. Perez, F. C. Difilippo, R. W. Ingle, and H. Weaver	
1.12. Note on ^{238}U + n: Resolved Resonance Energies	1-11
D. K. Olsen, G. de Saussure, R. B. Perez, F. C. Difilippo, and R. W. Ingle	
1.13. SIOB: A Fortran Code for Least-Squares Shape Fitting Several Neutron Transmission Measurements Using the Breit-Wigner Multilevel Formula	1-11
G. de Saussure, D. K. Olsen, and R. B. Perez	
1.14. Cross Sections for the $\text{Nb}(n,\alpha)$ and $\text{Nb}(n,\gamma)$ Reactions Between 1 and 20 MeV	1-11
G. L. Morgan and F. G. Perey	
1.15. Nuclear Data Needs for Fast Breeder Reactor Shielding	1-12
C. M. Obloz and F. G. Perey	

	<u>Page</u>
1.16. Recent Results for ^{55}Ni and ^{56}Fe at OKELA	1-12
F. G. Perey, G. T. Chapman, W. E. Kinney, and C. M. Perey	
1.17. The $\text{Ti}(n,\gamma)$ Reaction Cross Section for Incident Neutron Energies Between 0.3 and 20.0 MeV	1-12
G. L. Morgan and D. C. Larson	
1.18. Measurement of the Neutron Total Cross Section of Natural Magnesium from 9 keV to 39 MeV	1-13
D. C. Larson, J. A. Harvey, N. W. Hill, and H. Weigmann	
1.19. The $\text{Na}(n,\gamma)$ Reaction Cross Section for Incident Neutron Energies Between 0.2 and 20.0 MeV	1-13
D. C. Larson and G. L. Morgan	
1.20. Cross Sections for the $^{14}\text{N}(n,p)$, (n,α) , and (n,α_1) Reactions from 0.5 to 15 MeV	1-13
G. L. Morgan	
1.21. Cross Sections for the $^7\text{Li}(n,\alpha)$ and $^7\text{Li}(n,\alpha_1)$ Reactions Between 1 and 20 MeV	1-14
G. L. Morgan	
1.22. Absolute Measurement of the $^7\text{Li}(n,\alpha_1)^7\text{Li}^*$ (0.478-MeV) Cross Section	1-14
D. K. Olsen, G. L. Morgan, and J. W. McConnell	
1.23. A Proton-Recoil Telescope for In-Line Absolute Neutron Fluence Measurements	1-16
G. L. Morgan, D. K. Olsen, and J. W. McConnell	
1.24. Instruments for Use of ^6Li as a Standard	1-16
L. W. Weston	
1.25. Status Report to the DOE Nuclear Data Committee	1-17
F. G. Perey and J. C. Gentry	
1.26. Fission-Product Energy Release for Times Following Thermal-Neutron Fission of ^{235}U Between 2 and 14,000 Seconds	1-17
J. K. Dickens, J. F. Emery, T. A. Love, J. W. McConnell, K. J. Northcutt, R. W. Peelle, and H. Weaver	
1.27. Delayed Beta- and Gamma-Ray Production Due to Thermal-Neutron Fission of ^{235}U , Spectral Distributions for Times After Fission Between 2 and 14,000 Sec: Tabular and Graphical Data	1-17
J. K. Dickens, T. A. Love, J. W. McConnell, J. F. Emery, K. J. Northcutt, R. W. Peelle, and H. Weaver	
1.28. Fission-Product Energy Release for Times Following Thermal-Neutron Fission of ^{239}Pu Between 2 and 14,000 Seconds	1-18
J. K. Dickens, J. F. Emery, T. A. Love, J. W. McConnell, K. J. Northcutt, R. W. Peelle, and H. Weaver	
1.29. Fission-Product Energy Release for Times Following Thermal-Neutron Fission of ^{241}Pu Between 2 and 14,000 Seconds	1-18
J. K. Dickens, J. F. Emery, T. A. Love, J. W. McConnell, K. J. Northcutt, R. W. Peelle, and H. Weaver	
1.30. Fission Yields for Thermal-Neutron Fission of ^{241}Pu	1-19
J. K. Dickens	
1.31. ORCODE-22: A Computer Routine to Control a Nuclear Physics Experiment by a PDP-15/4-AMAC System, Written in Assembler Language and Including Many New Routines of General Interest	1-19
J. K. Dickens and J. W. McConnell	
1.32. NSFOOF: A Generalized Nonlinear Least-Squares Fitting Program for Use with ORNL ODF Files	1-20
R. M. Charlton and D. K. Olsen	

	<u>Page</u>
1.33. Calculations Pertaining to the Design of a Prebuncher for a 150-MeV Electron Linear Accelerator: Radial Motion	1-20
R. G. Alsmiller, Jr., F. S. Alsmiller, and J. Barish	
2. CROSS-SECTION THEORY, EVALUATIONS, AND EVALUATION TECHNIQUES	
2.0. Introduction	2-3
2.1. R-Matrix Evaluation of Carbon as a Neutron Scattering Standard Below 2 MeV	2-4
C. Y. Fu and F. G. Perey	
2.2. Multistep Hauser-Feshbach Codes with Precompound Effects: A Brief Review of Current and Required Developments and Applications Up to 40 MeV	2-4
C. Y. Fu	
2.3. Perturbation Formalism for the Complex Poles and Widths of the Transition Matrix with an Application to Intermediate Structure Phenomena	2-4
R. B. Perez, G. de Saussure, D. K. Olsen, and F. C. Difilippo	
2.4. On the Expansion of the Propagator in Power Series of the Coupling Constant	2-5
R. B. Perez	
2.5. The Effect of the Energy Dependence of the Neutron Widths on the Calculation of Average Reaction Cross Sections	2-5
F. C. Difilippo and R. B. Perez	
2.6. Evaluation of the ^{238}U Neutron Cross Sections for Incident Neutron Energies up to 4 keV	2-5
G. de Saussure, D. K. Olsen, R. B. Perez, and F. C. Difilippo	
2.7. Evaluation of the ^{232}Th Neutron Capture Cross Section Above 3 keV	2-6
G. de Saussure and R. L. Macklin	
2.8. Evaluation of $^{232}\text{Th}(n,n')$, $(n,2n)$ and $(n,3n)$ Cross Sections	2-6
C. Y. Fu	
2.9. Review and Combination of Experimental Results for Neutron Emission per Fission of ^{232}Th	2-6
R. Gwin	
2.10. Least-Squares Dosimetry Unfolding: The Program STAY'SL	2-6
F. G. Perey	
2.11. Uncertainty Analysis of Dosimetry Spectrum Unfolding	2-7
F. G. Perey	
2.12. Spectrum Unfolding by the Least-Squares Method	2-7
F. G. Perey	
2.13. The Data Covariance Files for ENDF/B-V	2-8
F. G. Perey	
2.14. Expectations for ENDF/B-V Covariance Files: Coverage, Strength, and Limitations	2-8
F. G. Perey	
2.15. Covariance Matrices of Experimental Data	2-8
F. G. Perey	
3. CROSS-SECTION PROCESSING, TESTING, AND SENSITIVITY ANALYSES	
3.0. Introduction	3-3
3.1. A Review of Multigroup Nuclear Cross-Section Processing: Proceedings of a Seminar-Workshop, Oak Ridge, Tennessee, March 14-16, 1978	3-5
D. K. Trubey and H. R. Hendrickson	

	<u>Page</u>
3.2. Cross-Section Probability Tables in Multi-Group Transport Calculations	3-5
D. E. Cullen, E. F. Flechaty, R. J. Doyas, C. R. Weisbin, and J. E. White	
3.3. Experience in Developing and Using the VITAMIN-C 171-Neutron, 36-Gamma-Ray Group Cross-Section Library	3-5
R. W. Roussin, C. R. Weisbin, J. E. White, R. Q. Wright, N. M. Greene, W. E. Ford III, J. B. Wright, and B. R. Diggs	
3.4. The Shielding Factor Method for Producing Effective Cross Sections: MINX/SPHINX and the CCIC Interface System	3-6
R. E. MacFarlane, C. R. Weisbin, and N. C. Paik	
3.5. Calculations of Several CSEWG Shielding Benchmarks with Preliminary ENDF/B-V Data	3-6
R. E. Maerker	
3.6. Projections of ENDF/B Version V Performance for Fast and Thermal Reactors Using Sensitivity Coefficients	3-8
C. R. Weisbin, J. H. Marable, J. Hardy, Jr., and R. D. McKnight	
3.7. Sensitivity Coefficient Compilation for CSEWG Data Testing Benchmarks	3-8
C. R. Weisbin, J. H. Marable, J. Hardy, Jr., and R. D. McKnight	
3.8. Compilation of Multigroup Cross-Section Covariance Matrices for Several Important Reactor Materials	3-9
J. D. Drischler and C. R. Weisbin	
3.9. Recent Developments in Sensitivity Analysis in the U.S.	3-9
E. M. Oblo	
3.10. Data Adjustment: A Cautiously Optimistic View for the Improvement of Design Performance Calculations and Data Assessment	3-10
C. R. Weisbin, E. M. Oblo, J. H. Marable, and M. Salvatores	
3.11. Sensitivity and Uncertainty Analysis for Functionals of the Time-Dependent Nuclide Density Field	3-12
M. L. Williams and C. R. Weisbin	
3.12. Sensitivity Theory for Depletion Analysis	3-12
M. L. Williams, J. R. White, J. H. Marable, and E. M. Oblo	
3.13. Sensitivity Theory for General Nonlinear Algebraic Equations with Constraints	3-13
E. M. Oblo	
3.14. Sensitivity Theory for Reactor Thermal-Hydraulics Problems	3-13
E. M. Oblo	
3.15. High Order Effects in Cross Section Sensitivity Analysis	3-14
E. Greenspan, Y. Karni, and D. Gilai	
3.16. Second-Order Generalized Perturbation Theory for Source-Driven Systems	3-14
E. Greenspan, D. Gilai, and E. M. Oblo	
3.17. SENDIN and SENTINEL: Two Computer Codes to Assess the Effects of Nuclear Data Changes	3-14
J. H. Marable, J. D. Drischler, and C. R. Weisbin	
3.18. COVERT and CAVALIER: Two Computer Codes for Estimating Uncertainties of Calculated Neutronics Parameters Using Standard Interface Files SENPRO and COVERX	3-15
J. D. Drischler, J. H. Marable, and C. R. Weisbin	
3.19. Uncertainties in the Breeding Ratio of a Large LMFBR	3-15
J. H. Marable and C. R. Weisbin	
3.20. Calculation of the Flux Covariance Matrix for the National Bureau of Standards Intermediate Energy Standard Neutron Field (NBS-INSF)	3-16
B. L. Broadhead, H. L. Dodds, Jr., J. L. Lucius, F. G. Percy, and C. R. Weisbin	

	<u>Page</u>
3.21. Sensitivity and Uncertainty Analysis Applied to the NBS-ISNF	3-17
B. L. Broadhead and J. H. Marable	
3.22. Sensitivity and Uncertainty Analysis for Mixed-Oxide Thermal Reactor Parameters	3-18
R. L. Childs, J. L. Lucius, V. C. Baker, and J. D. Drischler	
3.23. A Compendium of Energy-Dependent Sensitivity Profiles for the TRX-2 Thermal Lattice	3-20
E. T. Tomlinson, J. L. Lucius, and J. D. Drischler	
3.24. Sensitivity Analyses of Fast Reactor Systems Including Thorium and Uranium	3-21
J. H. Marable and C. R. Weisbin	
3.25. Advances in Fast Reactor Sensitivity and Uncertainty Analysis	3-22
J. H. Marable and C. R. Weisbin	
3.26. Application of Sensitivity and Uncertainty Methodology to Fast Reactor Integral Experiment Analysis	3-23
C. P. Weisbin, E. M. Oblow, J. H. Marable, R. W. Peelle, and J. L. Lucius	
3.27. Two-Dimensional Reactor Sensitivity Analysis Using Discrete-Ordinates Transport Theory	3-23
R. L. Childs, J. H. Marable, and C. R. Weisbin	
3.28. Sensitivity Analysis Applied to an Integral Test of Niobium Cross Sections	3-26
D. T. Ingersoll and B. W. Wehring	
3.29. A Comparison of Experimental and Calculated Resonance Integrals for UO_2 and ThO_2 Isolated Rods	3-26
V. C. Baker and J. H. Marable	
3.30. Analytical Inequalities Satisfied by the Cross-Section Self-Shielding Factors	3-29
D. G. Cacuci and M. A. Bjerke	

4. INTEGRAL EXPERIMENTS AND THEIR ANALYSES

4.0. Introduction	4-3
4.1. Analysis of a Neutron Scattering and Gamma-Ray Production Integral Experiment on Aluminum for Neutron Energies from 1 to 15 MeV	4-5
S. N. Cramer and E. M. Oblow	
4.2. Analysis of a Neutron Scattering and Gamma-Ray Production Integral Experiment on Silicon Dioxide for Neutron Energies from 1 to 15 MeV	4-5
S. N. Cramer and E. M. Oblow	
4.3. An Integral Measurement to Test Evaluated Neutron Cross Sections for Carbon	4-6
G. L. Morgan	
4.4. The Thermoluminescent Dosimeter Measurement of Gamma-Ray Energy Deposition Distributions in Reactor Shields	4-6
P. N. Stevens, C. E. Clifford, F. J. Muckenthaler, and W. Yoon	
4.5. Radiation Heating Studies in a Stainless Steel and Sodium Shield	4-7
R. E. Maerker, F. J. Muckenthaler, and C. E. Clifford	
4.6. Analyses of TSF Experiments on Radiation Heating in a Stainless Steel-Sodium CRBR Radial Shield Mockup Using a 32-inch Diameter Collimated Beam Source	4-8
R. E. Maerker	
4.7. Measurements and Calculations of the ORNL CRBR Upper Axial Shield Experiment	4-9
R. E. Maerker, F. J. Muckenthaler, and C. E. Clifford	

	<u>Page</u>
4.8. A Measurement of the Neutron Streaming in a CRBR Prototypic Coolant Pipe Chaseway	4-9
F. J. Muckenthaler and C. E. Clifford	
4.9. Discrete Ordinate Analysis of the TSF Experiment on Neutron Streaming in a CRBR Prototypic Coolant Pipe Chaseway	4-9
R. E. Maerker	
4.10. Measurements and Analysis of the CRBR Inconel-Stainless Steel Radial Shield Experiment	4-10
R. E. Maerker and F. J. Muckenthaler	
4.11. Final Analysis of a TSF Experiment on Neutron Streaming Through a Lattice of GCFR-Type Fuel Pins	4-10
C. O. Slater, M. B. Emmett, and F. J. Muckenthaler	
4.12. Design of a Prototypic GCFR Grid-Plate Shield Confirmation Experiment	4-11
D. T. Ingersoll, C. O. Slater, and M. L. Williams	
4.13. The GCFR Grid-Plate Shield Design Experiment	4-13
F. J. Muckenthaler, J. L. Hull, and J. J. Manning	
4.14. Shielding Experimental Methods	4-14
F. C. Maienschein and F. J. Muckenthaler	
4.15. U.S. Integral and Benchmark Experiments	4-15
F. C. Maienschein, R. E. Maerker, and F. J. Muckenthaler	
4.16. Investigation of Steel-Sodium-Iron Shields	4-15
E. M. Obloy and R. E. Maerker	

5. DEVELOPMENT OF METHODS FOR SHIELD AND REACTOR ANALYSES

5.0. Introduction	5-3
5.1. The DOT-IV Two-Dimensional Discrete Ordinates Transport Code with Space-Dependent Mesh and Quadrature	5-4
W. A. Rhoades, D. B. Simpson, R. L. Childs, and W. W. Engle, Jr.	
5.2. A New Weighted-Difference Formulation for Discrete Ordinates Calculations	5-4
W. A. Rhoades and W. W. Engle, Jr.	
5.3. Comparison of Rebalance Stabilization Methods for Two-Dimensional Transport Calculations	5-6
W. A. Rhoades, R. L. Childs, and W. W. Engle, Jr.	
5.4. A Comparison of Codes for Deep Penetration Transport Studies	5-8
W. A. Rhoades and E. T. Tomlinson	
5.5. CARP, A Computer Code and Albedo Data Library for Use by BREESE, the MORSE Albedo Package	5-10
M. B. Emmett and W. A. Rhoades	
5.6. FORM: Perspective Plotting Code for One- and Two-Dimensional Fluxes	5-11
D. T. Ingersoll	
5.7. GOFRR: A Computer Code to Generate Graphical Output of DOT and ANISN Fluxes and Reaction Rates	5-11
F. B. Sadler and D. L. Selby	
5.8. The Generation of a Computer Library for Discrete Ordinates Quadrature Sets: Part I: Fully Symmetric Quadratures and the DOQDP Computer Code, Part II: Common Quadratures and the Quadrature Library	5-11
J. P. Jena, P. J. Erickson, W. A. Rhoades, D. B. Simpson, and M. L. Williams	
5.9. The FANG Angular Folding Code for Channel Theory Analysis	5-12
M. L. Williams and F. B. Sadler	

	<u>Page</u>
5.10. Solid Angle Subtended at a Point by a cylinder	5-12
S. N. Cramer	
5.11. Application of the Fictitious Scattering Radiation Transport Model for Deep Penetration Monte Carlo Calculations	5-14
S. N. Cramer	
5.12. Methods for U.S. Shielding Calculations - Applications to FFTF and CRBR Design	5-14
W. W. Engle, Jr., F. R. Mynatt, and R. K. Disney	
5.13. Analytical Calculations of Neutron Slowing Down and Transport in the Constant-Cross-Section Problem	5-15
D. G. Cacuci	
5.14. Test of the Diffusion Theory; Difference Equations in Slab Geometry (Computer Code VANCER)	5-16
D. R. Vondy and T. B. Fowler	
5.15. VENTURE: A Code Block for Solving Multigroup Neutronics Problems Applying the Finite-Difference Diffusion-Theory Approximation to Neutron Transport, Version II	5-16
D. R. Vondy, T. B. Fowler, and G. W. Cunningham	
5.16. RODMOD - A Code for Control Rod Positioning	5-17
D. R. Vondy and T. B. Fowler	
5.17. Exploiting the Steady State, Continuous Fueling Reactor Model	5-17
D. R. Vondy, G. W. Cunningham, and T. B. Fowler	
5.18. Restrained Acceleration in Iteration	5-19
D. R. Vondy and T. B. Fowler	
5.19. The Code PERTUBAT for Processing Neutron Diffusion Theory Neutronics Results for Perturbation Analyses	5-19
D. R. Vondy and T. B. Fowler	
5.20. Reference Test Problems for the VENTURE Neutronics and Related Computer Codes	5-19
D. R. Vondy and T. B. Fowler	
5.21. Transport/Diffusion Comparisons for Small-Core LMFBK Disruptive Accidents	5-20
E. T. Tomlinson	
5.22. Development of Depletion Perturbation Theory for Coupled Neutron/Nuclide Fields	5-22
M. L. Williams	
5.23. The Computation of Homogenized Group Parameters	5-22
B. A. Worley and A. F. Henry	
6. ANALYSES FOR SPECIFIC SYSTEMS OR APPLICATIONS	
6.0. Introduction	6-3
<u>Liquid-Metal Fast Breeder Reactor Program</u>	
6.1. VENTURE Code Results for the Large-Core Sodium-Voiding Benchmark Problems	6-6
D. R. Vondy and T. B. Fowler	
6.2. Analysis of Streaming Through the Primary Coolant Pipe Chaseway of the Clinch River Breeder Reactor	6-7
M. L. Williams and W. W. Engle, Jr.	
6.3. Use of Variational Techniques for the Estimation of Neutron Detection Efficiency	6-9
S. C. Lin, J. C. Robinson, and D. L. Selby	

	<u>Page</u>
6.4. Evaluation of a Variational Method for Estimation of Neutron Detection Efficiency	6-10
D. L. Loyd, J. C. Robinson, and D. L. Selby	
6.5. Analysis of ZPPR-5 Source Level Flux Monitor (SLFM) Experiment	6-12
D. L. Selby, G. F. Flanagan, and D. L. Loyd	
6.6. Assessment of Gel-Sphere-Pack Fuel for Fast Breeder Reactors: Reactor Physics Considerations	6-14
D. L. Selby	
6.7. Calculational and Experimental Analysis of a ^{239}Pu Sample Irradiated in EBR-II	6-15
M. L. Williams and S. Raman	
6.8. Preliminary Neutronic Study of Actinide Transmutation in a Fast Reactor	6-17
M. L. Williams, J. W. McAdoo, and G. F. Flanagan	

Gas-Cooled Reactor Program

6.9. Interim Report on Core Physics and Fuel Cycle Analysis of the Pebble Bed Reactor Power Plant Concept	6-18
D. R. Vondy	
6.10. Pebble Bed Reactor Core Physics and Fuel Cycle Analysis	6-18
D. R. Vondy and B. A. Worley	
6.11. Analysis of Gas-Cooled Fast Reactor Shield Designs	6-19
D. E. Bartine and L. R. Williams	

Alternate Fuel Cycle Program^e

6.12. Denatured Fast Reactor Symbiosis	6-19
T. J. Burns, J. R. White, D. E. Bartine, J. M. Kallfelz, and D. M. Rowland	
6.13. Alternative Fuel Cycles for Breeder Reactors	6-22
T. J. Burns, D. E. Bartine, J. R. White, J. M. Kallfelz, and D. M. Rowland	
6.14. Preliminary Evaluation of Alternative Fuel Cycle Options Utilizing Fast Breeders	6-25
T. J. Burns and J. R. White	
6.15. Benchmark Experiments with Thorium in Fast Reactor Lattices	6-26
U. Schmocker, K. Gmuer, H. Graf, M. Jermann, C. McCombie, R. Richmond, S. Seth, D. E. Bartine, D. T. Ingersoll, and M. L. Williams	
6.16. The Role of Thorium Fuel Cycles	6-26
P. R. Kasten and D. E. Bartine	
6.17. Thorium Assessment Study Quarterly Progress Reports for Second and Third Quarters Fiscal 1977	6-27
I. Spiewak, D. E. Bartine, T. J. Burns, J. C. Cleveland, W. E. Thomas, and J. R. White	

^eSee also Paper 6.44.

Magnetic Fusion Energy Program⁺

6.18. Macroscopic Cross-Section Sensitivity Study for Fusion Reactor Shielding Experiments	6-28
Y. Seki, R. T. Santoro, E. M. Oblow, and J. L. Lucius	
6.19. Macroscopic Cross-Section Sensitivity Study for the TNS Integral Experiment	6-28
Y. Seki, R. T. Santoro, E. M. Oblow, and J. L. Lucius	
6.20. Cross-Section Sensitivity Analysis of a Streaming Experiment with a Two-Dimensional Model	6-31
Y. Seki, R. T. Santoro, E. M. Oblow, J. M. Barnes, and J. L. Lucius	
6.21. Comparison of One- and Two-Dimensional Sensitivity Calculations for a Fusion Reactor Shielding Experiment	6-31
Y. Seki, R. T. Santoro, E. M. Oblow, and J. L. Lucius	
6.22. Neutronics and Radiation Damage Calculations for Fusion Reactors	6-31
R. G. Alsmiller, Jr., T. A. Gabriel, and R. T. Santoro	
6.23. The Sensitivity of the First-Wall Radiation Damage to Fusion Reactor Blanket Composition	6-32
J. M. Barnes, R. T. Santoro, and T. A. Gabriel	
6.24. The Effects of Fe-Ni-Co-V Structural Alloys on Fusion Reactor Neutronic Performance	6-32
J. M. Barnes, B. L. Bishop, R. T. Santoro, and T. A. Gabriel	
6.25. Sensitivity of Primary Knock-On Atom Spectra and Displacement per Atom Cross Sections to Different Secondary Neutron Energy and Angular Distributions and "In-Group" Weighting Schemes	6-32
T. A. Gabriel and B. L. Bishop	
6.26. Two- and Three-Dimensional Neutronics Analyses of the TFTR Neutral-Beam Injectors	6-33
R. T. Santoro, R. A. Lillie, R. G. Alsmiller, Jr., and J. M. Barnes	
6.27. Design Calculations for a Neutron Collimator for the TFTR	6-35
R. A. Lillie, R. G. Alsmiller, Jr., and J. T. Mihaiczo	
6.28. Neutron-Photon Multigroup Cross Sections for Neutron Energies ≤ 60 MeV	6-37
R. G. Alsmiller, Jr., and J. Barish	
6.29. The Nuclear Performance of Molten Salt Fusion-Fission Symbiotic Systems for Catalyzed DD and DT Plasmas	6-37
M. M. H. Ragheb, R. T. Santoro, J. M. Barnes, and M. J. Saltmarsh	
6.30. Nuclear Performance Calculations for the ELMO Bumpy Torus Reactor (EBTR) Reference Design	6-37
R. T. Santoro and J. M. Barnes	
6.31. Monte Carlo Analysis of the Effects of a Blanket-Shield Penetration on the Performance of a Tokamak Fusion Reactor	6-38
R. T. Santoro, J. S. Tang, R. G. Alsmiller, Jr., and J. M. Barnes	

High-Energy Physics Program

6.32. Particle Fluxes and Energy Deposition in Infinite Homogeneous Air Due to Protons of Energy 1-, 5-, 10-, and 20-GeV	6-38
T. A. Gabriel, B. L. Bishop, J. D. Amburgey, and J. Barish	

⁺See also Paper 6.47.

	<u>Page</u>
6.33. Photon Dose Rate from Induced Activity in the Beam Stop of a 400-GeV Proton Accelerator	6-39
R. G. Alsmiller, Jr., T. A. Gabriel, and J. Barish	
6.34. Calculated Hadronic Transmission Through Iron Absorbers	6-39
T. A. Gabriel and B. L. Bishop	
6.35. Uranium Liquid-Argon Calorimeters: A Calculational Investigation	6-39
T. A. Gabriel	
6.36. CALOR: A Monte Carlo Program Package for the Design and Analysis of Calorimeter Systems	6-40
T. A. Gabriel, J. D. Amburgey, and B. L. Bishop	
6.37. High Energy ($E < 1000$ GeV) Intranuclear Cascade Model for Nucleons and Pions Incident on Nuclei and Comparisons with Experimental Data	6-40
H. W. Bertini, A. H. Culkowski, O. W. Hermann, N. B. Gove, and M. P. Cuthrie	
6.38. The Intranuclear-Cascade-Evaporation Model for Nonelastic Nuclear Collisions and Its Application in Biomedical and MFE Research	6-41
T. A. Gabriel and R. G. Alsmiller, Jr.	

Accelerator Breeding Studies

6.39. Medium-Energy Nucleon-Meson Transport Calculations and Electronuclear Fuel Production	6-41
R. G. Alsmiller, Jr., J. Barish, D. E. Bartine, T. J. Burns, and T. A. Gabriel	
6.40. Nucleon-Meson Transport Calculations and the Nuclear Performance of Target-Blanket Assemblies for Electronuclear Fuel Production	6-44
R. G. Alsmiller, Jr.	
6.41. Preliminary Report on the Promise of Accelerator-Driven Natural-Uranium-Fueled Light-Water-Moderated Breeding Power Reactors	6-44
E. Greenspan	

Miscellaneous Studies

6.42. Radiation Transport in Earth for Neutron and Gamma Ray Air-Over-Ground Sources	6-45
R. A. Lillie and R. T. Santoro	
6.43. Development of a Mid-Head Tissue Dose Response Function	6-45
D. K. Trubey, D. E. Bartine, J. V. Pace III, J. R. Knight	
6.44. Molten-Salt Reactors for Efficient Nuclear Fuel Utilization Without Plutonium Separation	6-47
J. P. Engel, W. A. Rhoades, W. R. Grimes, and J. F. Dearing	
6.45. Monte Carlo Simulation of Turbulent Atmospheric Transport: Comparisons with Experimental Data	6-48
R. G. Alsmiller, Jr., F. S. Alsmiller, H. W. Bertini, and C. L. Begovich	
6.46. Nuclear Accident Dosimetry: Calculations and Comparison with Experimental Data	6-50
R. T. Santoro, R. G. Alsmiller, Jr., and J. M. Barnes	
6.47. Calculated Atom Displacement and Gas Production Rates of Materials Using a Fusion Reactor First Wall Neutron Spectrum	6-51
T. A. Gabriel, B. L. Bishop, and F. W. Jiffer	

7. INFORMATION ANALYSIS AND DISTRIBUTION

7.0.	Introduction	7-3
7.1.	Recent Trends in Radiation Shielding: An RSIC Perspective D. K. Trubey, R. W. Roussin, and B. F. Maskewitz	7-9
7.2.	The Radiation Shielding Information Center: A Source of Computer Codes and Data for Fusion Reactor Materials Studies R. W. Roussin, D. K. Trubey, and B. F. Maskewitz	7-10
7.3.	Bibliography, Subject Index, and Author Index of the Literature Examined by the Radiation Shielding Information Center D. K. Trubey, R. W. Roussin, J. Gurney, and A. B. Gustin	7-10
7.4.	A Review of Information Processing in Medical Imaging: Proceedings of the Fifth International Conference Held at Vanderbilt University, Nashville, Tennessee, June 27-July 1, 1977 A. B. Brill, R. R. Price, W. J. McClain, and M. W. Landay	7-11
7.5.	Computing Technology Resource Sharing W. J. McClain, B. F. Maskewitz, and J. Gurney	7-11
7.6.	NRC Reactor Safety Research Data Repository Established at ORNL B. F. Maskewitz	7-12
7.7.	CREDO: A Reliability Data Analysis Center for Advanced Reactors G. F. Flanagan, N. M. Greene, P. M. Haas, and T. F. Bott	7-12
7.8.	Initial Data Collection Efforts of CREDO: Sodium Valve Failures T. F. Bott and P. M. Haas	7-15

APPENDIX

Scientific and Professional Activities	A-3
Engineering Physics Division Seminars at ORNL	A-9
Publications	A-11
Papers Presented at Scientific Meetings	A-19
Author Index	A-25
Organization Chart	A-27
Engineering Physics Divisional Manpower	A-29

INTRODUCTION

During the approximately 21-month period covered by this progress report substantial changes have occurred which have affected almost all aspects of the Division's activities. Both within the Division and in the national energy program there have been changes in name, organization, and personnel. At the beginning of the period, the Energy Research and Development Administration was in place at the federal level rather than the Department of Energy. The extent of the uncertainties that have developed in the national nuclear policy was not yet evident at that time. The major concern with proliferation and the investigation of alternative fission reactor fuel cycles as possible approaches to reducing one of the routes to proliferation have increased the requirements for reactor-physics scoping studies and have generated across the country innumerable analyses of reactor systems. Consideration of thorium cycles in which the ^{233}U fuel could be denatured by admixture with ^{235}U has made evident the very uncertain nature of the basic nuclear data for thorium and ^{233}U . The fact that these materials were ignored in virtually all measurement programs for 15 years or so is reflected in uncertainties which are much larger than those attainable by current experimental techniques.

A New Name - During the same time period when many additional reactor types were considered, there has been a strong drift away from fission energy sources, thus constraining rather strictly the possibilities for contributing in the general area of reactor physics and shielding. Therefore, we have attempted to intensify our efforts to redirect activities from reactor physics into adjacent areas in which we appear to have particular or potential competence. Our old name of Neutron Physics was never fully appropriate for the work of the Division and it became increasingly less so. In searching for a new name, we were guided in part by the fact that "systems analysis" appears in most definitions of engineering physics. The analysis of complex systems which involve detailed mathematical modeling and intensive computer use is considered to be a desired direction for extension of our work. Thus our new name of Engineering Physics. We have moved in a small way into work of this type. As examples, consider our activities related to the reliability of fission reactors and reactor systems or our major contribution to NASAP, which is a report on the feasibility of a denatured ^{233}U cycle. Numerous systems studies of the symbiosis possible with the coupling of diverse reactor types are included there.

Reorganizations - Since the last information meeting in May, 1977, the organization of the Division has changed as well as the name (see chart on p. A-26). The most important changes have included splitting the Nuclear Engineering Analysis Section into three groups and combining all experimental programs into one section. Overall, there continues a small trend toward a relative increase in the fraction of our work which is devoted to analysis. This is a trend that is common to most parts of the Laboratory and one that obviously has to be limited in the long term if the usefulness of the national laboratory is to survive. The combination of the experimental groups was forced in part by the reduced pace of operations at the Tower Shielding Facility to a part-time basis. Integration of the integral experiments for fission-reactor shielding at the TSF and those for fusion-reactor shielding in Bldg. 6025 has been introduced. For example, shielding materials (which tend to be quite expensive) are now shared between the two facilities. Closer integration of the integral and differential measurement programs should make more effective the sharing of personnel and instruments among the programs.

Division of the analytical section helps to accommodate the growth which had made it rather unwieldy in size and allows formation of a group to concentrate on development of methods, including those used for data processing and sensitivity analysis. This group also investigates the extensions of sensitivity analysis to novel areas. The most promising new application currently is the use of generalized perturbation theory for reactor-safety analysis and experiment design.

Forming two analytical groups concerned primarily with applications has allowed the concentration of new efforts by one group upon the extension of our activities into reactor safety and reliability and now into risk-benefit analysis for energy systems. The other group has expanded its analysis of thermal-reactor systems to include the pebble-bed gas-cooled reactor.

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Trends in Work Areas - These overall trends and the work of the Division can be visualized numerically by noting that six years ago one-half of the work of the Division was devoted to base-technology physics studies for the Liquid Metal Fast Breeder Reactor and this was all of the fission-reactor-connected work that we carried out. During the current year, LMFER base-technology physics is expected to be less than 20% of the effort, but the fission-reactor total has grown to nearly two-thirds. The other fission-reactor work consists of about 10% reactor safety, 15% breeder-reactor projects (including the Clinch River Breeder Reactor and the Gas-Cooled Fast Breeder Reactor), 10% alternate-cycle studies, and about 10% for the Nuclear Regulatory Commission. During the same period, our work in the basic energy sciences has remained at just over 20%. Our work for the Department of Defense has decreased by more than a factor of 3 to about 5%. Fusion-energy support has increased by a factor of 3 but still equals only 6% of the total.

In the reactor-physics area one of our greatest contributions has been the integration at the working level of elements of nominally diverse programs. Thus we have created over the years limited shielding programs for various government agencies and developed these into a reasonably coherent effort. The application of sensitivity analysis in reactor physics offers another good example where an essential vertical integration is being achieved. Thus there is now nearly unparalleled communication among the measurers and evaluators of nuclear data, the groups responsible for developing methods for cross-section processing and data testing, the data processors, the developers of analytical methods, groups performing integral experiments (in the shielding area), applications groups carrying out design-support or scoping studies, and information-analysis and distribution centers. These activities obviously have a variety of sponsors with rather diverse goals. It will obviously require much time to achieve anything like comparable integration of efforts in new areas.

Ceilings and Personnel Turnover - Present and future redirections of effort will be constrained rather sharply, in ways which can only be perceived imprecisely at present, with the imposition of personnel ceilings on the national laboratories. The redundancy of both ceilings on personnel and funding restraints may not be as important as the forced redirection of effort. In a Laboratory which sees its principal new directions of research as utilization of coal and the expanded reprocessing of reactor fuels, the allocation of a share of the ceiling to Engineering Physics is sharply constrained.

The imposition of personnel ceilings might cause problems of excessive stability of parts of the staff with limited opportunity for the introduction of new people. We recognize that this may be a problem and that we must attempt to devise methods of coping with it. In this connection, it is interesting to note that the turnover rate of both scientific and technical personnel in the Division over the past decade has been low, of the order of 4%/yr for the experimental groups. This is about the same as that for the Laboratory as a whole. In the analytical groups, on the other hand, the rate is remarkably higher, being consistently about 15% per year. This large difference may be explained by the alternative opportunities available for individuals with experience with measurement versus analysis, by the long-term nature of nuclear data goals, and by the existence of prime experimental facilities.

About This Report - In general, in an era of substantial redirection and uncertainty in goals, it would have been easy to become lost and work ineffectively. We have tried to recognize the importance of persevering toward those goals which are clearly established and of maintaining high standards of research. We hope that the material which follows reflects such perseverance and high standards and our pride in them. The report surely reflects high output and the coherence in diversity which characterizes our Division's work.

Most of our work is represented by abstracts and summaries of talks. By the very nature of publication of scientific work such accounts tend to be delayed and to appear fragmented. In order to provide a brief indication of the current status of the various areas of our work and a somewhat coherent overview of each area, the following sections are each preceded by a brief introduction.

Variation with Time of Activities of the Engineering Physics Division (in % of total)

Program ^a	FY 74	FY 75	FY 76	FY 77	FY 78	FY 79
LMFBR Physics (including shielding)	50	38	34	28	25	18
LMFBR Safety	—	1	5	9	10	7
CRBR	—	12	10	9	8	5
GCFR	1	4	2	3	6	7
Total BR	51	55	51	49	49	37
Alternate Reactors	2	2	4	8	11	10
NRC	—	5	5	5	5	16
Total Fission	53	62	60	62	65	63
Fusion	2	4	4	6	7	6
BES and HEP	21	20	23	22	21	19
DNA	18	13	9	7	5	5
EPRI	—	—	2	1	1	5
Miscellaneous	6	1	2	2	1	2
TOTAL	100	100	100	100	100	100
Subcontracts	3	2	1	1	6	11

* LMFBR = Liquid-Metal Fast Breeder Reactor; CRBR = Clinch River Breeder Reactor; GCFR = Gas-Cooled Fast Breeder Reactor; NRC = Nuclear Regulatory Commission; BES = Basic Energy Sciences; HEP = High-Energy Physics; DNA = Defense Nuclear Agency; EPRI = Electric Power Research Institute.

1. MEASUREMENTS OF NEUTRON CROSS SECTIONS AND RELATED QUANTITIES

1.0. INTRODUCTION

Though measurements were slowed a little by efforts to complete the ENDF/B-V data file discussed in Chapter 2, a vigorous effort has continued to utilize the powerful Oak Ridge Electron Linear Accelerator (ORELA) neutron source to measure cross sections of importance for many nuclear applications. Our work now includes a concerted effort on cross sections of particular importance for fusion device applications. At the higher energies covered by these measurements, multiple particle emission and charged secondary particles are relatively common. Here ORELA still offers the advantage of permitting studies at all important energies, but intensity becomes a significant limitation. We have made important progress by almost completing the needed measurements of gamma-ray production, by measuring some of the important total cross sections, and by measuring neutron emission spectra for a few materials. We hope to work further in this area and to expand our capabilities by the development of additional measurement techniques in the energy region important to fusion. As an example, note the measurements of cross sections for the $^{14}\text{N}(n,p)$ and (n,α) reactions using nitrogen incorporated into a gas scintillator.

We are now well along on a measurement of the average number of neutrons emitted in spontaneous fission of ^{252}Cf . This experiment has high leverage because this quantity has been used as a reference standard in a large share of all measurements of the energy dependence of $\bar{\nu}$ for the important fission reactor fuel materials. Measurements of this quantity to the necessary ± 0.2 accuracy has proved to be an internationally elusive goal because of the real complexity of the needed experiments, but the measurement under way here at ORNL using the liquid scintillator tank method appears to have a good chance to meet this need because of the strong effort being made and the unique accelerator and data-handling facilities available. Preliminary values obtained are toward the high end of the range of published values, and inclusion of our preliminary results with the world data base has led to improved agreement between results of integral experiments and calculations made using the preliminary ENDF/B-V data base. The related program to measure the energy dependence of the number of neutrons per fission, relative to the ^{235}U for ^{235}Cf , for the important fuel materials in the energy region of importance to fast reactors is now producing final results, and a set of measurements is virtually complete for ^{238}U . If an appropriate chamber can be obtained we propose to turn next to ^{239}Pu .

We have brought nearly to completion our work on fission and capture cross sections for ^{239}Pu and ^{241}Am , and now expect to pause in our work on these nuclides. The extensive work on the fission cross section of ^{235}U is virtually complete, and has led not only to greater knowledge of the cross sections near and above the fission threshold, but also to definitive high-resolution results over the whole subthreshold range down to the lowest-energy resonances. The results seem to imply some conflict with the accepted interpretation of subthreshold ^{235}U fission. Figure 1.2.1 (paper 1.2) shows some of these results with the remarkable structure that can be observed only with high beam intensity, and a detector containing very pure ^{235}U . Our work in the resonance region of ^{235}U has also come to a pause, leaving for now the anomaly that the cross sections for energies above 4 keV are usually represented as being "unresolved," while in fact our existing techniques would permit detailed representation of the resonance structure to at least twice this energy.

The work in fission and capture and resonance-region transmission is now turning to ^{238}U and ^{239}Pu in an effort to bring data for these alternative fuel materials to nearly the same level of accuracy as is now available for ^{235}U and ^{239}Pu . An extensive effort will be required because relatively little work has been performed on these isotopes in recent years, and in some regions capture in ^{238}U has never been measured.

We continue to uncover vital needs in the area of cross sections for structural materials in fast reactors. We believe that world-wide confusion has occurred for structural materials for which the resonance energy regions have been crudely represented in cross-section sets and workers have not been able to discern whether the difficulties have been with the average cross sections or the average resonance self-shielding. A simpler case of a strong need which will be relatively easily met is for the total cross sections of Li and Cr , where integral shield testing has consistently indicated inadequacies and where world experimental data has been surprisingly inadequate. Data for Li have recently been obtained here and plans are under way to study Cr .

The short-term fission-product decay heat program nears completion, with final results published for ^{235}U and ^{239}Pu . The results have estimated uncertainties in the few-percent range, although our results are lower than some other recent experiments by more than this amount. The new data, obtained in three or four laboratories, together with summation results from ENDF/B-IV, have been combined to yield a new standard which is near acceptance by the appropriate AHS committee. Figure 1.0.1 shows some of the results obtained for ^{239}Pu . This experimental program is an example of one having an unusually clear (and high) benefit/cost ratio for a differential nuclear experiment, and we are pleased to have helped with the near-resolution of this important problem. Questions remain for fast-fission decay heat and for ^{235}U .

Fig. 1.0.1. Comparison of fission-product decay power as derived from ORNL experimental results with decay power obtained from summation calculation [T. R. England et al., LA-UR77-627 (1977)]. To construct the curve from the ORNL data, the summation result was assumed at 1.4×10^{-5} and the experimental $\alpha + \gamma$ energy releases for a neutron pulse were successively added, starting with the experimental data for the $(1.0 \times 1.4) \times 10^{-5}$ region. The experimental error bars represent 1 σ uncertainty. The shaded band represents $\pm 1.5\%$ uncertainty assigned to calculation (Schmittroth and Schenter, Nucl. Sci. Eng. 63, 276, 1977).

ORELA remains the most powerful accelerator for wide-range neutron cross-section research, although a new Harwell Linac is about to be commissioned and the rebuilt one at BCMN, Geel, Belgium, is now competitive in some regions. To obtain the greatest possible utility from ORELA, a number of initiatives are under way. The one closest to fruition includes a completely rebuilt electron injector system which includes flytron bunching of

the electron bursts so that we may hope to obtain the maximum neutron production of ORELA even for bursts as narrow as 5 nsec. This improvement is now installed in the debugging stage. Additional efforts to improve ORELA effective power include continuing development work toward more productive electron guns, studies of possible improved target designs specialized for low- and high-energy neutron output, and the possible option to markedly increase the pulse repetition rate at the existing average power limit. We are also checking once more the possible utility of employing pseudorandom pulsing to reduce our dependence on pulse-overlap (^{10}B) filters which strongly reduce neutron output.

1.1. REVIEW OF CROSS-SECTION DATA IMPORTANT TO THE URANIUM-PLUTONIUM FUEL CYCLE IN THERMAL REACTORS*

L. W. Weston

(Summary of paper presented at seminar on "Nuclear Data Problems for Thermal Reactor Applications," Brookhaven National Laboratory, New York, N.Y. 22-24, 1978; to be published in Brookhaven National Laboratory report)

The status of cross sections important for thermal reactors is reviewed. Most cross sections important to the design of U-Pu-fueled light-water reactors with current fuel water ratios are known with reasonable accuracy. Three major remaining problems are (1) the discrepancy in integral measurements and differential calculations of ^{238}U resonance capture in highly self-shielded cases, (2) the discrepancy between irradiation experiments and other measurements of the ratio of capture-to-fission in ^{235}U , and (3) the uncertainty in the parameters of the 1-eV resonance of ^{240}Pu .

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

1.2. AN ASSESSMENT OF THE "THERMAL NORMALIZATION TECHNIQUE" FOR MEASUREMENT OF NEUTRON CROSS SECTIONS vs ENERGY*

R. W. Peelle G. de Saussure

(Abstract of paper presented at International Specialists' Symposium on Neutron Standards and Applications, Gaithersburg, Maryland, March 28-31, 1977, and published in "Neutron Standards and Applications," NBS Special Publication 495, 174-181, 1977)

Refined knowledge of the thermal-neutron cross sections of the fissile nuclides and of the (n, α) reaction standards, together with the reasonably well-known energy dependence of the latter, has permitted resonance-region and low-keV fissile nuclide cross sections to be based on these standards together with count-rate ratios observed as a function of energy using a pulsed "white" source. As one evaluates cross sections for energies above 20 keV, optimum results require combination of cross-section shape measurements with all available absolute measurements. The assumptions of the "thermal normalization method" are reviewed and an opinion is given for the status of some of the standards required for its use. The complications which may limit the accuracy of results using the method are listed and examples are given.

For the $^{235}U(n, f)$ cross section, the option is discussed of defining resonance-region fission integrals as standards. The area of the ~9-eV resonances in this nuclide may be known to one percent accuracy, but at present the fission integral from 0.1 to 1.0 keV is known to no better than about two percent. This uncertainty is based on the scatter among independent results and has not been reduced by the most recent measurements. The uncertainty now limits the accuracy attainable for the $^{235}U(n, f)$ cross section below about 50 keV.

Suggestions are given to indicate how future detailed work might overcome past sources of error.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

1.3. NEUTRON CAPTURE AND FISSION CROSS SECTIONS OF PLUTONIUM-241*

L. W. Weston J. H. Todd[†]*(Abstract of Nucl. Sci. Eng. 26, 464-468, 1978)*

Neutron capture and fission cross sections of ^{241}Pu have been measured from 0.01 eV to 30 keV, and their ratio has been measured up to 250 keV. The cross sections were normalized at thermal-neutron energies (0.02 to 0.03 eV) to the ENDF/B-IV evaluation. The source of pulsed neutrons was the Oak Ridge Electron Linear Accelerator. The gamma-ray detector used to detect capture and fission events was the "total energy detector," which is a low-efficiency detector whose average efficiency is forced to be proportional to the energy of the interacting gamma rays by weighting these events according to their pulse height in the scintillator. Fast-neutron scintillation detectors with pulse-shape discrimination were used to detect fission events. The shape of the neutron flux was measured relative to the $^{10}\text{B}(n,\alpha)$ cross section. The measurements are unique for ^{241}Pu in that absorption and fission were determined directly and simultaneously over a wide neutron energy range rather than indirectly by inferring capture from separate fission and total cross-section measurements. The results indicate that the neutron resonance region of the ENDF/B-IV evaluation underestimates capture by a factor of ~ 2 . Above the resonance region (~ 100 eV), there are no previous measurements of the differential capture cross section. These cross sections are important in plutonium-fueled reactors.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

[†] Instrumentation and Controls Division.

1.4. MULTILEVEL RESONANCE PARAMETERS OF PLUTONIUM-241*

L. W. Weston J. H. Todd[†]*(Abstract of Nucl. Sci. Eng. 26, 116-121, 1978)*

The data previously reported by the authors on the neutron fission and capture cross sections of ^{241}Pu have been simultaneously fit with the Adler formalism to obtain multilevel resonance parameters. The neutron energy range of the fit was 0.01 to 100 eV. The ^{241}Pu cross sections in the resonance region of neutron energies are complex, and the Adler parameters present an efficient method of reprocessing these cross sections, which are important for plutonium-fueled reactors. The parameters represent the data to an accuracy within the quoted experimental errors.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

[†] Instrumentation and Controls Division.

1.5. NEUTRON CAPTURE CROSS SECTION OF PLUTONIUM-240*

L. W. Weston J. H. Todd[†]*(Abstract of Nucl. Sci. Eng. 27, 147-148, 1977)*

The average ^{240}Pu capture cross section was measured from 200 eV to 350 keV. The cross section was normalized at thermal-neutron energies (0.02 to 0.03 eV), and this normalization was confirmed at the 1.06-eV resonance by the black resonance technique. The source of pulsed neutrons was the Oak Ridge Electron Accelerator. The capture gamma-ray detector used was the "total energy detector," which is a modification of the Moxon-Rae detector. The shape of the neutron flux was measured relative to the $^{10}\text{B}(n,\alpha)$ cross section up to 2 keV and the $^6\text{Li}(n,\alpha)$ cross section at higher neutron energies. The results of the measurement define the average capture cross section of ^{240}Pu over a wide neutron energy range to an accuracy of $\sim 8\%$, which is significantly better than previously known. The results indicate that the ENDF/B-IV evaluation is $\sim 25\%$ low above 30-keV neutron energy. The cross section is important in fast plutonium-fueled reactors.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

[†] Instrumentation and Controls Division.

1.6. MEASUREMENTS OF THE AVERAGE NUMBER OF PROMPT NEUTRONS EMITTED PER FISSION OF ^{233}Pu AND ^{235}U *

R. Gwin R. R. Spencer R. W. Ingle[†] J. H. Todd[‡] H. Weaver

(Abstract of ORNL/TM-6246, May 1978)

Measurements of the average number of prompt neutrons $\bar{\nu}_p(E)$ emitted in neutron-induced fission of ^{235}U and ^{239}Pu relative to $\bar{\nu}_p$ for spontaneous fission of ^{252}Cf have been made over an incident neutron energy range from 0.005 eV to 10 MeV. The incident neutrons were generated in pulses at the Oak Ridge Electron Linear Accelerator and their energies defined by time-of-flight methods. The samples were contained in fission chambers located in a large liquid scintillator. Fissions in the samples were identified by the simultaneous detection of prompt fission gamma rays in the scintillator and a pulse from the fission chamber. Fission neutrons were identified by the detection of the prompt gamma-ray cascade resulting from neutron absorption in gadolinium loaded in the liquid scintillator. The value of $\bar{\nu}_p(E)$ for ^{239}Pu at 0.0253 eV agrees with the value given in ENDF/B-IV, whereas the corresponding value for ^{235}U is about 0.7% larger than that given in ENDF/B-IV.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

[†] Instrumentation and Controls Division.

1.7. MEASUREMENT OF THE URANIUM-238 SUBTHRESHOLD FISSION CROSS SECTION FOR INCIDENT NEUTRON ENERGIES BETWEEN 0.6 AND 100 keV*

F. C. Difilippo[†] R. B. Perez G. de Saussure D. K. Olsen R. W. Ingle[‡]

(Abstract of Nucl. Sci. Eng. 68, 162-166, 1977)

We have measured the neutron-induced ^{238}U subthreshold fission cross section in the neutron energy range between 0.6 and 100 keV. A total of 28 fission clusters were identified. The well-known clusters at 721 and 1210 eV appeared resolved into their Class I components.

Average ^{238}U subthreshold fission cross sections were determined and compared with available results in the literature.

We interpreted the measurement in terms of fission doorway (Class II levels) arising from the assumption of the existence of a double-humped fission barrier for the $(^{238}\text{U} + n)$ compound nucleus at large deformations. On the basis of this model, several fission barrier parameters were determined.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

[†] IAEA Fellow now with Comision Nacional de Energia Atomica, Argentina.

[‡] Instrumentation and Controls Division.

1.8. THE NEUTRON-INDUCED ^{238}U SUBTHRESHOLD FISSION CROSS SECTION FOR NEUTRON ENERGIES BETWEEN 5 eV AND 3.5 MeV*

R. B. Perez F. C. Difilippo[†] G. de Saussure R. W. Ingle[‡]

(Summary of paper to be presented at IAEA Symposium on Physics and Chemistry of Fission, Berlin, Federal Republic of Germany, May 14-18, 1979)

A measurement of the ^{238}U fission cross section between 5 eV and 3.5 MeV has been performed at the ORELA Linac. The good resolution of this experiment (0.1% at 721 eV and 2% at 3.5 MeV) together with the high purity of the ^{238}U sample (less than 2 ppm ^{235}U) permitted the identification of 85 fission resonances or clusters of resonances below

200 keV. The shape of the ^{238}U to ^{235}U fission ratio was normalized¹ to 0.435 ± 0.004 for the energy interval from 2.35 to 2.95 MeV. The ENDF/B-IV evaluation of the $^{235}\text{U}(n, f)$ cross section² was used to obtain the ^{238}U fission cross section from the fission ratio.

In the energy region between 5 eV and 600 eV, besides the fission levels at 6.7, 20.9, and 36.7 eV previously reported,³ we have identified 13 additional fission resonances. Figure 1.8.1 shows the fission cross section between 600 eV and 400 keV. All the Class I levels in the fission clusters centered around 721 eV and 1211 eV are well resolved. Comparison with the ^{238}U capture cross-section data⁴ below 2 keV shows that the 27 fission resonances observed in this energy region can be unambiguously assigned to known s-wave resonances in ^{238}U . The fission widths for the 27 resolved Class I levels were computed from their fission areas. The largest fission widths correspond to the 721 eV and 1211 eV levels (1.56 ± 0.06 and 0.36 ± 0.01 MeV, respectively). The smaller fission resonances have fission widths ranging from 10 MeV up to a fraction of an MeV. Because the width of the intermediate structure is comparable to the Class I level spacing ($D_I = 25$ eV), we are in a weak coupling situation.⁵ Hence, we expect that in each fission cluster one quasi-

Fig. 1.8.1. The ^{238}U neutron-induced subthreshold fission cross section ($b \times eV^{1/2}$) for neutron energies between 600 eV and 400 keV. Note that the ordinate is linear between -0.2 and 0.2 $b \times eV^{1/2}$ and logarithmic above 0.2 $b \times eV^{1/2}$.

Class II level will carry most of the fission strength with a fission width nearly that of the Class I level and a neutron width arising from coupling with nearby Class I states. From Lynn's formulation⁶ and the barrier parameters for ^{238}U (Back et al.⁷), we have estimated a neutron width of 0.0005 MeV for the quasi-Class II level in the 721-eV fission cluster, which is more than 2 orders of magnitude smaller than the value of 1.7 ± 0.06 MeV for the s-wave level at 721 eV. Commenting on this discrepancy already noted in the data of Block et al.,⁸ Lynn⁶ suggested that the fission resonances in the 721-eV and 1211-eV clusters were not the s-wave levels to which they had been assigned but new resonances which because of their small neutron widths had not been observed⁹ because measurements of neutron capture or total cross sections. In view of the good energy alignment of the fission levels with known s-wave levels, it seems more likely that the quasi-Class II level is not observed because its small neutron width results in a fission area below the detection threshold of the measurement. Indeed, for reasonable values of the Class II partial widths, the fission area is of the order of 10^{-4} b x eV, and from the spread of the fission strength in the 721-eV and 1211-eV clusters, the Class II level fission width could be of the order of a few electron volts. Such a broad resonance would be particularly difficult to detect.

The small, but significant, fission widths measured for many of the low energy s-wave levels are too large to arise from direct penetration of the double-humped fission barrier, and we have not found a satisfactory explanation for their magnitude.

From the cumulative sum of fission levels as a function of energy, the average level spacing, D_{II} , for the Class II level was found to be 1 ± 0.3 keV. Contrary to expectation, the fission cross section exhibits a substantial enhancement at the opening of the ^{238}U 2^+ neutron inelastic channel at 45 keV. The enhancement of the fission cross section near 120 keV is tentatively interpreted as the contribution from a partially damped vibrational level centered near 145 keV and with a width of about 65 keV.

* Research sponsored by the DOE Office of Nuclear Energy Programs, Reactor Research and Technology Division.

† IAEA Fellow now with Comision Nacional de Energia Atomica, Argentina.

‡ Instrumentation and Controls Division.

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1.9. MEASUREMENT OF THE URANIUM-238 TO URANIUM-235 FISSION CROSS-SECTION RATIO FOR NEUTRON ENERGIES BETWEEN 0.1 AND 25 MeV*

F. C. Difilippo⁺ R. B. Perez G. de Saussure D. K. Olsen R. W. Ingle[‡]

(Abstract of Nucl. Sci. Eng. 66, 42-64, 1977)

The ^{238}U to ^{235}U fission cross-section ratio has been measured for incident neutrons from 0.1 to 25.0 MeV. A ratio shape measurement was normalized by the threshold cross-section method. A fission cross-section ratio from 0.436 to 1.004 was found averaged over neutron energies from 2.35 to 2.95 MeV.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

⁺ IAEA Fellow now with Comision Nacional de Energia Atomica, Argentina.

[‡] Instrumentation and Controls Division.

1.10. STATISTICAL TESTS FOR THE DETECTION OF INTERMEDIATE STRUCTURE: APPLICATION TO THE STRUCTURE OF THE ^{238}U NEUTRON CAPTURE CROSS SECTION BETWEEN 5 keV AND 0.1 MeV*

R. B. Perez G. de Saussure R. L. Macklin⁺ J. Halperin[‡]

(Abstract of paper to be published in Physical Review)

Recent measurements of the ^{238}U neutron capture cross section show a considerable amount of fluctuation between 5 and 100 keV. Statistical tests performed on the measured cross section suggest that the fluctuations are now compatible with the statistical model of nuclear reactions and imply the existence of intermediate structure, which is interpreted in terms of doorway states. The validity of the statistical tests used here is confirmed by numerical experiments with simulated cross sections generated by the Monte Carlo method, in accordance with the statistical method.

The behavior of the capture cross section is compared to those of the subthreshold fission cross section and of the inelastic scattering cross section. The threshold anomaly at the opening of the inelastic channel at 45 keV is seen in the form of a rounded step.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

⁺ Physics Division.

[‡] Chemistry Division.

1.11. MEASUREMENT AND RESONANCE ANALYSIS OF 150-m NEUTRON TRANSMISSIONS THROUGH URANIUM-238*

D. K. Olsen G. de Saussure R. B. Perez F. C. Difilippo⁺
R. W. Ingle[‡] H. Weaver

(Abstract of paper to be published in Nuclear Science and Engineering; also Abstract of ORNL-TH-661, October 1977, and of paper published in Trans. Am. Nucl. Soc. 17, p. 670, 1977)

Neutron transmissions through 0.076-, 0.254-, 1.080-, and 3.620-cm-thick samples of isotopically enriched ^{238}U have been measured from 0.88 to 100.0 keV using a time-of-flight technique over a path length of 150 meters, the ORELA pulsed neutron source, and a 13-mm-thick Li-glass detector. In order to obtain resonance parameters, these transmissions from 0.88 to 4.00 keV have been simultaneously least-squares shape fitted with a multilevel Breit-Wigner cross-section formalism. In general, large neutron widths are obtained, resulting in an s-wave strength function of $(1.208 \pm 0.045) \times 10^{-4}$ over the

interval from 0.0 to 4.0 keV. An absolute energy scale accurate to 2 parts in 10,000 was established.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

† An IAEA Fellow now with Comision Nacional de Energia Atomica, Argentina.

‡ Instrumentation and Controls Division.

1.12. NOTE ON $^{238}\text{U} + n$ RESOLVED RESONANCE ENERGIES*

D. K. Olsen G. de Saussure R. B. Perez
F. C. Difilippo† R. W. Ingle‡

(Abstract of Nucl. Sci. Eng. 68, 141-144, 1978)

Neutron transmission measurements from 100 eV to 170 keV at 150 m through four ^{238}U samples are reported. The energy calibration is described, and the resultant ^{238}U resolved resonance energies are found to be intermediate between those from other workers. In addition, some energies for sharp resonances in ^{23}Na , ^{27}Al , ^{32}S , and ^{206}Pb are given.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

† IAEA Fellow now with Comision Nacional de Energia Atomica, Argentina.

‡ Instrumentation and Controls Division.

1.13. SIOB: A FORTRAN CODE FOR LEAST-SQUARES SHAPE FITTING SEVERAL NEUTRON TRANSMISSION MEASUREMENTS USING THE BREIT-WIGNER MULTILEVEL FORMULA*

G. de Saussure D. K. Olsen R. B. Perez
(Abstract of ORNL/TM-0488, ENDF-001, May 1977)

The FORTRAN-IV code SIOB was developed to least-square fit the shape of neutron transmission curves. Any number of measurements on a common energy scale for different sample thicknesses can be simultaneously fitted. The computed transmission curves can be broadened with either a Gaussian or a rectangular resolution function or both, with the resolution width a function of energy. The total cross section is expressed as a sum of single-level or multilevel Breit-Wigner terms and is Doppler broadened using the fast interpolation routine QUICKW. The number of data points, resonance levels and variables which can be handled simultaneously is limited only by the overall dimensions of two arrays in the program and by the stability of the matrix inversion. In a test problem seven transmissions each with 3750 data points were simultaneously fitted using 74 resonances and 110 variable parameters. The problem took 47 min of CPU time for 3 iterations on an IBM-91.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

1.14. CROSS SECTIONS FOR THE $\text{Nb}(n,\alpha)$ AND $\text{Nb}(n,\gamma)$ REACTIONS BETWEEN 1 AND 20 MeV*

G. L. Morgan F. G. Perey

(Abstract of ORNL/TM-0820, June 1977, and of paper presented at Symposium on Neutron Cross Sections, 10-16 MeV, May 2-6, 1977, Brookhaven National Laboratory, Upton, N.Y.; BNL-WG-00081, 805-817, 1977)

Differential cross sections for the production of secondary neutrons and gamma rays from neutron interactions in niobium have been measured at 129 deg (lab) for incident neutron energies in the range 1 to 20 MeV. An electron linac was used as a white neutron source. Incident neutron energies were determined using time-of-flight techniques for a

*Research sponsored by DOE Office of Basic Energy Sciences, Division of Nuclear Sciences.

 

A review of neutron and gamma-ray cross-section data needs for fast reactor shielding is presented in light of the recent advances made in assessing these needs through sensitivity studies. Total and partial cross sections and energy and angular distribution data for neutrons are surveyed, as well as gamma-ray production cross sections. The strengths and deficiencies of currently available benchmark-quality integral experiments are also discussed with respect to their use in creating adjusted cross-section libraries for design work. The availability of first-round covariance data in ENDF/B-IV and plans for ENDF/B-V are also reviewed. This latter information makes it possible to quantitatively assess the quality of current cross-section data libraries and also puts adjustment and data assessment procedures on a firmer basis.

1.16. RECENT RESULTS FOR Ni AND ^{57}Fe AT ORELA*

[illegible]

Deceased.

G. L. Morgan D. C. Larson

1. *Chlorophyll a* and *Chlorophyll b* were determined by the method of Arar and Collins (1987) using a spectrophotometer (Shimadzu 1601) with 10 mm quartz cuvettes. The optical density was measured at 663 nm and 646 nm, respectively. The concentrations of *Chlorophyll a* and *Chlorophyll b* were calculated using the following equations:

Research supported by the Office of Basic Energy Sciences, Division of Nuclear Sciences.

1.18. MEASUREMENT OF THE NEUTRON TOTAL CROSS SECTION OF NATURAL MAGNESIUM FROM 9 keV TO 39 MeV*

D. C. Larson J. A. Harvey[†] N. W. Hill[‡] H. Weigmann[§]

(Abstract of *IEEE TM-8410*, June 1978.)

The neutron transmission through a 5.09-cm sample of natural magnesium has been measured for neutron energies between 9 keV and 39 MeV. The Oak Ridge Electron Linear Accelerator (ORELA) was used to provide the neutrons, which were detected at the 200-m flight path by a NE-110 proton recoil detector. The experimental results are tabulated and compared with the total cross section in the ENDF/B-IV file for magnesium.

* Research jointly sponsored by Defense Nuclear Agency and DOE Office of Basic Energy Sciences, Division of Nuclear Sciences.

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1.19. THE $\text{Na}(\text{n},\gamma)$ REACTION CROSS SECTION FOR INCIDENT NEUTRON ENERGIES BETWEEN 0.2 AND 20.0 MeV*

D. C. Larson G. L. Morgan

(Abstract of *IEEE TM-8381*, May 1978.)

Differential cross sections for neutron-induced gamma-ray production from sodium have been measured for incident neutron energies between 0.2 and 20.0 MeV. The Oak Ridge Electron Linear Accelerator (ORELA) was used to provide the neutrons and a NaI spectrometer to detect the gamma rays at 125 deg. The data presented are the doubly differential cross section, $d^2\sigma/dE_\gamma dE_n$, for gamma-ray energies between 0.35 and 10.6 MeV for coarse intervals in incident neutron energy. The integrated yield of gamma rays of energies greater than 350 keV with higher resolution in the neutron energy is also presented. Finally, the excitation function for the 440-keV gamma ray is given for incident neutron energies between 400 keV and 20 MeV. The experimental results are compared with the Evaluated Nuclear Data Files (ENDF).

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

1.20. CROSS SECTIONS FOR THE $^{14}\text{N}(\text{n},\text{p})$, (n,d) , AND (n,t) REACTIONS FROM 0.5 TO 15 MeV*

G. L. Morgan

(Abstract of *IEEE TM-8389*, Sept. 1978, and of paper to be published in *Nuclear Science and Engineering*.)

Cross sections have been measured for the $^{14}\text{N}(\text{n},\text{p})$ reaction from 0.6 to 7.0 MeV and for the (n,d) and (n,t) reactions from 1 to 15 MeV and 4 to 15 MeV, respectively. The data were obtained using a gaseous scintillator containing N_2 and Xe mixtures. A linac was used as a pulsed, white neutron source with a 29-m flight path. The results of the measurement are compared to the current evaluated file for nitrogen; agreement is good for neutron energies below 8 MeV, but the measurement is substantially higher than the evaluation for neutron energies near 10 MeV.

* Research sponsored by Defense Nuclear Agency.

1.21. CROSS SECTIONS FOR THE ${}^7\text{Li}(n,\gamma)$ AND ${}^7\text{Li}(n,\alpha)$ REACTIONS BETWEEN 1 AND 20 MeV*

G. L. Morgan

(Abstract of ORNL TM-2140, March 1976)

Differential cross sections for the production of secondary neutrons and gamma rays from neutron interactions in ${}^7\text{Li}$ have been measured at 50 and 126 deg (lab) for incident neutron energies in the range 1 to 20 MeV. An electron linac was used as a white neutron source. Incident neutron energies were determined using time-of-flight techniques for a source-to-sample distance of 48 m. Secondary spectra were determined by analysis of the pulse-height distributions observed in a NE-213 scintillation counter. The results are compared to the current evaluated data file (ENDF/B-IV, MAT 1272).

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

1.22. ABSOLUTE MEASUREMENT OF THE ${}^7\text{Li}(n,\gamma){}^7\text{Li}^*$ (0.478-MeV) CROSS SECTION*

D. K. Olsen G. L. Morgan J. W. McConnell

(Abstract of ORNL TM-2140, March 1976)

The ${}^7\text{Li}(n,\gamma){}^7\text{Li}^*$ (0.478-MeV cross section could provide a useful standard for other gamma-ray-production measurements and is of interest to fusion work. Gamma-ray spectra from neutron bombardment of ${}^7\text{Li}$ contain only the 0.478-MeV line whose production cross section has a relatively low threshold and is both isotropic with angle and smooth with energy. Because of these properties and also to verify the normalization of our ${}^7\text{Li}(n,\alpha)$ work, we have measured this ${}^7\text{Li}$ cross section from threshold to 5 MeV employing the ORELA white neutron source, a Ge(Li) photon detector, and a solid-state proton-recoil flux detector.

One thousand 35-nsec FWHM neutron pulses per second were produced from the ORELA water-cooled Ta target. After being filtered through a 0.93-g/cm² ${}^1\text{B}$ filter to inhibit time overlap and 14.0 cm of Th to attenuate gamma flash, the flux was collimated into an 17-cm-diam beam which traversed a proton-recoil flux monitor and scattering sample at 21.430 and 22.181 m, respectively. The scattering sample consisted of a 15.2- x 15.2- x 0.63-cm isotopically enriched ${}^7\text{Li}$ plate enclosed in an airtight stainless-steel container and mounted at 40 deg to the beam direction in a perpendicular plane.

Recoil protons from the radiator of the flux monitor, which consisted of 2.0- and 0.67-mg/cm² polyethylene foils with a 5.9-mg/cm² Al absorber sandwiched between, were counted at a laboratory angle of 15 deg with a 0.30-mm-thick Si solid-state detector mounted with an entrance aperture of 1.27 cm² and located 30.6 cm from the radiator center. Gamma rays were observed with a 95-cm³ true-coaxial Ge(Li) detector positioned 28 cm from the sample at an angle of 125 deg to the beam direction and shielded from low-energy photons with 1.8 mm of Pb.

Fast timing pulses were obtained from the Ge(Li) detector with an ORTEC 473A constant fraction discriminator gated with slow rise time reject and from the proton-recoil detector with an ORNL Q3066a cross-over timing discriminator. From these fast pulses, neutron flight times for both detectors were measured with an EG&G TDC100 time digitizer operated in the single stop mode. Simultaneous slow pulses produced 2048- and 256-channel pulse-height spectra, respectively, for each of 160 timing bins.

Events from an ${}^{241}\text{Am}$ alpha source mounted in the proton-recoil detector provided a pulse-height energy calibration for the Si diode and assured that after the clock deadtime correction there were no counting losses. Using known (n,γ) cross sections, analysis of the two-dimensional proton recoil data provided a flux measurement with an estimated systematic uncertainty of 45%.

The efficiency of the Pb-shielded Ge(Li) detector was measured *ex situ* with standard sources to be $0.270 \times 10^{-3} \pm 2\%$. The counting rate and line shape of the Ge(Li) detector were monitored with 662-keV gamma radiation from a ^{137}Cs source mounted near the ^7Li sample.

After the clock deadtime correction, events from the ^{137}Cs source showed no counting losses in the Ge(Li) detector at long flight times; however, over the energy range of interest, at short flight times peak tailing and a consistent 10% counting loss from pileup and electronic deadtimes were measured. After correcting for this measured loss of Ge(Li) counts, the Li 0.478-MeV gamma-ray-production cross section was calculated.

Corrections for neutron multiple scattering, neutron flux and gamma-ray attenuation, and the small solid-angle variations from the finite sample size were computed using Monte Carlo procedures. At all energies, the multiple scattering and flux attenuation corrections were <11 and 5%, respectively. Gamma-ray self-absorption was 1.3%.

Figure 1.22.1 compares our resulting cross sections with those of ENDF/IV (ref. 1) and the recent measurements of Smith² and Morgan.³ Earlier measurements are discussed in ref. 2. Our plotted errors are statistical standard deviations from both detectors quadratically added to an additional 5% to account for uncorrelated uncertainties in extracting counts from distorted peak shapes and possible uncertainties in the pileup correction. Quadratic addition of an estimated 6% correlated uncertainty in the normalization gives an 8% absolute error for each data point. The cross sections of Morgan³

ORNL-DWG 78-2207

Fig. 1.22.1. Cross sections for the production of the 0.478-keV gamma ray in ^7Li by neutron inelastic scattering.

have small relative errors (not shown) and a 10% absolute error, whereas the data of Smith² have 5 and 9% for relative and absolute errors, respectively. The only serious discrepancy is from threshold to 1.3 MeV where both ORELA cross sections are in excellent agreement with one another and larger than either the Smith² or evaluated³ values. Above 2.0 MeV, our cross sections are on the average 6% smaller than the Morgan³ values. From 1.3 to exactly 2.4 MeV our cross sections agree well with the Smith² data. Above exactly 2.4 MeV they are smaller.

The two ORELA measurements used different beam lines, ⁷Li samples, photon detectors, flux determinations and procedures, and are consistent within their quoted uncertainties; consequently, a suitable average of these results provides a ⁷Li(α , γ) 0.478-keV gamma-ray-production cross section accurate in both shape and magnitude to 8%.

*Research sponsored by DOE Office of Basic Energy Sciences, Division of Nuclear Sciences.

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1.23. A PROTON-RECOIL TELESCOPE FOR IN-LINE ABSOLUTE NEUTRON FLUENCE MEASUREMENTS*

G. L. Morgan D. K. Olsen J. W. McConnell

Abstract of paper to be published in Nuclear Instruments and Methods

A proton-recoil telescope has been designed for absolute neutron fluence monitoring at the Oak Ridge Electron Linear Accelerator. This telescope uses a combination radiator designed to compensate for the rapid decrease in neutron intensity over the energy range from 0.5 to 5.0 MeV. The telescope has greater than 99% transmission and can be used for in-line fluence measurements. Absolute uncertainties vary from 2% to 10% depending on the neutron energy.

*Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

1.24. INSTRUMENTS FOR USE OF ⁶Li AS A STANDARD*

L. W. Weston

Abstract of paper presented at the International Conference on Neutron Standards and Measurements, March 1-4/1, 1978, Bethesda, Maryland, and published in "Neutron Standards and Measurements," NEA Special Publication 442, 48-49, October 1978

A review is given of the instruments which use the ⁶Li(α , n)³H reaction as a standard in the measurements of neutron flux. These instruments consist of scintillation detectors, gaseous ionization detectors, and external particle detectors. The Li glass scintillator is the most versatile because of minimized effects of the angular distribution of the reaction products and simplicity. The gaseous ionization detector has lower gamma-ray sensitivity than does the scintillation detector. Surface-barrier diodes may be used to detect the reaction products when a neutron spectrometer is desired. A choice of instruments to use this reaction must consider gamma-ray sensitivity, effects of the angular distribution of the reaction, and the range of the triton in the detector.

*Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

1.25. STATUS REPORT TO THE DOE NUCLEAR DATA COMMITTEE*

Compiled and edited by:
F. G. Perey J. C. Gentry

(Abstract of ORNL/TN-6888, April 1978)

This report was prepared for the DOE Nuclear Data Committee and covers work performed at ORNL since March 1977 in areas of nuclear data of relevance to the U.S. applied nuclear energy program. The report was mostly generated through a review of abstracts of work completed to the point of being subjected to some form of publication in the open literature, formal ORNL reports, ORNL technical memoranda, progress reports, or presentation at technical conferences. As much as possible we have reproduced the complete abstract of the original publication with only minor editing. In a few cases progress reports were written specifically for this publication. The editors have selected the materials to be included in this report on the basis of perceived interests of DOE Nuclear Data Committee members and cannot claim completeness.

*Research sponsored by DOE Office of Basic Energy Sciences, Division of Nuclear Sciences.

1.26. FISSION-PRODUCT ENERGY RELEASE FOR TIMES FOLLOWING THERMAL-NEUTRON FISSION OF ^{235}U BETWEEN 2 AND 14,000 SECONDS*

J. K. Dickens J. F. Emery T. A. Love J. W. McConnell
K. J. Northcutt R. W. Peelle H. Weaver

(Abstract of paper presented at June 1977 Meeting of the Nuclear Energy Agency Committee on Reactor Physics, Petten, The Netherlands; also abstract of NUREG-14, October 1977)

Fission-product decay energy-release rates have been measured for thermal-neutron fission of ^{235}U . Samples of mass 1 to 10 μg were irradiated for 1 to 100 sec using the fast pneumatic-tube facility at the Oak Ridge Research Reactor. The resulting beta- and gamma-ray emissions were counted for times-after-fission between 2 and 14,000 sec. The data were obtained for beta and gamma rays separately as spectral distributions, $\lambda(E_\beta)$ vs E_β and $\lambda(E_\gamma)$ vs E_γ . For the gamma-ray data the spectra were obtained using an NaI detector, while for the beta-ray data the spectra were obtained using an RE-lit detector with an anticoincidence mantle. The raw data were unfolded to provide spectral distributions of modest resolution. These were integrated over E_β and E_γ to provide total yield and energy integrals as a function of time after fission. Results are low compared to the present 1973 AHS decay-heat standard.

A complete description of the experimental apparatus and data-reduction techniques is presented. The final integral data are given in tabular and graphical form and are compared with published data.

*Research sponsored by Nuclear Regulatory Commission, Division of Nuclear Regulatory Research.

†Analytical Chemistry Division.

1.27. DELAYED BETA- AND GAMMA-RAY PRODUCTION DUE TO THERMAL-NEUTRON FISSION OF ^{235}U , SPECTRAL DISTRIBUTIONS FOR TIMES AFTER FISSION BETWEEN 2 AND 14,000 SEC: TABULAR AND GRAPHICAL DATA*

J. K. Dickens T. A. Love J. W. McConnell J. F. Emery
K. J. Northcutt R. W. Peelle H. Weaver

(Abstract of NUREG-14, October 1977, and NUREG-14, August 1978)

Fission-product decay energy-release rates have been measured for thermal-neutron fission of ^{235}U . Samples of mass 1 to 10 μg were irradiated for 1 to 100 sec using the fast pneumatic-tube facility at the Oak Ridge Research Reactor. The resulting beta- and gamma-ray emissions were counted for times after-fission between 2 and 14,000 sec. The

data were obtained for beta and gamma rays separately as spectral distributions, $\Sigma(E_\beta)$ vs t_d and $\Sigma(E_\gamma)$ vs t_d . For the gamma-ray data the spectra were obtained using a NaI detector, while for the beta-ray data the spectra were obtained using an NE-110 detector with an anticoincidence mantle. The raw data were unfolded to provide spectral distributions of moderate resolution. These distributions are given in graphical and tabular form as differential cross-section values of $d\sigma/dE$ fission for gamma-ray energy intervals ranging from 10 keV for $E_\beta < 0.18$ MeV to 100 keV for $E_\beta > 6.8$ MeV, and beta-ray energy intervals ranging from 20 keV for $E_\beta < 0.25$ MeV to 160 keV for $E_\beta > 6.4$ MeV. Counting-time intervals range from 1 sec for times-after-fission (t_d) < 6 sec to 4000 sec for $t_d = 10^4$ sec. The graphical representations also include calculated spectra using summation methods and the ENDF/B-IV fission yield and decay scheme data base.

* Research sponsored by U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research.

Analytical Chemistry Division.

1.28. FISSION-PRODUCT ENERGY RELEASE FOR TIMES FOLLOWING THERMAL-NEUTRON FISSION OF ^{235}Pu BETWEEN 2 AND 14000 SECONDS*

J. K. Dickens J. F. Emery T. A. Love J. W. McConnell
K. J. Northcutt R. W. Peelle H. Weaver

(Abstract of NUREG-01-84, April 1978)

Fission-product decay energy-release rates have been measured for thermal-neutron fission of ^{235}Pu . Samples of mass 1 to 5 μg were irradiated for 1, 5, or 100 sec using the fast pneumatic-tube facility at the Oak Ridge Research Reactor. The resulting beta- and gamma-ray emissions were counted for times-after-fission between 2 and 14000 sec. The data were obtained for beta and gamma rays separately as spectral distributions, $\Sigma(E_\beta)$ vs t_d and $\Sigma(E_\gamma)$ vs t_d . For the gamma-ray data the spectra were obtained using a NaI detector, while for the beta-ray data the spectra were obtained using an NE-110 detector with an anticoincidence mantle. The raw data were unfolded to provide spectral distributions of modest resolution. These were integrated over E_β and E_γ to provide total yield and energy integrals as a function of time after fission. The final integral data are given in tabular and graphical forms, including a complete analysis of uncertainties.

Beta-ray energy emission rates fall somewhat below summation calculations based on the ENDF/B-IV fission-product file except for times less than 15 sec after fission and times near 2000 sec after fission. Photon energy emission falls above the summation calculation for all measured time integrals, by amounts ranging from 1 to 33%. The data were summed to obtain equivalent decay power for an infinite irradiation normalized to summation calculation at 14000 sec after shutdown. On this basis the summation calculation is equal to or falls below the experimental points by amounts up to 5%.

* Research sponsored by U. S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research.

Analytical Chemistry Division.

1.29. FISSION-PRODUCT ENERGY RELEASE FOR TIMES FOLLOWING THERMAL-NEUTRON FISSION OF ^{239}Pu BETWEEN 2 AND 14000 SECONDS*

J. K. Dickens J. F. Emery T. A. Love J. W. McConnell
K. J. Northcutt R. W. Peelle H. Weaver

(Abstract of NUREG-01-84, April 1978)

Fission-product decay energy-release rates have been measured for thermal-neutron fission of ^{239}Pu . Samples of mass 1 and 5 μg were irradiated for 1 to 50 sec using the fast pneumatic-tube facility at the Oak Ridge Research Reactor. The resulting beta- and gamma-ray emission spectra were recorded for times-after-fission between 2 and 14000 sec. The data were obtained for beta and gamma rays separately as spectral distributions,

$N(E_\gamma)$ vs E_γ and $N(E_\beta)$ vs E_β . For the gamma-ray data the spectra were obtained using an NaI detector, while for the beta-ray data and spectra were obtained using an NE-110 detector with an anticoincidence mantle. The raw data were unfolded to provide spectral distributions of modest resolution. Total yield and energy integrals as a function of time after fission were obtained from integration over E_γ and E_β . The final differential spectral and integral data are given in tabular and graphical form.

The total integral energy-release data are compared with similar data previously obtained for ^{235}U and ^{239}Pu . Following a pulse of fissions, the total decay-heat power for ^{241}Pu fission is greater by 45% than that for ^{235}U for times after a fission pulse <2500 sec, and greater by up to 40% than that for ^{239}Pu . For times after a fission pulse greater than 5000 sec, the decay-heat power for ^{241}Pu asymptotically approaches that for ^{239}Pu and is less than that for ^{235}U by 25% at 14000 sec.

On an infinite irradiation basis, the decay-heat power for ^{241}Pu is equal to that for ^{235}U for times after shutdown less than 100 sec and less than that for ^{235}U for times after shutdown greater than 300 sec. The decay-heat power for ^{241}Pu is greater than that for ^{239}Pu for times after shutdown less than 50 sec, asymptotically approaching and becoming equal to the ^{239}Pu decay-heat power for longer times after fission.

*Research sponsored by U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research.

†Analytical Chemistry Division.

1.30. FISSION YIELDS FOR THERMAL-NEUTRON FISSION OF ^{241}Pu *

J. K. Dickens

(Abstract of paper to be published in *Nuclear Science and Engineering*)

Cumulative fission yields for 17 fission products (16 mass chains) created by thermal-neutron fission of ^{241}Pu were determined from analysis of gross fission-product gamma-ray spectra obtained using a large-volume Ge(Li) detector. Uncertainties assigned to nine of the measured yields are smaller than existing evaluated uncertainties.

*Research sponsored by U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research.

1.31. ORCODE.77: A COMPUTER ROUTINE TO CONTROL A NUCLEAR PHYSICS EXPERIMENT BY A PDP-15 + CAMAC SYSTEM, WRITTEN IN ASSEMBLER LANGUAGE AND INCLUDING MANY NEW ROUTINES OF GENERAL INTEREST*

J. K. Dickens J. W. McConnell

(Abstract of ORNL/NUREG/TM-33, May 1977)

ORCODE.77 is a versatile data-handling computer routine written in MACRO (assembler) language for a PDP-15 computer with EAE (extended arithmetic capability) connected to a CAMAC interface. The Interrupt feature of the computer is utilized. Although the code is oriented for a specific experimental problem, there are many routines of general interest, including a CAMAC scaler handler, an executive routine to interpret and act upon three-character teletype commands, concise routines to type out double-precision integers (both octal and decimal) and floating-point numbers and to read in integers and floating-point numbers, a routine to convert to and from PDP-15 FORTRAN-IV floating-point format, a routine to handle clock interrupts, and our own DECTAPE handling routine. Routines having specific applications which are applicable to other very similar applications include a display routine using CAMAC instructions, control of external mechanical equipment using CAMAC instructions, storage of data from an analog-to-digital converter, analysis of stored data into time-dependent pulse-height spectra, and a routine to read the contents of a Nuclear Data 5050 analyzer and to prepare DECTAPE output of these data for subsequent analysis by a code written in PDP-15-compiled FORTRAN-IV.

*Research sponsored by U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research.

1.32. LSFODF: A GENERALIZED NONLINEAR LEAST-SQUARES FITTING PROGRAM FOR USE WITH ORELA ODF FILES*

D. H. Wharton[‡] D. K. Olsen

(Abstract of ORNL/TM-6546, in press)

The Fortran-10 program LSFODF has been written on the ORELA PDP-10 in order to perform non-linear least-squares curve fitting with user supplied functions and derivatives on data which can be read directly from ORELA-data-format (ODF) files. LSFODF can be used with any user supplied function and derivatives; has its storage requirements specified in this function; has P-search and n-search capabilities; and can output the input data and fitted curve in an ODF file which then can be manipulated and plotted with the existing ORELA library of ODF programs. A description of the fitting formalism, input instructions, five test cases, and a program listing are given.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

[‡]ORAU Summer Student from The University of North Carolina, Asheville.

1.33. CALCULATIONS PERTAINING TO THE DESIGN OF A PREBUNCHER FOR A 150-MeV ELECTRON LINEAR ACCELERATOR: RADIAL MOTION*

R. G. Alsmiller, Jr. F. S. Alsmiller[‡] J. Barish[‡]

(Abstract of paper to be presented at Particle Accelerator Conference, Accelerator Engineering and Technology, March 12-14, 1979, San Francisco, California)

In a previous paper¹ calculated results based on a one-dimensional ballistic model were presented to indicate the extent to which a current pulse of 150-keV electrons containing 1 μ C of charge and having a duration of 15 msec (FWHM) could be bunched by a combination of accelerating and decelerating voltage gaps followed by a drift space. In the present paper a very approximate model is used to determine the extent to which the prebuncher performance determined previously is modified by radial electron motion. It is found that if a longitudinal magnetic field of 1 kG is applied in the region containing the voltage gaps and 3 kG is applied in the drift space, the prebuncher performance is essentially unaffected by radial electron motion.

* Research sponsored by DOE Office of Basic Energy Sciences, Division of Nuclear Sciences.

[‡]Research participant.

[‡]Computer Science Division.

1. R. G. Alsmiller, Jr. et al., "Calculations Pertaining to the Design of a Prebuncher for an Electron Linear Accelerator," ORNL/TM-5419 (1977), to be published in *Particle Accelerators*.

2. CROSS-SECTION THEORY, EVALUATIONS, AND EVALUATION TECHNIQUES

2.0. INTRODUCTION

The evaluation effort in the Division started in 1969, with a commitment to provide evaluations on calcium, iron and lead for inclusion in the Evaluated Nuclear Data File (ENDF/B - Version III). Since then our group has achieved national prestige in emphasizing development and usage of physics models in evaluation work. In addition to routine use of R-matrix analyses, global optical model searches, and DWBA calculations in our evaluations, we have maintained leadership in the development and application of multistep Hauser-Feshbach/precompound model codes with gamma-ray production capability. This year, the cross-section evaluation effort culminated in the release of ENDF/B - Version V. We provided full evaluations of C, F, Na, Mg, Si, Ca, Fe, Cu, Pb, ^{239}Pu and ^{241}Pu to this national effort, as well as partial evaluations for K, Cl, ^{232}Th , ^{233}U , ^{235}U and ^{238}U .

The evaluation capability in the Division complements both the measurement program at ORELA and the methods development and application program. Our state-of-the-art evaluation program interacts with and influences the measurement program at ORELA by fully exploiting measured results and recommending measurements required to supplement missing or inadequate data, as well as measurements most useful to the design of nuclear model codes. The methods development and application program benefits from our in-depth knowledge of cross sections. Frequent requests are made for results which are either not contained in the current evaluated files, or are inadequate to meet a goal. Such cross sections have also been requested and used by other divisions within the Laboratory, for example the Solid State and Fusion Energy Divisions.

Responding to cross-section needs for the Magnetic Fusion Energy Program, we have recently improved our multistep Hauser-Feshbach code to conserve angular momentum in the precompound mode of reaction to add its versatility and validity up to 50 MeV. Ongoing endeavors to further improve this powerful code include unifying the compound and precompound reactions and incorporating anisotropic angular distributions for outgoing particles emitted from the precompound mode. Completion of this effort will keep us in the forefront of model development and application to evaluation needs.

Another new development for ENDF/B-V is the inclusion for the first time of extensive cross-section uncertainty files, including correlations among cross sections at various energies and for different reactions and materials. We have had a significant role in this area, a role which has ranged from inventing formats for the files to demonstrating the importance of uncertainty information. Uncertainty files are useful to an experimenter or program planner to set priorities for potential new experiments by judging their impact relative to existing data. They are used in applied work to determine the uncertainty in a computed system performance parameter due to uncertainties in the various input cross sections. Most of the evaluations we contributed to ENDF/B-V contain uncertainty files. Current areas of activity regarding cross-section uncertainties include developing and documenting techniques which can be used for proper evaluation of uncertainty files, and developing methods for properly combining data and extracting uncertainties when there exist a variety of measurements for a given cross section.

The "dosimetry" effort, to characterize through local reaction-rate measurements the neutron flux spectrum producing radiation damage, has long been recognized to be sensitive to the uncertainties in the cross sections of the materials employed in the dosimetry packages. Unfortunately, the analysis techniques were not well developed to handle these uncertainties and the required uncertainty files were generally not available. As a result of communication with dosimetry experts during the development of the uncertainty files, it became apparent to us that some "least-squares" analysis techniques already in limited use for cross-section adjustment should be applicable to the dosimetry problem. We have helped to develop these techniques, and full application should eventually lead to optimal dosimetry analysis techniques when initial uncertainty files for all the problem input quantities have become available. At the same time, dosimetry benchmark results will become more helpful to evaluation of the dosimetry reaction cross sections.

As a result of all these efforts, we are pleased to note that cross-section evaluation, often considered to be an art, is now well on the way to being a science.

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2.1. R-MATRIX EVALUATION OF CARBON AS A NEUTRON SCATTERING STANDARD BELOW 2 MeV*

C. Y. Fu F. G. Perey

(Abstract of paper to be published in Atom. Data and Nucl. Data Tables)

Cross-section and polarization data for neutron interaction with carbon below the inelastic threshold (4.81 MeV) were evaluated and analyzed by R-matrix theory. Six sets of total cross-section data below 2 MeV were partially smoothed and least-square fitted with nondiagonal weighting to obtain averages and their estimated uncertainties for use in the R-matrix search program. The resulting differential scattering cross sections below 2 MeV are recommended as standards. The good quality of the fits well above 2 MeV and a high-precision (0.1%) thermal value enhance the confidence in the recommended values below 2 MeV.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Nuclear Power Development.

2.2. MULTISTEP HAUSER-FESHBACH CODES WITH PRECOMPOUND EFFECTS: A BRIEF REVIEW OF CURRENT AND REQUIRED DEVELOPMENTS AND APPLICATIONS UP TO 40 MeV*

C. Y. Fu

(Abstract of paper presented at Symposium on Neutron Cross Sections from 10 to 40 MeV, Brookhaven National Laboratory, May 3-6, 1977; BNL-NCS-60651, 453-466, 1977)

Mainly for applications to fusion energy research, several multistep Hauser-Feshbach codes with precompound effects have been developed, and an effort to intercompare the codes is in progress. Extensive reviews on the relevant theories have recently become available, and in this paper a few points of interest with respect to the developments and applications of the codes are discussed.

* Research sponsored by DOE Office of Basic Energy Sciences, Division of Nuclear Sciences.

2.3. PERTURBATION FORMALISM FOR THE COMPLEX POLES AND WIDTHS OF THE TRANSITION MATRIX WITH AN APPLICATION TO INTERMEDIATE STRUCTURE PHENOMENA*

R. B. Perez C. de Saussure D. K. Olsen F. C. Difilippo[†]

[Abstract of Phys. Rev. D 17(8), 964-977, 1978]

The complex poles and widths of the transition T matrix are determined by the trajectory equations which consist of a set of first-order nonlinear differential equations. A hierarchy of approximate solutions to the trajectory equations is developed by iterative methods. The results of this formalism are compared with exact solutions for the case of some strongly interacting pairs of resonances in two iron isotopes. In the presence of intermediate structure the average neutron reaction cross section is interpreted in terms of a resonant strength function which exhibits peaks at neutron energies corresponding to "doorways" levels.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

[†] An IAEA fellow now with Comisión Nacional de Energía Atómica, Argentina.

2.4. ON THE EXPANSION OF THE PROPAGATOR IN POWER SERIES OF THE COUPLING CONSTANT*

R. B. Perez

(Abstract of paper to be published in Journal of Mathematical Physics)

We show by simple rearrangements of the Lippmann-Schwinger equation that the Propagator (Green's Function) of a linear physical field can be expanded in power series valid for values of the coupling constant either close to unity or very large. The new series satisfy the Generalized Resolvent Operator Equation first derived by Mockel.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

2.5. THE EFFECT OF THE ENERGY DEPENDENCE OF THE NEUTRON WIDTHS ON THE CALCULATION OF AVERAGE REACTION CROSS SECTIONS*

F. C. Difilippo[†] R. B. Perez

(Abstract of ORNL/TM-5550, December 1977)

We study the validity of neglecting the energy dependence of the neutron width in the calculation of average reaction cross sections.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

[†]An IAEA fellow now with Comisión Nacional de Energía Atómica, Argentina.

2.6. EVALUATION OF THE ²³⁵U NEUTRON CROSS SECTIONS FOR INCIDENT NEUTRON ENERGIES UP TO 4 keV*

G. de Saussure D. K. Olsen R. B. Perez F. C. Difilippo[†]

(Abstract of ORNL/TM-0152, January 1978; also abstract of paper to be published in Progress in Nuclear Energy)

This report describes an evaluation of the ²³⁵U cross sections below 4 keV. Recent measurements and reanalyses of older data are discussed. Evaluated resonance parameters are obtained for 164 s-wave and 280 p-wave levels. The capture widths of the first three s-wave levels are significantly lower than in the ENDF/B-IV evaluation. The s-wave strength function above 1.5 keV is systematically larger than in ENDF/B-IV. Statistical and systematic uncertainties are evaluated for the resonance parameters and for the smooth backgrounds. The statistical distributions of the resonance parameters are compared with theoretically expected distributions.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

[†]An IAEA fellow now with Comisión Nacional de Energía Atómica, Argentina.

2.7. EVALUATION OF THE ^{232}Th NEUTRON CAPTURE CROSS SECTION ABOVE 3 keV*G. de Saussure R. L. Macklin[†]

(Abstract of SAND 80-0801, December 1980)

This memo describes an evaluation of the ^{232}Th neutron capture cross section in the neutron energy range from 3 keV to 20 MeV. Most existing differential measurements are reviewed, and some data are renormalized to current values of the standards. Several experimentally determined sets of average resonance parameters are also discussed.

From 3 to 50 keV the evaluated cross section is described by a set of average statistical resonance parameters. Above 50 keV the evaluated capture cross section is a smooth curve which follows the trend of the most recent measurements. The evaluated capture cross section is compared with many measurements and uncertainty estimates are given.

*Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

[†]Physics Division.

2.8. EVALUATION OF $^{232}\text{Th}(n,n')$, $(n,2n)$ AND $(n,3n)$ CROSS SECTIONS*

C. Y. Fu

(Abstract of SAND 80-0810, May 1981)

A "minor" evaluation effort consistent with time limitations has been performed to update the (n,n') , $(n,2n)$, and $(n,3n)$ cross sections for ^{232}Th . The results should represent a substantial improvement over ENDF/B-IV MAT 1296, which was based on data taken ten years ago. This report describes the techniques used in the evaluation, presents tabulations of the recommended cross sections, and graphically compares the cross sections with the most recent available data and with ENDF/B-IV values.

*Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

2.9. REVIEW AND COMBINATION OF EXPERIMENTAL RESULTS FOR NEUTRON EMISSION PER FISSION OF ^{232}Th *

R. Swin

(Abstract of SAND 81-0846, May 1981)

An analysis has been made of the existing experimental data for $\bar{\nu}(E)$ of ^{232}Th , the average number of neutrons emitted per fission of ^{232}Th . This analysis suggests that a single linear function of the neutron energy describes the data within their uncertainties. The parameters of the linear function and their uncertainties are given.

*Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

2.10. LEAST-SQUARES DOSIMETRY UNFOLDING: THE PROGRAM STAY'SL*

F. G. Perey

(Abstract of SAND 81-0876, SAND-1.4, October 1981)

A PDP-10 FORTRAN IV computer program STAY'SL, which solves the dosimetry unfolding problem by the method of least squares, is described. The solution (the output spectrum

and its covariance matrix) is calculated by minimizing chi-square based on the input data (the activation data, the input spectrum, the dosimetry cross sections and their uncertainties given by covariance matrices). The solution reflects therefore the uncertainties in all of the input data and their correlations. The correlations among the various dosimetry cross sections are taken into account; however, the activation data, input spectrum and cross sections as classes are assumed to be uncorrelated with each other. The code and sample problem are available from the Radiation Shielding Information Center (RSIC) at Oak Ridge National Laboratory.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

2.11. UNCERTAINTY ANALYSIS OF DOSIMETRY SPECTRUM UNFOLDING*

F. G. Perey

Abstract of paper presented at Second ANS-EPRI Meeting, April 29-May 1, 1976, in Atlanta, Georgia, Paper E-1, 1976, and published in ANS-EPRI, Vol. 19, 1976, 117; also abstract of one of the papers published in ANS-EPRI, "Contributions to the Reactor Spectrum Unfolding," February 1976.

The propagation of uncertainties in the input data is analyzed for the usual dosimetry unfolding solution. A new formulation of the dosimetry unfolding problem is proposed in which the most likely value of the spectrum is obtained. The relationship of this solution to the usual one is discussed.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

2.12. SPECTRUM UNFOLDING BY THE LEAST-SQUARES METHOD*

F. G. Perey

Abstract of paper presented at ANS-EPRI Meeting, April 29-May 1, 1976, in Atlanta, Georgia, Paper E-1, 1976, and published in ANS-EPRI, Vol. 19, 1976, 117; also abstract of one of the papers published in ANS-EPRI, "Contributions to the Reactor Spectrum Unfolding," February 1976.

The method of least squares is briefly reviewed, and the conditions under which it may be used are stated. From this analysis, a least-squares approach to the solution of the dosimetry neutron spectrum unfolding problem is introduced. The mathematical solution to this least-squares problem is derived from the general solution. The existence of this solution is analyzed in some detail. A χ^2 -test is derived for the consistency of the input data, which does not require the solution to be first obtained. The fact that the problem is technically nonlinear, but should be treated in general as a linear one, is argued. Therefore, the solution should not be obtained by iteration. Two interpretations are made for the solution of the code STAY'SL, which solves this least-squares problem. The relationship of the solution to this least-squares problem to those obtained currently by other methods of solving the dosimetry neutron spectrum unfolding problem is extensively discussed. It is shown that the least-squares method does not require more input information than would be needed by our current methods in order to estimate the uncertainties in their solutions. From this discussion it is concluded that the proposed least-squares method does provide the best complete solution, with uncertainties, to the problem as it is understood now. Finally, some implications of this method are mentioned regarding future work required in order to exploit fully its potential.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

2.13. THE DATA COVARIANCE FILES FOR ENDF/B-V*

F. G. Perey

(Abstract of ORNL/TM-5938, ENDF-249, July 1977)

In this report we present the formats and procedures for the data covariances of ENDF/B-V.

*Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

2.14. EXPECTATIONS FOR ENDF/B-V COVARIANCE FILES: COVERAGE, STRENGTH AND LIMITATIONS*

F. G. Perey

(Abstract of paper presented at RSIC Seminar-Workshop on Theory and Application of Sensitivity and Uncertainties Analysis, Oak Ridge, Tennessee, August 22-24, 1978; proceedings to be published in ORNL/RSIC-42)

The concept of the estimated data covariance files of ENDF/B is explored in detail. The relationship between uncertainties and "possible mistakes" is discussed in order to avoid likely misunderstanding concerning the use of the data covariance files. Some perceived strengths and weaknesses of the ENDF/B-V data covariance files are mentioned.

*Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

2.15. COVARIANCE MATRICES OF EXPERIMENTAL DATA*

F. G. Perey

(Abstract of paper presented at the International Conference on Neutron Physics Data for Reactors and Other Applied Purposes, Harwell, United Kingdom, September 15-18, 1978)

A complete statement of the uncertainties in data is given by its covariance matrix. It is shown how the covariance matrix of data can be generated using the information available to obtain their standard deviations. Determination of resonance energies by the time-of-flight method is used as an example. The procedure for combining data when the covariance matrix is nondiagonal is given. The method is illustrated by means of examples taken from the recent literature to obtain an estimate of the energy of the first resonance in carbon and for five resonances of ^{235}U .

*Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

3. CROSS-SECTION PROCESSING, TESTING, AND SENSITIVITY ANALYSES

3.0. INTRODUCTION

During 1978, the AMPX-II neutron-gamma multigroup cross-section processing system was packaged and released for distribution through the Radiation Shielding Information Center. This version of AMPX, which was featured in an RSIC Seminar-Workshop on Multi-group Nuclear Cross-Section Processing, represents a major update of the previously distributed version and offers a wide variety of cross-section production and handling capabilities not available in the earlier version. Two other codes (the MINX-SPHINX and the NUOY neutron cross-section processing codes) have been linked to the AMPX gamma-ray module, and as a result we now have a capability for independent verification of multi-group cross sections. Work is continuing in areas to further improve the systems. In particular, a fundamental review of the treatment of the unresolved energy region has been completed and a unique approach has been developed. Work still in progress is directed toward developing a more accurate technique for handling elastic removal, for the two-dimensional space-energy collapse, and for proper processing of the transport cross section.

Because of our need for accurate data processing and large-core analyses, we have during the past year participated in a DOE program to compare benchmark solutions of fast breeder reactor problems with the same data base but with different processing codes and with different transport codes using the same processed data. The intent was to determine the state of precision of our calculational tools. This work has led to an increased understanding of the effects of various methods approximations (which will be useful, as well, in our data adjustment studies).

Several cross-section libraries have been successfully processed, tested and packaged for distribution through RSIC. A 218-neutron-group cross-section library for criticality safety research was released as DLC-43/CSRL. In addition, the coupled master cross-section library consisting of 171 neutron groups and 36 gamma-ray groups and identified as VITAMIN-C was released as DLC-41, together with a 126-36 subset (DLC-42/CLEAR). The VITAMIN-C library was a cooperative development sponsored by the DOE Office of Fusion Energy and Division of Reactor Research and Technology, and the CLEAR library is the subset tailored for fast reactor applications only.

Efforts are under way to compare the applicability of various cross-section libraries to specific types of transport calculations. The comparisons are made relative to VITAMIN-C, and one goal is to develop a broad-group library applicable for LWR shielding applications. DLC-47/BUGLE, a cross-section library consisting of 45 neutron groups and 16 gamma-ray groups, will be revised as a result of this work, and both VITAMIN-C and the revised broad-group set have been nominated as possible standards for LWR shielding problems.

Since April, 1977, the chairmanship of the Data Testing Subcommittee of CSEWG (Cross Section Evaluation Working Group) has resided in our Division. This subcommittee has the responsibility for the quality of the ENDF/B cross sections, and as a result our emphasis in data testing has markedly increased. We are currently testing data for thermal and fast reactor and shielding applications, for actinide production, and for dosimetry applications. We have had direct input into ENDF/B-V, which is due for release in early 1979. This will be the first time that an ENDF/B general release will have been made after the results of systematic data testing were considered, and, as a result, we feel it will be a better file. The sensitivity coefficients we have developed were used to alert CSEWG to the expected results of proposed evaluation changes.

In connection with our data testing, we will be significantly upgrading our uncertainty analysis program because of the recent availability of covariance files for all major materials (included in ENDF/B-V). As part of our CSEWG leadership role, integral experiment covariance files and quantitative estimates of methods approximations have been developed as part of the benchmark specifications. This information is essential to any credible uncertainty analysis.

The ENDF sensitivity and uncertainty analysis modular code system has been successfully automated through the use of a driver control module which links the other modules. This system has been released for distribution through RSIC. In addition, an

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RSIC Seminar-Workshop was conducted in which the FORSS system was presented and discussed. Documentation (input instructions, etc.) of the FORSS system is also available through RSIC. New features of the system, in addition to driver control, include (1) an expanded adjustment code (UNCOVER) which treats large arrays of covariances, (2) an increasing interaction with multidimensional diffusion and transport codes (VENTURE and DOT-IV), and (3) an extensively improved FIDO input system.

Reactor physics applications of the sensitivity and uncertainty analysis system have included an adjustment of large multigroup cross-section sets based on eight benchmark experiments and extensive sensitivity and uncertainty files, including, in particular, ^{235}U inelastic scattering and fission spectrum contributions. The sensitivity and uncertainty analysis has been applied to two-dimensional models of large LMFBRs of several designs (conventional Pu and ^{20}Pu denatured ^{235}U), as well as to benchmarks.

An approach to methods uncertainties has been developed and preliminary estimates of such effects have been included in data adjustment. Sensitivity methods for time-dependent burnup and depletion have also been developed. One example of the latter is a modification made to the ORIGEN actinide code to calculate nuclide adjoint functions. Forward and adjoint nuclide functions can be combined to compute sensitivity coefficients for functionals of the nuclide field, such as an irradiated sample composition.

Theoretical work has also been completed for sensitivity theory in depletion problems in which the nuclide and neutron fields are coupled nonlinearly, as for power reactor calculations. It has been shown that three adjoint functions (for the nuclide field, the flux shape and the flux normalization) are required in this methodology. Work has begun on writing a computer module in the VENTURE reactor code system to calculate the three adjoint functions.

A byproduct of the ORNL sensitivity work has been the development of an extensive library of multigroup sensitivity profiles for fast reactor, thermal reactor, and shielding problems, with emphasis on the CSEWG benchmarks. The data library is available from RSIC as DLC-45/SENPRO. A multigroup covariance matrix library, DLC-44/COVERX, has also been prepared and is distributed via RSIC. Both data libraries can be applied with various modules of the FORSS system.

FORSS sensitivity techniques are also being expanded and adapted to other (non-nuclear) problems. Significant progress has been made in adjoint sensitivity theory, and a general theory for reactor thermal-hydraulics problems has been derived and tested. The success of the method led to extensive applications in fast and thermal reactor safety analysis programs. In particular, the fast reactor safety code MELT-III developed by Hanford Engineering Development Laboratory has been adapted to sensitivity needs by incorporation of routines for solving adjoint thermal-hydraulics equations. A similar effort is under way to develop light-water reactor sensitivity methods for use with the TRAC or RETRAN codes. Much important development work still remains to be done and a major research effort for coupled transient thermal-hydraulics and transient neutronics is being pursued. This effort involves both theoretical and applications-oriented studies with emphasis on numerical methods problems and on understanding the physical basis for adjoint (importance) function behavior.

3.1. A REVIEW OF MULTIGROUP NUCLEAR CROSS-SECTION PROCESSING: PROCEEDINGS OF A SEMINAR-WORKSHOP, OAK RIDGE, TENNESSEE, MARCH 14-16, 1978*

Compiled by
D. K. Trubey H. R. Hendrickson

(Abstract of ORNL/RSIC-41, October 1978)

These proceedings consist of 18 papers given at a seminar-workshop on "Multigroup Nuclear Cross-Section Processing" held at Oak Ridge, Tennessee, March 14-16, 1978. The papers describe various computer code systems and computing algorithms for producing multigroup neutron and gamma-r cross sections from evaluated data and experience with several reference data libraries.

* Research sponsored jointly by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology; by Defense Nuclear Agency; and by Nuclear Regulatory Commission.

3.2. CROSS-SECTION PROBABILITY TABLES IN MULTI-GROUP TRANSPORT CALCULATIONS*

Dermott E. Cullen† Ernest F. Plechaty† Richard J. Doyas†
Charles R. Weisbin John E. White†

(Abstract of paper presented at RSIC Seminar-Workshop on Multigroup Nuclear Cross-Section Processing, Oak Ridge, Tennessee, March 14-16, 1978; Oak Ridge National Laboratory Report ORNL/RSIC-41, 165-179, 1978)

The use of cross-section probability tables in multigroup transport calculations is presented. Emphasis is placed on how probability table parameters are generated in a multigroup cross-section processor and how existing transport codes must be modified to use them. In order to illustrate the accuracy obtained using probability tables, results are presented for a variety of neutron and photon transport problems.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

† Lawrence Livermore Laboratory.

‡ Computer Sciences Division.

3.3. EXPERIENCE IN DEVELOPING AND USING THE VITAMIN-C 171-NEUTRON, 36-GAMMA-RAY GROUP CROSS-SECTION LIBRARY*

R. W. Roussin C. R. Weisbin J. E. White† R. Q. Wright†
N. M. Greene† W. E. Ford III† J. B. Wright B. R. Diggs†

(Abstract of paper presented at RSIC Seminar-Workshop on Multigroup Nuclear Cross-Section Processing, Oak Ridge, Tennessee, March 14-16, 1978; Oak Ridge National Laboratory Report ORNL/RSIC-41, 107-119, 1978)

The Department of Energy (DOE) Divisions of Magnetic Fusion Energy (DMFE) and Reactor Research and Technology (DRRT) jointly sponsored the development of a coupled, fine-group cross-section library. The 171-neutron, 36-gamma-ray group library is intended to be applicable to fusion reactor neutronics and LMFBR core and shield analysis. Versions of the library are available from the Radiation Shielding Information Center (RSIC) at Oak Ridge National Laboratory in both AMPX and CCCC formats. Computer codes for energy group collapsing, interpolation on Bondarenko factors for resonance self-shielding and temperature corrections, and various other useful data manipulations are available.

The experience gained in the utilization of this library is discussed. Indications are that this venture, which is designed to allow users to derive problem-dependent cross sections from a fine-group master library, has been a success.

* Research sponsored jointly by the Defense Nuclear Agency, DOE Division of Magnetic Fusion Energy and DOE Division of Reactor Research and Technology.

‡ Computer Sciences Division.

3.4. THE SHIELDING FACTOR METHOD FOR PRODUCING EFFECTIVE CROSS SECTIONS: MINX/SPHINX AND THE CCCC INTERFACE SYSTEM*

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(Abstract of paper presented at RSIC Seminar-Workshop on Multigroup Nuclear Cross-Section Processing, Oak Ridge, Tennessee, March 14-16, 1978; Oak Ridge National Laboratory Report ORNL-5015-41, 32-36, 1978)

The Shielding Factor Method is an economical designer-oriented method for producing the coarse-group space and energy self-shielded cross sections needed for reactor-core analysis. Extensive experience with the ETOX/IDX and ENDRUN/TDOWN systems has made the SFM the method of choice for most US fast-reactor design activities. The MINX/SPHINX system was designed to expand upon the capabilities of the older SFM codes and to incorporate the new standard interfaces for fast reactor cross sections specified by the Committee for Computer Code Coordination. MINX is the cross-section processor. It generates multigroup cross sections, shielding factors, and group-to-group transfer matrices from ENDF/B-IV and writes them out as CCCC ISOTXS and BRKOXS files. It features detailed pointwise resonance reconstruction, accurate Doppler broadening, and an efficient treatment of anisotropic scattering. SPHINX is the space-and-energy shielding code. It uses specific mixture and geometry information together with equivalence principles to construct shielded macroscopic multigroup cross sections in as many as 240 groups. It then makes a flux calculation by diffusion or transport methods and collapses to an appropriate set of cell-averaged coarse-group effective cross sections. The integration of MINX and SPHINX with the CCCC interface system provides an efficient, accurate, and convenient system for producing effective cross sections for use in fast-reactor problems. The system has also proved useful in shielding and CTR applications.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

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3.5. CALCULATIONS OF SEVERAL CSEWG SHIELDING BENCHMARKS WITH PRELIMINARY ENDF/B-V DATA*

R. E. Maerker

(Reprint of Trans. Am. Nucl. Soc. 28, 737-738, 1978)

Calculations using preliminary ENDF/B-V data have been compared with the results of four shielding benchmark experiments performed at the Tower Shielding Facility and adopted by CSEWG to "Phase II" to test new evaluations.

The first benchmark experiment tests the neutron total cross section in the range 1 to 10 MeV, averaged over energy intervals on the order of 0.5 MeV. The results of these experiments, which are simple transmission experiments in good geometry that measure only uncollided fluxes behind relatively thick samples of material, exist for iron, oxygen, nitrogen, sodium, and a Type 310 stainless steel. The uncollided fluxes through each sample are calculated on an energy mesh fine enough to include all structure in the total cross section for energies above 0.5 MeV (for example, there are over 4000 energy points for iron), and they are then folded with the resolution function of the NE-213 spectrometer used in the experiment. The samples were long and narrow in order to reduce the effects of scattering into the detector to negligible proportions; hence, the name "broomstick" has been applied to describe these experiments, because of the shape of the samples. Comparisons using both Versions IV and preliminary V, when they are different, with the results of the broomstick experiments will both be presented.

The second benchmark experiment tests the gamma-ray production cross sections arising from neutron radiative capture at 0.0253 eV. The results of these experiments⁶ are available for iron, a Type 321 stainless steel, nitrogen, sodium, aluminum, copper, titanium, calcium, potassium, chlorine, silicon, nickel, zinc, barium, and sulfur in a reduced form that gives the capture spectrum in units of millibarns per capture in 0.5-MeV gamma-ray intervals starting at 1 MeV. No transport calculations are necessary to compare

preliminary ENDF/B-V data with these reduced data. Only those evaluations in preliminary ENDF/B-V that include gamma-ray production cross sections (i.e., the so-called "shielding" materials) can be tested with this benchmark, of course. Thus the results for zinc cannot be tested.

The third benchmark experiment tests the gamma-ray production cross sections arising from fast-neutron interactions ($E_n > 1$ MeV) averaged over a fixed neutron spectrum.⁷ The materials investigated were iron, a Type 321 stainless steel, oxygen, sodium, aluminum, copper, titanium, calcium, potassium, chlorine, silicon, nickel, zinc, barium, and sulfur, with all but zinc available in preliminary ENDF/B-V to be tested. The experimental results are reduced to units of millibarns per interaction in 0.5-MeV gamma-ray intervals starting in the vicinity of 1 MeV, and represent values of $4\pi \frac{d\sigma(E_\gamma, 90^\circ)}{d\Omega}$. Again, no transport calculations are necessary to test preliminary ENDF/B-V.

The last benchmark experiment to be used is actually only a benchmark candidate, since the results are not yet universally available. However, an extensive earlier analysis of this two-dimensional (2-D) experiment included the obtaining of sensitivity profiles, 1-D scaling factors and fine-group results, as well as 2-D discrete ordinate calculations of the measurements using ENDF/B-IV (ref. 8). The experiment consisted of integral neutron flux measurements behind a series of shields that mock up possible upper axial shield configurations in an LMFBR. The general configuration investigated was

X cm stainless steel 304 + Y cm sodium + Z cm carbon steel ,

where X was either 31 or 47, Y was 155, 309, or 460, and Z varied between 0 and 90. In all, 22 combinations of X, Y, and Z were analyzed over a range of attenuations up to 10^{12} .

This experiment can be calculated in two ways using preliminary ENDF/B-V, neither of which involves expensive 2-D discrete ordinate calculations.

The first method involves folding the sensitivity profiles obtained earlier for iron and sodium with the relative changes in the cross sections between preliminary Version V and Version IV, where the sensitivities and relative cross-section changes are both expressed in a fine-group structure. The resulting preliminary Version V corrections to the Version IV calculations are only approximate because of the limits imposed by first-order perturbation theory and the consideration of only the most important interactions (i.e., total, elastic, and inelastic) and materials (i.e., changes in the nickel and chromium cross sections in the stainless steel are ignored).

The second method involves processing the preliminary Version V data into a fine-group structure and scaling the results of a 1-D calculation with the scaling factors obtained earlier. This procedure should adequately predict the result of using preliminary Version V data in a full-blown 2-D calculation.

*Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

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3.6. PROJECTIONS OF ENDF/B VERSION V PERFORMANCE FOR FAST AND THERMAL REACTORS USING SENSITIVITY COEFFICIENTS*

C. R. Weisbin J. H. Marable J. Hardy, Jr.† R. D. McKnight‡

(Abstract of paper presented at American Nuclear Society Topical Meeting on Advances in Reactor Physics, Gatlinburg, Tennessee, April 10-12, 1978; and published in CONF-7801, 73-82, 1978)

As part of our participation in the Cross Section Evaluation Working Group, the authors have generated, compiled, and documented an extensive set of sensitivity coefficients based upon ENDF/B-IV for use in conjunction with the data testing of the ENDF/B-V nuclear data file. The effort also included the application and testing of a computer retrievable sensitivity format (SENPRO) through the use of computer programs for editing, mode changing, and folding with projected changes to the nuclear data file.

Proposed reductions to $^{235}\text{U}(\nu)$ and $^{235}\text{U}(n,f)$ in the fast energy range have significant impact for uranium-fueled fast critical assemblies. The long-standing LMFBR ^{235}U calculated overprediction is not resolved by proposed Version 5 cross-section modifications for $^{238}\text{U}(n,\gamma)$ and $^{239}\text{Pu}(n,f)$. The upward evaluation for the $^{239}\text{Pu}(n,f)/^{235}\text{U}(n,f)$ ratio improves criticality predictions for plutonium-fueled fast assemblies. For thermal reactors, changes to the ^{235}U resonance parameters significantly reduce the long-standing ^{235}U discrepancy. Reduced resonance capture in the 1 eV ^{240}Pu resonance has significant implications for LWR fuel cycle studies.

* Research sponsored by Electric Power Research Institute and DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

† Bettis Atomic Power Laboratories.

‡ Argonne National Laboratory.

3.7. SENSITIVITY COEFFICIENT COMPILATION FOR CSEWG DATA TESTING BENCHMARKS*

C. R. Weisbin J. H. Marable J. Hardy, Jr.† R. D. McKnight‡

(Abstract of Brookhaven National Laboratory Report BNL-ND-24668, ENDF-266, August 1978)

This report compiles and documents sensitivity coefficients generated by member laboratories of the Cross Section Evaluation Working Group (CSEWG) in the course of ENDF/B data testing. The work will be continuously updated as new benchmarks and new types of data are developed. The scope of this effort is sensitivities to many cross-section data types for fast reactors, thermal reactors, shielding, etc. Detailed tabulations of the energy-integrated sensitivity coefficient (i.e., the change in response due to a uniform one percent change in cross section over the entire energy domain), sensitivities to changes in individual resonance parameters, and illustrative examples of energy-dependent sensitivity profiles are given. The entire compilation of energy-dependent coefficients is available in a computer-retrievable format along with programs for editing, mode changing, folding with projected changes to the reference data file, and performance parameter uncertainty estimation.

* Research sponsored by Electric Power Research Institute and DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

† Bettis Atomic Power Laboratories.

‡ Argonne National Laboratory.

3.8. COMPILATION OF MULTIGROUP CROSS-SECTION COVARIANCE MATRICES FOR SEVERAL IMPORTANT REACTOR MATERIALS*

J. D. Drischler C. R. Weisbin

(Abstract of ORNL-5319, ENDF-335, October 1977)

This report presents multigroup cross-section covariance matrices for fission in ^{235}U , ^{238}U , ^{239}Pu , and ^{241}Pu ; capture in ^{235}U , ^{238}U , ^{239}Pu , ^{240}Pu , and ^{241}Pu ; fission neutron yield ($\bar{\nu}$) for ^{235}U , ^{238}U , ^{239}Pu , and ^{240}Pu ; elastic scattering for Na and Fe; non-elastic reactions for Na and Fe; first-level inelastic scattering for ^{238}U ; and all reactions provided in the ENDF/B-IV covariance description of N, O, and C. Other data files generated are included for reference but have not yet been tested. The report presents the multigroup data in six, ten, and fifteen energy group forms corresponding to weighting of the covariance data with fission (GODIVA), LMFBR (ZPR-6/7) and 1/E spectra, respectively. The data are illustrated and tabulated in an edited form for convenience. The multigroup covariance files are currently available from RSIC and NNCS in the COVERX format, a computer retrievable format designed for data uncertainty analysis and described herein. As new covariance files become available with subsequent issues of ENDF/B, this report will be added to or revised by the issuance of dated pages that can be incorporated in the report.

*Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

3.9. RECENT DEVELOPMENTS IN SENSITIVITY ANALYSIS IN THE U.S.*

E. M. Oblo

(Abstract of paper presented at the June 1977 Meeting of the Nuclear Energy Agency Committee on Reactor Physics, Petten, The Netherlands)

Over the past two years there have been several new developments in the U.S. in the field of sensitivity analysis. In general, these advances can be broken down into four categories:

1. Developments in sensitivity theory for new systems of equations.
2. Applications of neutronics sensitivity methodology to more complicated problems.
3. Development of higher order accuracy sensitivity theories.
4. Renewed interest in cross-section adjustment methodology.

A great deal of work is being done on problems in the first category. The reactor core physics and shielding sensitivity theory is being extended to time-dependent, coupled decay (burnup) neutronics problems and also to fuel management and actinide waste disposal problems. A sensitivity theory has also been worked out for the nonlinear drift-kinetic equations of plasma physics, and progress has been made on a general nonlinear sensitivity theory for algebraic econometric modeling problems. In addition, the recent development of channel theory makes it possible to visualize the transport (in energy, space, and angle) of the most important components of the radiation field in a specific problem. Application of channel theory in complicated shield design problems will be a major step toward a quantitative approach to shield optimization.

In the second category are sensitivity studies to determine nuclear data needs for fusion reactor systems and the development of a multidimensional sensitivity code to analyze critical experiments and their effect on core-physics design of fast reactors. Work on multidimensional analysis of thermal reactors is also being pursued.

In the third category a variational approach to sensitivity theory has been used to define high order interpolation schemes to calculate the effects of density or material changes and reactivity insertions on reactor and shield performance. A second-order perturbation theory has also been developed for density and material changes in shielding problems. A novel approach to sensitivity theory in core-physics problems using integral transport theory has also been studied.

Finally, cross-section adjustment methods in conjunction with sensitivity theory and cross-section covariance data have received renewed interest in the U.S. Adjustment approaches incorporated into the U.S. FORSS system have been used to study the effects of integral experiments in the uncertainty analysis of shielding and core physics design problems, and other work has been carried out on an adjustment code for fusion and LWR systems. Much improvement has been made in the definitions and handling of covariance data for both differential and integral measurements in the U.S. approach to this problem. This work emphasizes completeness and an analytic capability to determine both calculational design uncertainties with and without integral experiments and differential and integral measurement program goals.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

3.10. DATA ADJUSTMENT: A CAUTIOUSLY OPTIMISTIC VIEW FOR THE IMPROVEMENT OF DESIGN PERFORMANCE CALCULATIONS AND DATA ASSESSMENT*

C. R. Weisbin E. M. Oblow J. H. Marable M. Salvatores†

(Reprints of Trans. Am. Nucl. Soc. 27, 861-888, 1977)

The purpose of this paper is to discuss experience obtained using data adjustment techniques for (a) quantifying uncertainties in calculated reactor performance, and (b) determining measurement priorities to reduce these uncertainties below prescribed design target accuracies. Data adjustments have been performed for fast reactor cores, fast reactor shields, and for light-water reactor applications. These investigations have been a continual learning process! The results often signal clear and bold action; however, many of the calculations have been found to be particularly sensitive to the input covariances, to the completeness of the material/reaction set included in the adjustment, to the technique (or lack thereof) for the incorporation of methods uncertainties, and to the difficulties associated with the range of applicability of such adjustments (particularly for shielding applications with relatively high sensitivities).

Since many of these difficulties are being resolved with the introduction of expanded covariance files, inclusion of more integral experiments, and incorporation of methods uncertainties, the view of the authors is one of cautious optimism leading to the development of large libraries of cross sections and covariance files. These files will include, consistently, the input from differential and integral experiments.

Current estimated values for the standard deviations in calculated k and central $^{235}\text{U}(x,y)$ for ZPR-6/7 (ref. 1) are 2 and 7%, respectively, using covariance files that include the effects of uncertainties in the ^{235}U fission standard. (Our earlier work in this area using more preliminary covariance files is discussed in refs. 5 and 6.) Similar uncertainties were obtained for other reaction rate ratios (e.g., $^{235}\text{U}(x,y)$) and other critical experiments (e.g., ZPR-3/56B). The calculations and measurements reported for ZPR-6/7 lead to calculation/experiment ratios of 0.9911 and 1.086, respectively (both within the nuclear-data-induced uncertainties). Using a cross-section adjustment scheme, along with assigned uncorrelated standard deviations for ZPR-6/7 and ZPR-3/56B of 1 and 2% for eigenvalue and reaction rate ratios, the standard deviations for k and breeding ratio in a 1200-MW(e) model of an LMFBR are reduced to 0.6 and 1.7%. This is to be compared to 2 and 6% if the same analysis was based solely on differential data. Statistical tests indicated that the adjustment was consistent with the ^{235}U information. The inclusion of the integral data caused significant reductions to be made in the $^{235}\text{U}(x,y)$ cross sections, the inevitable result based on a substantially overpredicted (8%) central $^{235}\text{U}(x,y)$ ratio and an assumed integral experiment uncertainty of 2%. The reduction of the $^{235}\text{U}(x,y)$ cross section was not found to be particularly sensitive to the cross-section covariance description. However, a hypothetical 3% increase over the reported experimental $^{235}\text{U}(x,y)$ value, combined with an assumed uncertainty of 3.5% (rather than 2%), clearly reduced the need for adjustments to the $^{235}\text{U}(x,y)$ cross section. Such sensitivity to the input integral experiment data emphasizes the need for reliable integral covariance data and suggests that the adjustments to particular cross-section energy groups be treated with circumspection.

Another study was recently performed² to explore the constraints imposed on the $^{235}\text{U}(x, \gamma)$ standard (proposed ENDF/B-IV) by information deduced from clean integral measurements (GODIVA and JEZEBEL)⁷ and to demonstrate how uncertainties in fission cross-section standards propagate in an uncertainty analysis and interpretation of those experiments. Inclusion of integral information was effective in reconciling all parameters other than $\langle 26\sigma_f/^{235}\text{U} \rangle$ measurement in JEZEBEL, for which current calculation and measurement are in disagreement. However, as pointed out in the paper, the lack of information with respect to inelastic covariances as well as that for the fission spectrum suggests that one be careful in assessing data needs for $^{235}\text{U}(x, \gamma)$ based on the analysis of these benchmark assemblies. Covariance files must be complete with respect to cross sections of high sensitivity, particularly those that are less well known (i.e., have large covariances).

In an analysis of light-water reactor criticals (i.e., TRX-2),⁸ a tentative adjusted set of cross sections was obtained, which was consistent with the microscopic measurements. The actual adjustments to the cross-section data were minimal such that the recommended evaluation, based solely on differential measurements, could be used without adjustment while maintaining consistent agreement with calculated TRX-2 performance. This was particularly important since it was found that uncertainties due to computational methods were comparable to the nuclear-data-induced uncertainties in the calculation as well as the uncertainties in the reported measurements. Schemes to define and include estimates for methods approximations in the adjustment procedures are only now under development.

In our LMFBR shielding analysis,⁹ results from an adjustment including 21 fast reactor shield configurations were used to draw conclusions about the need for more accurate data for sodium and iron elastic and discrete inelastic cross sections above 1 MeV and the values of the total cross section in the vicinity of important minima. The experience with adjustment in this problem suggests that one must be wary! Shielding calculations, in general, are far more sensitive to differential data, which are usually much less accurate. Small adjustments can have large effects on predicting shield performance. Analyses using only a small subset of the TSF experiments resulted in adjustments that failed by a wide margin to predict the results of the remaining measurements. The need for completeness in the scope of integral experiments included in the adjustment (i.e., experiments that, collectively, are sensitive to a broad range of data) is clearly evident.

In light of the cumulative experience reported above, large-scale efforts are now under way to perform adjustments with sets of clean, integral benchmark information using documented covariance files for integral and differential data. The latter is being extended to include most important reaction types and correlations (e.g., x, σ_{in}). Intensified effort is under way¹⁰ to identify and separate methods and data uncertainties. Our adjustment tool has been extended to include correlated uncertainties between integral experiments as well as approximate methods uncertainties. The adjusted data are being tested by direct recalculation of criticals used in the adjustment, and more complicated experiments explicitly left out. Covariance files are output directly for the adjusted data and the conventional statistical tests are used to demonstrate consistency with the *a priori* information. In summary, this larger effort should be considered as a first step in the development of adjusted libraries and uncertainties for use in design, whose genealogy is well-documented and whose contents are publicly available for review.

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† Comitato Nazionale Energia Nucleare, Casaccia, Italy.

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3.11. SENSITIVITY AND UNCERTAINTY ANALYSIS FOR FUNCTIONALS OF THE TIME-DEPENDENT NUCLIDE DENSITY FIELD*

M. L. Williams C. R. Weisbin

(Abstract of ORNL-6838, ENDF-863, April 1978)

An approach to extend the present ORNL sensitivity program to include functionals of the time-dependent nuclide density field is developed. An adjoint equation for the nuclide field has been derived previously using generalized perturbation theory; our derivation makes use of a variational principle and results in the same equation. The physical significance of this equation is discussed and compared to the time-dependent neutron adjoint equation. Computational requirements for determining sensitivity profiles and uncertainties for functionals of the time-dependent nuclide density vector are developed within the framework of the existing FORSS system, significantly extending the current capability. The development, testing, and use of an adjoint version of the ORIGEN isotope generation and depletion code are documented. Finally, a sample calculation is given which estimates the uncertainty in the plutonium inventory at shutdown of a PWR due to assumed uncertainties in uranium and plutonium cross sections.

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3.12. SENSITIVITY THEORY FOR DEPLETION ANALYSIS*

M. L. Williams J. R. White J. H. Marable E. M. Oblow

(Abstract of paper presented at 1978 Seminar-Workshop on Theory and Application of Sensitivity and Uncertainty Analysis, Oak Ridge, Tennessee, August 22-24, 1978; proceedings to be published by ORNL-6840)

This study discusses a general perturbation theory for two of the common approximations used in burnup analysis. In this test these types are called the uncoupled and

the coupled formalisms. The distinction lies in how the interaction between the nuclide and neutron fields is treated. It is shown that based on the coupled approximation three adjoint equations (for the flux shape, the flux normalization, and the nuclide density) are required to fully develop a perturbation theory which accounts for the coupled variations in the neutron and nuclide fields arising from variations in initial conditions and nuclear data. The adjoint equations are derived in detail using a variational principle.

A sensitivity methodology is established which seeks to estimate the change in various computed quantities when the input parameters to the burnup calculation are varied. The perturbation expressions are used to define sensitivity coefficients for responses that depend on the coupled interaction between the neutron and nuclide fields. The relation between coupled and non-coupled sensitivity theory is illustrated.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology, and by Electric Power Research Institute.

† Computer Sciences Division.

3.13. SENSITIVITY THEORY FOR GENERAL NONLINEAR ALGEBRAIC EQUATIONS WITH CONSTRAINTS*

E. M. Oblow

(Abstract of ORNL/TM-8815, April 1977, and of Nucl. Sci. Eng. 65, 187-191, 1978)

The general development of a sensitivity theory for nonlinear algebraic equations with constraints is presented. Adjoint equations suitable for evaluating derivatives of system response functions with respect to input parameters are derived. The role of the solution of the constrained problem in eliminating nonessential constraints is highlighted. Two sample problems, one linear and one nonlinear, are solved to illustrate the theory.

* Research sponsored by DOE Office of Fusion Energy, Division of Development and Technology.

3.14. SENSITIVITY THEORY FOR REACTOR THERMAL-HYDRAULICS PROBLEMS*

E. M. Oblow

(Abstract of ORNL/TM-8303, July 1978, and of paper to be published in Nucl. Sci. Eng.)

A sensitivity theory based on reactor physics experience was successfully developed for a reactor thermal-hydraulics problem. The new theory is derived for the case of nonlinear, transient heat and mass transfer in a typical reactor subassembly. Suitable adjoint equations for heat and fluid flow are presented along with methods for deriving the sources and boundary and final conditions for these equations. Expressions for the sensitivity of any integral temperature response to problem input data are also presented. The theory is applied to a sample problem describing the steady-state thermal-hydraulic conditions in a CRBR fuel channel. For this case, sensitivity coefficients are derived for several thermal response functions (i.e., peak clad and peak fuel temperature) for all physical input data (i.e., the heat transfer coefficient, thermal conductivities, etc.). A typical uncertainty analysis for peak clad and peak fuel temperature was also performed using uncertainty information about the physical data. Conclusions are drawn about the applicability of this approach to more general problems and the procedures for its implementation in conjunction with large safety or thermal-hydraulics codes are outlined. The method is also compared with currently used response surface techniques.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

3.15. HIGH ORDER EFFECTS IN CROSS SECTION SENSITIVITY ANALYSIS*

E. Greenspan[†] Y. Karni[†] D. Gilai[†]

(Abstract of paper presented at ANS Technical Workshop on Theory and Application of Sensitivity and Uncertainty Analysis, Oak Ridge, Tennessee, August 21-24, 1978; proceedings to be published in ANS/AC10-88)

Two types of high order effects associated with perturbations in the flux shape are considered: Spectral Fine Structure Effects (SFSE) and non-linearity between changes in performance parameters and data uncertainties. SFSE are investigated in Part I using a simple single resonance model. Results obtained for each of the resolved and for representative unresolved resonances of ²³⁸U in a ZPR-6/7 like environment indicate that SFSE can have a significant contribution to the sensitivity of group constants to resonance parameters. Methods to account for SFSE both for the propagation of uncertainties and for the adjustment of nuclear data are discussed.

A Second Order Sensitivity Theory (SOST) is presented, and its accuracy relative to that of the first order sensitivity theory and of the direct substitution method is investigated in Part II. The investigation is done for the non-linear problem of the effect of changes in the 297 keV sodium minimum cross section on the transport of neutrons in a deep-penetration problem. It is found that the SOST provides a satisfactory accuracy for cross section uncertainty analysis. For the same degree of accuracy, the SOST can be significantly more efficient than the direct substitution method.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

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3.16. SECOND-ORDER GENERALIZED PERTURBATION THEORY FOR SOURCE-DRIVEN SYSTEMS*

E. Greenspan[†] D. Gilai[†] E. M. Oblov

(Abstract of Nucl. Sci. Eng. 88, 1-8, 1978)

A second-order generalized perturbation theory (GPT) for the effect of multiple system variations on a general flux functional in source-driven systems is derived. The derivation is based on a functional Taylor series in which second-order derivatives are retained. The resulting formulation accounts for the nonlinear effect of a given variation accurate to third order in the flux and adjoint perturbations. It also accounts for the effect of interaction between any number of variations. The new formulation is compared with exact perturbation theory as well as with perturbation theory for altered systems. The usefulness of the second-order GPT formulation is illustrated by applying it to optimization problems. Its applicability to areas of cross-section sensitivity analysis and system design and evaluation is also discussed.

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3.17. SENDIN AND SENTINEL: TWO COMPUTER CODES TO ASSESS THE EFFECTS OF NUCLEAR DATA CHANGES*

J. H. Marable J. D. Drischler C. R. Weisbin

(Abstract of ANS/EN-6400, ENDF-260, July 1977)

A description is given of the computer code SENTINEL, which provides a simple means for finding the effects on calculated reactor and shielding performance parameters due to proposed changes in the cross-section data base. This code uses predetermined detailed sensitivity coefficients in SENPRO format. Knowledge of details of the particular reactor

and/or shielding assemblies is not required of the user. Also described is the computer code SENDIN, which converts unformatted (binary) sensitivity files to card image form and vice versa. This is useful for transferring sensitivity files from one installation to another.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

3.18. COVERT AND CAVALIER: TWO COMPUTER CODES FOR ESTIMATING UNCERTAINTIES OF CALCULATED NEUTRONICS PARAMETERS USING STANDARD INTERFACE FILES SENPRO AND COVERX*

J. D. Driscoll J. H. Marable C. R. Weisbin

(Abstract of OERL/TM-6076, ENDF-256, August 1978)

A description is given of the computer code CAVALIER, which provides a simple means for estimating the uncertainties in calculated neutronics parameters that result from uncertainties in nuclear data. This code uses predetermined detailed multigroup sensitivity coefficients in SENPRO format for specific reactor and/or shielding assemblies and multigroup cross-section covariance uncertainties in COVERX format. Also described is the computer code COVERT, which converts unformatted (binary) covariance files to card image form and vice versa. This is useful for transferring covariance files from one installation to another. These two codes, along with extended compilations of sensitivity coefficients and covariance matrix libraries, are currently available through the Radiation Shielding Information Center (RSIC) at Oak Ridge National Laboratory.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

3.19. UNCERTAINTIES IN THE BREEDING RATIO OF A LARGE LMFBR*

J. H. Marable C. R. Weisbin

(Abstract of paper presented at American Nuclear Society Topical Meeting on Advances in Reactor Physics, Gatlinburg, Tennessee, April 10-12, 1978, and published in CONF-780401, 231-243, 1978)

An uncertainty analysis for eight fast integral benchmark experiments was performed to find the most consistent set of values for both the integral and differential data as well as their associated covariances. Extensive covariance data including inelastic scattering for ^{235}U and uncertainties in the fission spectrum temperature were used in the adjustment.

The resulting covariances for the adjusted data were folded with calculated performance parameter sensitivities for a two-dimensional model of a conventional homogeneous plutonium-oxide fueled fast breeder reactor. Based on the estimated covariance files we find that the inclusion of integral experiment data reduced the uncertainty in the static breeding ratio to the design goal of two percent. Some apparent inconsistencies in the data were alleviated by including in an approximate way the effect of methods uncertainties and other possible systematic errors.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

3.20. CALCULATION OF THE FLUX COVARIANCE MATRIX FOR THE NATIONAL BUREAU OF STANDARDS INTERMEDIATE ENERGY STANDARD NEUTRON FIELD (NBS-ISNF)*

B. L. Broadhead[‡] H. L. Dodds, Jr.[‡] J. L. Lucius[‡] F. G. Perey C. R. Weisbin

Report of Div. Am. Nucl. Soc. 80, 890-896, 1978

In many dosimetry applications, quantification of the uncertainty in the flux spectrum of a given field is a necessity. In particular, dosimetry unfolding and other least-squares adjustment schemes require an estimation of the group flux uncertainties as well as group-to-group uncertainty correlations. However, due to the lack of sufficient data and the tools to manipulate these data, these uncertainties and their correlations have been neglected in the past. However, with the advent of several sensitivity codes and the availability of cross-section covariance data from ENDF/B-IV, this is no longer the case.

This study utilized the ORNL FORSS sensitivity system¹ along with cross-section covariance matrices generated from ENDF/B-IV uncertainty files to identify the major contributors to the NBS-ISNF flux spectrum uncertainty and to generate a full flux covariance matrix. The full flux covariance matrix is a unique, new result of this work in that previous work considered only the diagonal elements of the matrix or, in many cases, no uncertainties at all.

The procedure used to generate the full flux covariance matrix is summarized in the following two steps:

1. Using the NBS-ISNF model supplied by NBS,² perform a sensitivity analysis on the NBS-ISNF central region flux; i.e., the region utilized for dosimetry calibrations.
2. Fold the sensitivities obtained from Step 1 with ENDF/B-IV and ORNL-generated data covariance matrices to obtain the flux spectrum covariance matrix for NBS-ISNF.

The sensitivity analysis of Step 1 yields total relative sensitivity coefficients, defined as relative changes in a given response (the NBS-ISNF central region flux) per unit change in given cross sections per unit volume in phase space which are integrated over energy, and sensitivity profiles, which are plots of the energy-dependent relative sensitivity per unit lethargy versus energy. The total relative sensitivity coefficients allow an importance ranking of the major system constituents; i.e., ¹²C, ²⁷Al, ¹⁰B, and the fission source, while sensitivity profiles allow identification of various mechanisms peculiar to the NBS-ISNF facility.

Step 2 is the generation of the full flux covariance matrix due to uncertainties in the ¹²C total cross section and the fission spectrum. Cross-section covariance data for ¹²C were processed³ from ENDF/B-IV, while the fission spectrum covariance matrix was generated assuming a 2% uncertainty in the effective neutron temperature. Covariance data for ¹⁰B and ²⁷Al were not available, but it was felt that these materials would not significantly affect the results, since ¹⁰B is a standard and ²⁷Al was suspected and is shown to be unimportant.

The results of the sensitivity analysis show the dominance of the fission spectrum and ¹²C scattering in the NBS-ISNF facility with total relative sensitivity coefficients of the central region total flux to ¹²C total cross section, ¹⁰B absorption cross section, ²⁷Al total cross section, and the fission spectrum of 0.42, -0.20, -0.01, and 1.0, respectively.

The flux covariance matrix due to both ¹²C and the fission spectrum of NBS-ISNF is shown in Table 3.20.1. It should be noted that the covariance matrix is presented as a correlation matrix using standard deviations (° relative standard deviation) with corresponding matrix elements which have been multiplied by 1000 for clarity. It is significant that the relative standard deviations are quite small, affirming the validity of the term "standard field." But it is also significant that the elements of the matrix are strongly correlated. Thus, there is a strong motivation for not using only the diagonal elements of the flux covariance in a dosimetry cross-section adjustment application.

Table 3.20.1. NBS-ISNF flux correlation matrix (10⁻³)
 Note: Group 15 has been omitted

GROUP BOUNDARY	REL. ST. DEV.															
0.17330 06	4.9708	1000														
0.36790 07	2.7889	996	1000													
0.27250 07	0.3046	963	959	1000												
0.13530 07	0.4448	-802	-803	-631	1000											
0.67210 06	0.7631	-920	-920	-815	960	1000										
0.49790 06	0.9410	-937	-936	-809	923	995	1000									
0.40760 06	1.0062	-937	-936	-855	892	978	997	1000								
0.30200 06	1.1005	-947	-946	-872	889	978	991	994	1000							
0.22370 06	1.2182	-933	-932	-860	893	964	981	983	994	1000						
0.67300 05	1.1044	-854	-854	-780	844	899	904	901	928	955	1000					
0.31830 05	1.0941	-821	-820	-740	841	892	894	898	912	951	965	1000				
0.15030 05	1.0046	-673	-672	-593	749	785	788	774	797	836	852	884	1000			
0.12340 04	0.9156	-506	-505	-504	698	730	720	722	729	774	753	790	815	1000		
0.16700 03	0.8426	-475	-474	-399	627	644	641	641	639	700	680	734	791	991	1000	
0.13710 02																
0.10000-03																

As a result of this study, a quantitative foundation has been established which indicates that the NBS-ISNF flux spectrum uncertainty with respect to cross sections is dominated by uncertainties in the fission spectrum. This is in agreement with results of previous studies^{*} which were not based on a detailed quantitative analysis.

Finally, the full flux spectrum covariance matrix for the NBS-ISNF presented in Table 3.20.1 is strongly correlated with relatively small standard deviations, which indicates that the full flux covariance matrix should be used in cross-section adjustment procedures — not just the diagonal elements.

* Research sponsored by Nuclear Regulatory Commission, Division of Reactor Safety Research.

† University of Tennessee.

‡ Computer Sciences Division.

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3.21. SENSITIVITY AND UNCERTAINTY ANALYSIS APPLIED TO THE NBS-ISNF*

B. L. Broadhead[†] J. H. Marable

(Abstract of paper presented at RSIC Seminar-Workshop on Theory and Application of Sensitivity and Uncertainties Analysis, Oak Ridge, Tennessee, August 22-24, 1977; proceedings to be published in ORNL/RSIC-82)

Sensitivities and uncertainties have been obtained for input into a least-squares consistency analysis of the NBS-ISNF flux spectrum. One-dimensional ANISN transport calculations were used in the calculation of sensitivity coefficients. These sensitivity coefficients were then folded with covariance files to obtain the uncertainty in the NBS-ISNF central region flux spectrum due to cross-section and fission-spectrum uncertainties.

This flux covariance file, along with reaction rate ratio sensitivities and reaction cross sections with their covariances, was used to adjust the NBS-ISRF flux spectrum, reaction cross sections, and activations based on integral reaction rate ratio measurements and their uncertainties.

An initial calculated vs. experimental discrepancy for $^{235}\text{U}/^{238}\text{U}$ of 12.6% was removed by the adjustment. However, the cross sections and fluxes were adjusted by more than their standard deviations and the resulting chi-square per degree of freedom was 7, which indicated extreme inconsistency.

A finer energy mesh and replacement of a Maxwellian fission source by a preliminary-ENDF/B-V fission source gave a calculated vs. experimental discrepancy of only 4.3%. The resulting revised adjustment yielded a chi-square per degree of freedom of 1.2 which indicates good consistency among the various data. This revised adjustment included off-diagonal elements of the flux covariance matrix. If these off-diagonal elements are omitted, the chi-square per degree of freedom becomes 1.6, indicating only fair consistency.

* Research sponsored by Nuclear Regulatory Commission, Division of Reactor Safety Research.
† University of Tennessee.

3.22. SENSITIVITY AND UNCERTAINTY ANALYSIS FOR MIXED-OXIDE THERMAL REACTOR PARAMETERS*

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(Reprint of Trans. Am. Nucl. Soc. 30, 687-688, 1978)

The purpose of the work reported in this paper was to determine the sensitivity of mixed-oxide thermal reactor parameters to changes in nuclear data and the uncertainty associated with calculated performance parameters due to uncertainties in nuclear data. The sensitivity coefficients obtained were used to estimate the impact of ENDF/B-V.

The assembly studied was lattice U-L212, one of a series of water-moderated critical lattice experiments carried out at Battelle Northwest Laboratories.¹ This lattice is fueled with UO_2 -2 wt % PuO_2 (6% ^{239}Pu) and optimally moderated with borated water.

A 57-energy-group structure was selected for the analysis. This includes 35 thermal groups and 22 epithermal groups. The thermal (upscattering) range included energies up to 1.8 eV. The main consideration in the selection of the 22 epithermal groups was the adequate treatment of the four lowest ^{238}U resonances. One group was selected for each of these resonances. Resonance self-shielded cross sections were obtained from ENDF/B-IV data using the AMPX² system which utilizes the Nordheim method in the resolved resonance range. A cell calculation consisting of a single fuel pin surrounded by moderator with a reflected boundary condition and an axial buckling to account for the finite height of the fuel pin was performed to obtain the flux distribution within the cell. Energy-dependent flux disadvantage factors were used to obtain cell-averaged cross sections.

A full-core diffusion theory k calculation was performed using the VENTURE³ computer program. A two-dimensional, cylindrical (R-Z) model of the geometry with a homogenized core was used. The multiplication factor calculated was 1.0010, which agrees very well with the experiment.

The FORSS⁴ computer code system was used to perform a sensitivity and uncertainty analysis for lattice U-L212. The energy-integrated sensitivities for k are shown in Table 3.22.1. These numbers represent the percent change in k per percent change in the nuclear data uniformly in energy. Sensitivities were also obtained for four reaction rate ratios. The sensitivities were combined with covariance information obtained from error files for ^{235}U , ^{238}U , ^{239}Pu , and ^{240}Pu .

The estimated uncertainties in the calculated values of the performance parameters due to uncertainties in U and Pu cross sections are shown in Table 3.22.2.

Table 3.22.1

Nuclear Data	Energy-Integrated Sensitivity for k
$^{239}\text{Pu } \bar{\nu}$	7.9309-1
^{239}Pu fission	4.4398-1
^{239}Pu total	2.7560-1
^{239}Pu capture	-1.6858-1
^{240}Pu capture	-1.5757-1
^{238}U capture	-1.5323-1
$^{235}\text{U } \bar{\nu}$	1.4748-1
H total	1.4369-1
^{238}U total	-1.0241-1
H capture	-7.5710-2
^{235}U fission	7.2629-2
^{235}U total	5.6785-2
$^{236}\text{U } \bar{\nu}$	5.2789-2
^{240}Pu capture	-3.5525-2
^{238}U fission	3.5151-2
H total	3.0211-2
^{235}U capture	-1.5926-2
Zr capture	-5.4712-3

Table 3.22.3

Nuclear Data	Uncertainty (%)
$^{239}\text{Pu } \bar{\nu}$	0.47
$^{239}\text{Pu}(\nu, f)$	0.35
$^{238}\text{U}(\nu, \gamma)$	0.28
$^{239}\text{Pu}(\nu, \sigma_f)$	0.27
$^{240}\text{Pu}(\nu, \gamma)$	0.21

Table 3.22.2

Parameter	Uncertainty (%)
k	0.71
^{240}U (ratio of ^{238}U fissions-to- ^{235}U fissions)	2.4
^{240}U (ratio of epithermal-to- ^{238}U captures)	2.9
$^{239}\text{Pu}(\nu, \sigma_f)/^{235}\text{U}(\nu, \sigma_f)$	0.75
$^{240}\text{Pu}(\nu, \gamma)/^{235}\text{U}(\nu, \sigma_f)$	5.4

The largest contributions to the k uncertainty are given in Table 3.22.3. These numbers represent the uncertainty associated with the nuclear data listed only.

The contributions to k uncertainty occur predominately below 0.6 eV for the ^{239}Pu data. The ^{238}U capture contribution is mostly above 100 eV. The ^{240}Pu capture contribution is from the 1.05-eV resonance.

The predominant sources of the uncertainties in the reaction rate ratios are shown in Table 3.22.4.

The uncertainty in ^{240}U is predominately caused by the uncertainty in ^{238}U capture above 100 eV.

The effect of changing from ENDF/B-IV to ENDF/B-V was estimated by folding the percentage change in the cross sections from ENDF/B-IV to ENDF/B-V with energy-dependent sensitivity coefficients. The predominant projected changes are shown in Table 3.22.5.

Table 3.22.4

Parameter	Nuclear Data	Uncertainty (%)
^{240}U	$^{238}\text{U}(\nu, \sigma_f)$	2.3
^{240}U	$^{238}\text{U}(\nu, \gamma)$	2.9
$^{239}\text{Pu}(\nu, \sigma_f)/^{235}\text{U}(\nu, \sigma_f)$	$^{239}\text{Pu}(\nu, \sigma_f)$	0.63
	$^{235}\text{U}(\nu, \sigma_f)$	0.41
$^{240}\text{Pu}(\nu, \gamma)/^{235}\text{U}(\nu, \sigma_f)$	$^{240}\text{Pu}(\nu, \gamma)$	5.3

Table 3.22.5

Cross-Section Change	Impact (%)	Parameter
$\sigma_{f, \text{Pu}}^{\text{Pu}}$	0.11	k
$\sigma_{f, \text{Pu}}^{\text{Pu}}(\text{e}, \text{e})$	0.08	
$\sigma_{f, \text{U}}^{\text{U}}(\text{e}, \text{e})$	0.20	
ϕ (fission spectrum)	-0.08 1.91	β_{eff}
$\sigma_{f, \text{Pu}}^{\text{Pu}}(\text{e}, \text{e})$	-1.90	$\sigma_{f, \text{Pu}}^{\text{Pu}}(\text{e}, \text{e}) / \sigma_{f, \text{U}}^{\text{U}}(\text{e}, \text{e})$
$\sigma_{f, \text{U}}^{\text{U}}(\text{e}, \text{e})$	-1.42	β_{eff}

This work has resulted in significant new understanding of the sensitivities and uncertainties associated with performance parameters calculated for thermal mixed-oxide reactors. The ENDF/B-V projections indicate the impact of the proposed data changes on the calculated parameters for reactors of this type.

*Research sponsored by Electric Power Research Institute.

Computer Sciences Division.

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3.23. A COMPENDIUM OF ENERGY-DEPENDENT SENSITIVITY PROFILES FOR THE TRX-2 THERMAL LATTICE**

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Abstract of ORNL-5250, ORNL-234, March 1978

Energy-dependent sensitivity profiles for five responses calculated for the TRX-2 thermal lattice with the ORNL sensitivity code system FORSS are presented here both in graphical form and in SENPRO format. The responses are the multiplication factor, k_{eff} ; the ratio of epithermal-to-thermal captures in ^{235}U , β_{eff} ; the ratio of epithermal-to-thermal fissions in ^{235}U , β_{eff} ; the ratio of fissions in ^{235}U to fissions in ^{238}U , β_{eff} ; and the ratio of captures in ^{235}U to fissions in ^{235}U , CR. A summary table of the total sensitivities is also presented.

*Research sponsored by Electric Power Research Institute.

Companion report to ORNL-5250 (ENDF-234), "Compilation of Sensitivity Profiles for Several CELENG Fast Reactor Benchmarks," by J. H. Marable, J. L. Lucius, and C. R. Weisbin.

Computer Sciences Division.

3.24. SENSITIVITY ANALYSIS OF FAST REACTOR SYSTEMS INCLUDING THORIUM AND URANIUM*

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The Cross Section Evaluation Working Group (CSEWG) has, in conjunction with the development of the fifth version of ENDF/B, assembled new evaluations for ^{232}Th and ^{238}U (ref. 1). It is the purpose of this paper to describe briefly some of the more important features of these evaluations relative to ENDF/B-IV, to project the change in reactor performance based on the newer evaluated files and sensitivity coefficients for interesting design problems, and to indicate preliminary results from ongoing uncertainty analyses. It is important to note that, as of this writing, the current file is a "pre-preliminary" file developed in an expeditious fashion for alternative reactor system design studies. The preliminary ENDF/B-V file is expected to be approved in March and modifications to it are anticipated before general release following a period of integral data testing.

Mulki Bhat of BNL is coordinating the evaluation of ^{232}Th including recent measurements and evaluations of de Saussure,² Poenitz,³ Leonard,⁴ Derrien,⁵ Behrens,⁶ Blons,⁷ Fu,⁸ and Gwin.⁹ At higher energies (>50 keV) recent independent evaluations point to a significant reduction (-15 to 20%) in the ^{232}Th capture cross section. Independent evaluations for ^{232}Th inelastic show marked differences in cross-section and inelastic representation. Furthermore, the form of the fission spectrum has been changed to an energy-dependent Watt spectrum. The inelastic and fission spectrum shape changes impact directly the fast fission in ^{232}Th .

Stewart and Young¹⁰ have prepared a preliminary evaluation for ^{238}U . The fission cross section from 50 keV to 20 MeV is based mainly on Behrens' ratio measurements updated using the preliminary version V standard. More recent absolute fission measurements of Poenitz¹¹ indicate 2 to 8% increases in the ^{238}U fission cross section in the 200-keV to 1-MeV range, relative to ENDF/B-IV, with associated uncertainties of 2 to 3%. The ^{238}U capture, $\bar{\nu}$, and fission spectrum files are also clearly of interest for breeding and reactivity predictions.

The number of proliferation-resistant reactor designs¹² and symbiotic systems using ^{238}U and ^{232}Th is quite large. In this paper, we consider only 20 denatured ^{238}U - ^{232}Th and ^{238}Pu - ^{232}Th systems and analyze reactivity and breeding ratio sensitivities for major cross sections of interest. For example, in the 20 denatured system, the multiplication factor sensitivity to the $^{232}\text{Th}(\sigma_{\text{c}})$ cross section was calculated to be -0.15. Applying the energy-dependent k -profile, we project that the use of the pre-preliminary ENDF/B-V file should result in a k_{eff} $\approx$ 1.5% larger than that calculated with ENDF/B-IV.

Almost all (-95%) of this change comes from the 100-keV to 1-MeV range. For comparison, we note that for an LMFBTR using ^{238}U which is not denatured, McKnight and Turski¹³ calculate a total relative sensitivity of k to $^{232}\text{Th}(\sigma_{\text{c}})$ of -0.35 which is twice as large as that for the 20 denatured reactor design.

With respect to breeding ratio, a comparison¹⁴ between use of ENDF/B-IV $^{232}\text{Th}(\sigma_{\text{c}})$ and $^{232}\text{Th}(\sigma_{\text{c}})$ data reported by Macklin and Halperin¹⁵ (which is lower than the pre-preliminary ENDF/B-V file) gives a limiting example of calculated effects of breeding ratio sensitivity in the 20 denatured LMFBTR. The comparison shows a decrease in calculated breeding ratio of .5% in going from version IV to the Macklin data. More significant, perhaps, is the corresponding decrease of 50% in the calculated breeding gain. The same comparison made on a Pu-Th LMFBTR ^{238}U -producing "transmuter" again led to a 50% decrease in the breeding gain. This latter reactor type has been proposed for use in a symbiotic relationship with 20 denatured reactors. For such a compound system the doubling time estimates went from 34 years based on version IV data to 90 years based on the Macklin data. Each of these reactor types is currently being reanalyzed using the pre-preliminary ENDF/B-V data.

de Saussure and Macklin¹⁶ have provided estimated uncertainties in their evaluated ^{232}Th capture cross section. Below 50 keV this uncertainty is expressed in terms of uncertainties in the average level spacings, reduced neutron widths, and capture widths. From 50 keV to 1.5 MeV, they estimate a full correlated 1% uncertainty; above 1.5 MeV, a

30. uncertainty estimate is quoted. The folding of these uncertainties [due only to $^{232}\text{Th}(x,y)$] with sensitivity coefficients of LMFBR-type reactors led to predictions of up to 12% in the breeding ratio depending on the degree of denaturing. This 12% applies to a system that has no ^{235}U .

In the above we have pointed out some of the major proposed modifications in the ^{232}Th and ^{233}U files in the fast energy range. We have shown some of the impact of such modifications due to the sensitivities of typical LMFBR designs presently under consideration. A preliminary uncertainty file for $^{232}\text{Th}(x,y)$ has been applied to predict uncertainties in performance parameters due to uncertainties in the data.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

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3.25. ADVANCES IN FAST REACTOR SENSITIVITY AND UNCERTAINTY ANALYSIS*

J. H. Marable C. R. Weisbin

(Abstract of paper presented at DOE Summer Workshop on Theory and Application of Sensitivity and Uncertainty Analysis, Oak Ridge, Tennessee, August 21-24, 1978; proceedings to be published in ANL-80-11)

A review of present methods and existing computer codes at ORNL as well as elsewhere indicates an enormous capability not only to calculate sensitivity coefficients but also to apply them to a variety of purposes. Extensive files of sensitivity coefficients have been created and gathered for fast benchmark experiments and for two-dimensional models of fast power reactors of several designs, including k-reset effects when appropriate. These sensitivity files have been combined with covariance files which quantify the uncertainty in the differential nuclear data to predict the uncertainty in calculated performance parameters. Recent applications of these files have been made to adjust a large amount of both integral and differential data by incorporating benchmark experiment values and their covariances. By means of adjustments of this type, it is hoped that the differential data base as well as the integral data base can be improved.

However, there are still many limitations to our present capabilities. One of these limitations has been our inability to include in a complete and systematic way the effect

of methods uncertainties on the determination of adjusted data, which depends in general not only on experimental measurements, but also on estimates of covariances associated with the measurements and the methods. Also, the uncertainty in adjusted data contains contributions from uncertainties in covariance estimates, which contributions we have heretofore neglected. A new and comprehensive approach to include effects of methods uncertainties is presented here, and all sources which contribute to the uncertainty of the adjusted data are considered. This new approach is demonstrated using rough estimates for the methods uncertainties as applied to a simplified representation of the ZPR-6/7 fast benchmark. The results indicate that it may be essential to include methods uncertainties if integral experiments are to be used for the creation of adjusted nuclear data libraries. A careful evaluation of methods bias and uncertainties must still be performed.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

3.26. APPLICATION OF SENSITIVITY AND UNCERTAINTY METHODOLOGY TO FAST REACTOR INTEGRAL EXPERIMENT ANALYSIS*

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(Abstract of Nucl. Sci. Eng., 21, 222, 1976)

This paper presents the first results of a comprehensive application of the sensitivity theory developed for the FORSS code system to the analysis of fast reactor integral experiments. A variety of assemblies and performance parameters were studied to determine the nuclear data sensitivity as a function of nuclide, reaction type, and energy. Comprehensive libraries of energy-dependent sensitivity coefficients were developed in a computer retrievable format for several critical assemblies. Uncertainties induced by nuclear data were quantified using preliminary energy-dependent relative covariance matrices evaluated with ENDF/B-IV cross sections and processed for $^{235}\text{U}(x, y)$, $^{238}\text{U}(x, y)$, $^{239}\text{Pu}(x, y)$, $^{240}\text{Pu}(x, y)$, and $^{241}\text{Pu}(y)$. Computational results, cross-section covariances, and integral results and their covariances were used in a consistent fashion to improve uncertainty estimates of fast reactor core performance. A first attempt was made to quantify specifications for new cross-section measurements required to satisfy specific design goals at minimum experimental cost.

An analysis of several critical experiments indicated that design accuracy goals of 0.5% in k and 2% in the central ^{238}U capture: ^{239}Pu fission ratio ($\sigma_{f239}/\sigma_{c238}$) ratio in mixed oxide liquid-metal fast breeder reactor cores are unlikely to be attained in the near future. This assumes that the nuclear data are based only on microscopic measurements, and the current cross-section measurement program is not changed dramatically. Current estimates are 2.3% in k and 7.3% in central reaction ratio using only differential covariance information. Using the measurements in ZPR-6/7 for k and central $\sigma_{f239}/\sigma_{c238}$ in a cross-section adjustment scheme with assigned uncorrelated standard deviations of 1 and 2%, respectively, standard deviations of the same parameters were computed to be 0.7 and 1.8%. Results of integral experiments, therefore, are needed to improve uncertainty estimates of reactor performance for current fast reactor design work.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

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3.27. TWO-DIMENSIONAL REACTOR SENSITIVITY ANALYSIS USING DISCRETE-ORDINATES TRANSPORT THEORY*

R. L. Childs[†] J. H. Marable C. R. Weisbin

(Abstract of Trans. Am. Nucl. Soc., 41, 444-445, 1977)

The purpose of the work reported in this paper is to develop methods for performing reactor sensitivity analysis using two-dimensional (2-D) discrete-ordinates transport theory and to apply these methods to obtain the sensitivities of calculated performance parameters for the advanced fuels critical experiments (being performed on ZPR-9 at

Argonne National Laboratory) to changes in multigroup cross sections. The multidimensional transport sensitivity capability so developed represents a significant advance in our current capability to perform uncertainty analyses on designs with more realistic and more complex geometries.

Sensitivity coefficients may be defined as the percent change in a performance parameter (such as k or breeding) per percent change in some other parameter (such as an atom density or a cross section). When many sensitivity coefficients are desired, perturbation theory is the most efficient calculational method. For k sensitivities, ordinary perturbation theory is used, while linear ratios (such as breeding ratio) or bilinear ratios (such as reactivity worth) require generalized perturbation theory.

One-dimensional reactor sensitivity studies have been performed using both diffusion theory² and discrete-ordinates transport theory.³ The extension to 2-D discrete-ordinates transport theory is complicated by the source terms in generalized perturbation theory which are positive in some portions of phase space and negative in others. Discrete-ordinates computer programs normally employ algorithms that require non-negative sources and fluxes to obtain accurate solutions of the transport equation without excessive space mesh refinement. The method used for this work was to separate the sources into positive and negative components, solve the positive and negative portions separately, and then combine the components to obtain the generalized fluxes. This procedure is repeated for each outer iteration of the iterative solution which uses the method of successive approximations. This method has been implemented in the DOT-IV computer program.⁴ A "sweeping" procedure is used to remove any fundamental-mode component of the solution after each outer iteration.⁵ The computer programs VIP (ref. 6) and JULIET (ref. 7) were used to obtain sensitivity coefficients using the forward and adjoint flux moments and cross sections.

The model specified for the reference carbide benchmark assembly¹ was provided in R-Z geometry. The DOT-IV discrete-ordinates computer program was used with an S_8 angular quadrature and mesh spacings of approximately 3 cm. As a methods verification exercise, a series of four-group, P_0 scattering calculations was performed. The sensitivities obtained from the perturbation theory calculations were compared with sensitivities obtained by a direct method which involved separate transport calculations for a base configuration and two perturbed configurations corresponding to changes of $\pm 10\%$ for the parameter of interest. A quadratic fit through the three values is used to obtain the slope evaluated at the base configuration value. The two perturbations considered were density changes for ^{239}Pu and ^{14}C in the core. The k sensitivities obtained from ordinary perturbation theory were 0.5235 and -0.0215, respectively, while the direct method sensitivities were 0.5256 and -0.0225. This agreement is considered entirely satisfactory. Generalized perturbation theory was employed in the consideration of the central reaction rate ratio for ^{239}Pu absorption to ^{239}Pu capture. For this case, the generalized adjoint source is an energy-dependent, isotropic point source in space. A first-collision source method was used to calculate the generalized adjoint flux. For the ^{239}Pu and ^{14}C density perturbations, the reaction rate ratio sensitivities obtained from generalized perturbation theory were -0.0462 and 0.0795, respectively, while the direct method sensitivities were -0.0457 and 0.0791. This agreement is noteworthy. Density change comparisons are considered adequate for methods verification since they involve considerable cancellation of positive and negative contributions and are very sensitive to computational difficulties.

Subsequent to the verification step, calculations were performed using a resonance self-shielded cross-section library appropriate to the ZPR-9 assembly with 29 energy groups and a P_0 scattering expansion. Two performance parameters, k and the central reaction rate ratio for ^{239}Pu capture-to- ^{239}Pu fission, were considered. The sensitivity of these parameters to changes in nuclear data uniformly over all energy is shown in Table 3.27.1. Energy-dependent profiles for many reactions and several performance parameters are available in 29 groups and are being put into the SENPRO library which is available from RSIC. The 2-D analysis allows the results to be given by region. The sensitivities in Table 3.26.1 are about 30% higher for the radial blanket than for the axial blanket. Thus, a 1-D sensitivity analysis for this specific case would probably be reasonably accurate. The core sensitivities dominate for all cases in Table 3.27.1 except for ρ of ^{239}Pu in the reaction rate ratio case where the blanket sensitivities are considerably larger and of opposite sign of that in the core.

Table 3.27.1. Total sensitivity coefficients for the carbide benchmark critical

	k Sensitivities				Central β^{22}/β^{49} Sensitivities			
	Core	Radial Blanket	Axial Blanket	Entire Reactor	Core	Radial Blanket	Axial Blanket	Entire Reactor
Fission:								
^{239}Pu	.5739	0	0	.5739	-.0970	0	0	-1.0970 ^a
^{240}Pu	.0184	0	0	.0184	-.0006	0	0	-.0006
^{241}Pu	.0103	0	0	.0103	-.0019	0	0	-.0019
^{235}U	.0073	.0010	.0007	.0089	-.0018	.0003	.0002	-.0012
^{238}U	.0923	.0034	.0025	.0983	.0046	.0011	.0008	.0060
Capture:								
^{239}Pu	-.0467	0	0	-.0467	-.0305	0	0	-.0305
^{240}Pu	-.0066	0	0	-.0066	-.0043	0	0	-.0043
^{238}U	-.1966	-.0133	-.0096	-.2195	-.1431	-.0043	-.0030	-.8496 ^a
Fe	-.0086	-.0004	-.0003	-.0093	-.0042	-.0001	-.0001	-.0045
Ni	-.0039	-.0001	-.0001	-.0041	-.0017	-.0000	-.0000	-.0018
Cr	-.0038	-.0002	-.0001	-.0041	-.0028	-.0001	-.0000	-.0029
Nu:								
^{239}Pu	.7856	0	0	.7856	-.0030	0	0	-.0030
^{240}Pu	.0262	0	0	.0262	-.0002	0	0	-.0002
^{238}U	.1506	.0051	.0038	.1595	-.0007	.0017	.0012	.0022

^aA direct effect is included.

In conclusion, 2-D reactor sensitivity analysis techniques have been developed and substantially verified. The use of transport calculations allows studies to be performed where the use of diffusion theory is questionable. The methods provide an additional tool that may be used by reactor designers to perform sensitivity and uncertainty analyses and by cross-section evaluators to determine the effect of changing basic nuclear data. Such analyses are currently being employed to understand the nuclear performance of advanced fuels systems and to improve reference multigroup data libraries.

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† Computer Sciences Division.

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3.28. SENSITIVITY ANALYSIS APPLIED TO AN INTEGRAL TEST OF NIOBIUM CROSS SECTIONS*

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(Abstract of paper presented at RSIC Seminar-Workshop on Theory and Application of Sensitivity and Uncertainty Analysis, Oak Ridge, Tennessee, August 22-24, 1978; proceedings to be published in ORNL/RIC-40)

Simultaneous neutron and gamma-ray measurements were made of the leakage spectra from a ²⁵²Cf spontaneous fission source located at the center of a solid 9.6-cm-thick spherical shell of niobium. The NE-213 data were unfolded above 1 MeV and compared on an absolute basis to one-dimensional transport calculations, thus providing an integral test of Nb cross-section data and multigroup processing. Several coupled neutron-gamma-ray cross-section sets were investigated including the fine-group VITAMIN-C library.

Calculations based on ENDF/B-III and -IV data were in good agreement with the measured neutron spectrum below 10 MeV but diverged from the measurement above 10 MeV. ENDF/B-III data were in generally good agreement with the gamma-ray measurement while ENDF/B-IV gave an overestimation of the leakage. Sensitivity analysis showed that inelastic reactions were the dominate gamma-ray production mechanism for Nb, and that the gamma-ray leakage had much larger cross-section sensitivities than the neutron leakage. Based on this integral test, it is suggested that a reevaluation of ENDF/B-IV Nb data be made for inelastic processes above 1 MeV and all processes above 10 MeV.

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3.29. A COMPARISON OF EXPERIMENTAL AND CALCULATED RESONANCE INTEGRALS FOR UO₂ AND ThO₂ ISOLATED RODS*

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(*Trans. Am. Nucl. Soc.* 30, 738-739, 1973)

Resonance integrals for UO₂ and ThO₂ isolated rods have been calculated for several rod radii using ENDF/B-IV data. These results are presented as functions of the surface-to-mass ratio (S/M) and are compared with experimental results represented by the Hellstrand Correlation¹ for UO₂ and as reported by Hardy and Palowitch² for ThO₂. A novel method was used for obtaining a flat asymptotic flux in the moderator surrounding the rods.

The calculations were based on a cylindrical rod in an H₂O moderator of large radius. The model thus depended only on a single radial coordinate. Multigroup cross sections in 131 groups³ were obtained by a MINX⁴-SPHINX⁵-AMPX⁶ sequence. The Bondarenko scheme employed by MINX was used to account for temperature and dilution effects. The multigroup transport code ANISN⁷ was used to account for the neutron flux and reaction rate calculations.

The definition used for the resonance integral is

$$RI = \frac{1}{V_{\text{fuel}}} \int_{\text{fuel}} d^3r \int_{\text{res}} du \frac{\sigma_c(u) \phi(r,u)}{\phi_{\text{as}}}, \quad (1)$$

where $\sigma_c(u)$ is the capture cross section for the nuclide of interest, $\phi(r,u)$ is the spatially dependent flux per unit lethargy in the fuel, V_{fuel} is the volume of the fuel, and ϕ_{as} is the asymptotic flux; i.e., that flux that would exist in the absence of resonance absorption.

The corresponding multigroup representation for the resonance integral is

$$RI = \left(\frac{V_{\text{mod}}}{V_{\text{fuel}}} \right) \left[\frac{\sum_g \sum_s V_s \sigma_{c,g} \bar{\phi}_{s,g}}{\sum_g \left(\sum_s V_s \sigma_{c,g} \right) \frac{1}{\Delta U}} \right], \quad (2)$$

where g is the energy group, V_{mod} is the volume of the moderator, and V_s is the volume of interval s corresponding to the multigroup flux $\bar{\phi}_{s,g}$. The asymptotic flux here is obtained by averaging the flux over the moderator volume and over the entire lethargy width ΔU .

To obtain a flat asymptotic flux (i.e., $1/E$) in the moderator, a multigroup neutron source S_g was introduced into the moderator. For an infinite, nonmultiplying medium, assuming no leakage or upscatter, the neutron balance equation which determines this source is

$$-L_{R,g} \bar{\phi}_g = \sum_{g'=g+1}^{J_g} \sigma_{R,g'} \bar{\phi}_{g'} = S_g, \quad (3)$$

where $L_{R,g}$ is the "removal" cross section

$$L_{R,g} = L_{T,g} - \sigma_{R,g}(g' \rightarrow g). \quad (4)$$

The summation in Eq. (3) is carried out for all groups corresponding to energies below the upper cutoff energy for the resonance integral. Thus J_g is the energy group above which no source neutrons are introduced.

Requiring an asymptotic flux in the moderator implies $\phi(r) = \text{constant}$. Setting the constant to unity, a computer program was written to compute the multigroup source in Eq. (3). This calculated distributed source was then used in the moderator for the ANISN calculation of the rod model. Although there was some slight spatial variation of the flux within the moderator, the average moderator flux was remarkably flat ($1/E$).

Two isolated rod cases were analyzed. The surface-to-mass ratios for each case were varied by changing the rod radii. Figures 3.29.1 and 3.29.2 show the resonance integrals calculated for the UO₂ Case 1 and ThO₂ Case 2 studies. Calculated ²³⁵U resonance integrals are in excellent agreement with the Hellstrand Correlation while the ²³²Th resonance integrals are approximately 6% lower than experimental values presented by Hardy and Palowitch. The source of the ²³²Th discrepancy is believed to be the ENDF/B-IV data and/or the group structure used. (The group structure was tailored to finely resolve the first four ²³⁵U resonances below 100 eV, which leaves most of the ²³²Th resonances spanned by rather broad groups.)

In summary, the method for calculating resonance integrals using ENDF/B-IV data and transport theory proves to be accurate for ²³⁵U rods by providing a means by which to calculate an "ideal" experiment involving an isolated rod residing in an asymptotic

Fig. 3.29.1. Resonance integrals vs. S/M for UO_2 rods.

Fig. 3.29.2. Resonance integrals vs. $(S/M)^{1/2}$ for ThO_2 rods.

background flux field. Additional studies are under way to identify and ameliorate the sources of differences between the calculated and experimental resonance integrals for $^{232}ThO_2$ rods.

background flux field. Additional studies are under way to identify and ameliorate the sources of differences between the calculated and experimental resonance integrals for $^{232}ThO_2$ rods.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Nuclear Research and Applications.

Computer Sciences Division.

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3.30. ANALYTICAL INEQUALITIES SATISFIED BY THE CROSS-SECTION SELF-SHIELDING FACTORS*

Dan G. Cacuci M. A. Bjerke†

(Abstract of paper presented at RSIC Seminar-Workshop on Multigroup Nuclear Cross-Section Processing, Oak Ridge, Tennessee, March 14-16, 1978; Oak Ridge National Laboratory Report ORNL/RSIC-41, 227-252, 1978)

Recently, questions have arisen concerning the limits of the range of values within which a cross-section self-shielding factor (f-factor) is restricted. In this study, best upper and lower bounds for the range of values of an arbitrary f-factor have been derived analytically starting from the definitions which form the basis for the computer codes currently employing the self-shielding method. A fundamental assumption used in the present derivation is that the fluxes and cross-sections be elements of a Hilbert Space. The strict upper and lower bounds for the f-factors have been obtained by applying well-known inequalities from the theory of functions. In particular, it is shown that total and partial reaction f-factors can exceed unity. The conditions leading to such a situation are discussed from both a mathematical and a physical point of view.

Due to its ease of application, it is planned to implement the results of this study in a routine checking procedure in codes employing the self-shielding method.

*Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

†Computer Sciences Division.

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4.0. INTRODUCTION

Our integral experiments are designed to test either radiation penetration analysis techniques for certain types of shields or the nuclear data important to specified applications. In one extreme, we measure count rates in specialized detectors in the steady radiation field of a complex configuration driven by a reactor or a 14-MeV neutron source, while in another extreme we measure at the Oak Ridge Electron Linear Accelerator (ORELA) as a function of incident energy and outgoing angle and energy using samples too thick to permit deduction of differential cross sections. The work in either case is closely tied to the Division's efforts in radiation shielding methods development and the advanced application of these techniques to difficult practical problems.

The unique Tower Shielding Facility is the most important source of integral information in support of fission reactor shielding design. Up to the beginning of the present reporting period its work had for some time been concentrated in proof-testing design tools for the most difficult LMFBR shield design problems revealed in FFTF and CRBRP design. As a benefit of the recent slowdown in LMFBR work, we have published a number of reports displaying and analyzing results in these critical shielding areas.

The current experimental program at the TSF is sponsored by the Gas Cooled Fast Reactor program, particularly to study shielding problems exacerbated by the radiation streaming characteristics of this type of reactor system. Figure 4.0.1 shows an example of the complex position dependence of the fast neutron flux in the neighborhood of the core support structure in current designs, a region in which estimated radiation damage is at a maximum. The GCFR shielding studies will continue this year with an experiment investigating pertinent aspects of designs having a thorium blanket.

Fig. 4.0.1. Radial fast-neutron detector traverses beyond GCFR grid plate shield mockup. These traverses, made at the Tower Shielding Facility directly behind the B₄C subassemblies (shown schematically in top view above the curves), illustrate the neutron streaming through the GCFR grid plate shield mockup when the subassemblies were spaced 1.27 cm apart.

Our newest venture in integral benchmark experiment and analysis is designed to assure that penetration analysis techniques are adequate for the design of shields in fusion power facilities. The first experiments involved measurements of neutron and gamma-ray spectra through an appropriately optimized model shield using 14-MeV neutrons from a $T(d,n)$ source. Some of the neutron spectra obtained are shown in Fig. 4.0.2.

ORNL-DWG. 78-21480

THE ATTENUATION OF 14-MeV NEUTRONS BY STAINLESS STEEL

Fig. 4.0.2. The attenuation of 14-MeV neutrons by two thick shields of stainless steel (type 304) is shown. The neutrons were produced by the $T(d,n)$ reaction and were detected with a 68.7-cc NE-213 detector located 154.5 cm from the target.

The source region for this apparatus was designed to have a neutron spectrum resembling that from a fusion device, including scattering effects. The whole experiment facility was thoroughly analyzed using transport calculations to assure that experimental results would not be confused by unmeasured backgrounds. Concurrent with analysis of the initial results such as Fig. 4.0.2, this experiment and analysis effort will move to study a simple duct penetration arrangement designed to represent some of the analysis problems which will be characteristic of shields for magnetic fusion energy devices. (Note: Sensitivity studies associated with these experiments are discussed in Papers 6.19 - 6.22.)

4.1. ANALYSIS OF A NEUTRON SCATTERING AND GAMMA-RAY PRODUCTION INTEGRAL EXPERIMENT ON ALUMINUM FOR NEUTRON ENERGIES FROM 1 TO 15 MeV*

S. N. Cramer E. M. Oblow

Abstract of ORNL/OR-0140, January 1978

Monte Carlo calculations were made to analyze the results of an integral experiment with an aluminum sample to determine the adequacy of ENDF/B-IV neutron scattering and gamma-ray production cross-section data for aluminum. The experimental results analyzed included energy-dependent NE-213 detector neutron and gamma-ray count rates at a scattering angle of 125 deg and pulse-height spectra for scattered neutrons and gamma-rays. The experiments were carried out with the ORELA 1- to 20-MeV pulsed neutron source. The pulse-height data were unfolded to generate secondary neutron and gamma-ray spectra at 125 deg as a function of incident neutron energy. Multigroup Monte Carlo calculations using the MORSE code and ENDF/B-IV cross sections were made to analyze all reported results. Discrepancies between calculated and measured responses were found for secondary neutron scattering data above 10 MeV and for gamma-rays produced at energies between 4 and 7 MeV. A detailed analysis has not yet been performed to determine the reasons for these discrepancies.

* Research sponsored by Defense Nuclear Agency under Subtask PE083.

4.2. ANALYSIS OF A NEUTRON SCATTERING AND GAMMA-RAY PRODUCTION INTEGRAL EXPERIMENT ON SILICON DIOXIDE FOR NEUTRON ENERGIES FROM 1 TO 15 MeV*

S. N. Cramer E. M. Oblow

Abstract of ORNL/OR-0141, January 1978

Monte Carlo calculations were made to analyze the results of an integral experiment with a sample of SiO₂ to determine the adequacy of ENDF/B-IV neutron scattering and gamma-ray production cross-section data for silicon and oxygen. The experimental results analyzed included energy-dependent NE-213 detector neutron and gamma-ray count rates at a scattering angle of 90 deg and pulse-height spectra for scattered neutrons and gamma rays. The experiments were carried out with the ORELA 1- to 20-MeV pulsed neutron source. The pulse-height data were unfolded to generate secondary neutron and gamma-ray spectra at 90 deg as a function of incident neutron energy. Multigroup Monte Carlo calculations using the MORSE code and ENDF/B-IV cross sections were made to analyze all reported results. No outstanding discrepancies between calculated and measured responses were found on the neutron data below 12 MeV. Possible discrepancies in the inelastic scattering data above 12 MeV are indicated. This is consistent with a previous analysis of an oxygen experiment. A more detailed analysis will have to be performed before any definite conclusions can be drawn from these comparisons.

* Research sponsored by Defense Nuclear Agency under Subtask PE083.

4.3. AN INTEGRAL MEASUREMENT TO TEST EVALUATED NEUTRON CROSS SECTIONS FOR CARBON*

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Measurements have been made of the spectra of scattered neutrons from a thick carbon sample for incident neutron energies from 1 to 20 MeV. This experiment was performed as part of a program at ORNL to provide data suitable for testing the evaluated neutron data files for those elements which are important in the analysis of radiation transport problems for thermonuclear neutron sources.

The experimental technique is capable of sufficient resolution in the energy and angle of both incident and secondary neutrons to provide data with localized sensitivity. Thus, although it is necessary to compare the evaluated cross-section file to the experiment by means of a calculation to include the finite-sample effects, there is still an essentially direct differential relation between the calculated secondary spectra and the cross-section data.

This technique offers the additional advantage of testing the cross-section sets in the same context as they are used in applied transport problems. That is, the process by which microscopic evaluated cross sections are converted to tailored group constants is also checked.

Comparisons of the measurements and calculations based on the current evaluation, ENDF/B-IV MAT 1274, are presented in graphical form. The comparisons indicate the evaluated data for elastic scattering are very good up to incident energies of about 8 MeV. Above 8 MeV there are discrepancies of the order of 20-40%. For the angular range of this measurement, the evaluated inelastic scattering to the 4.43-MeV level appears to be 10-20% too low for $E_{\text{inc}} < 10$ MeV, but rather good above $E_{\text{inc}} = 10$ MeV.

*Research sponsored by DOE Office of Basic Energy Sciences, Division of Nuclear Sciences.

4.4. THE THERMOLUMINESCENT DOSIMETER MEASUREMENT OF GAMMA-RAY ENERGY DEPOSITION DISTRIBUTIONS IN REACTOR SHIELDS*

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This paper describes the progress made in the adaptation of thermoluminescent dosimeters (TLDs) to the measurement of gamma-ray energy deposition distributions in reactor shields. The TLD has been extensively used by the Argonne National Laboratory (ANL) to measure gamma-ray heating rates in the reactor core, blanket, and control rods of the LMFBR critical assembly. Encouraged by ANL's success and because of project requirements, an experimental program was undertaken to investigate the use of TLDs to measure gamma-ray heating in reactor shields.

Accuracy requirements for the shielding application entailed additional refinements in the basic readout equipment supplied by the Harshaw Chemical Company and in the experimental and readout procedures. A more precise measurement of gamma-ray energy deposition on an absolute basis was achieved and also a better understanding of the sensitivity of TLDs to neutron exposure. The procedures required to correct for the TLD's neutron response were developed. A series of gamma-ray heating benchmark experiments and supporting transport calculations were performed as an integral part of the TLD program. These experiments included gamma-ray energy deposition measurements in typical reactor shield configurations due to pure isotopic gamma-ray sources as well as mixed neutron and gamma-ray fields typical of fast reactor radiation environments. Measurements were made with LiF-TLDs and a high-pressure ion chamber as well as the CaF₂-TLDs.

A TLD dosimetry system acceptable for use in shielding experiments was successfully developed having accuracies of $\pm 5\%$ or better on an absolute basis. This accuracy can be expected for gamma-ray energy deposition measurements in even the maximum neutron-to-gamma ratio for radiation environments in reactor shields.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

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4.5. RADIATION HEATING STUDIES IN A STAINLESS STEEL AND SODIUM SHIELD*

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Transactions of the Am. Nucl. Soc. 27, 770-777, 1970

The purpose of this work was twofold: first, to establish a series of benchmark-type measurements of radiation heating in a stainless-steel and sodium system using two types of thermoluminescent detectors (TLDs) and an ion chamber, and second, to determine the accuracy of calculated heating rates using a standard design cross-section set based on ENDF/B-IV, together with derived response functions for each detector.

The experiment was performed at the Tower Shielding Facility using the large beam collimator.¹ The shield configuration in which heating measurements were made consisted of a spectral modifier followed by a blanket followed by 46 cm of stainless steel, 15 cm of sodium, and 13 cm of stainless steel. Both ^7LiF and $\text{CaF}_2\text{:Mn}$ TLDs were placed in situ at various locations beyond the blanket and several ion chamber measurements were also made in the stainless steel. In addition, neutron Bonner ball measurements were made throughout this configuration expanded to include an additional 15 cm of stainless steel, 120 cm of sodium, and 10 cm of stainless steel. This complete experiment was a followup to one performed earlier, in which the gamma-ray heating detectors were placed in air gaps between slabs of iron and stainless steel.²

The calculations used a 51n-25 γ coupled cross-section set that included delayed as well as prompt fission gamma rays for the blanket region and also the delayed 2.75- and 1.37-MeV gamma rays from ^{24}Na . The analysis was performed with version 3.5 of the two-dimensional discrete ordinates code DOT, starting with the absolute disk source at the collimator.³ An asymmetric 160-angle set biased in the forward direction was used throughout the calculations in lieu of a first-collision source routine. A total CPU running time on the IBM 360/195 of 11 h was required to calculate the fluxes throughout the complete series of configurations (each Bonner ball measurement was not made in situ but with 10 cm of steel behind it), even though bootstrapping techniques were used at several boundaries.

The gamma-ray spectral effect of replacing the TLD material with iron (the so-called $1/f$ factors) was calculated by Simons⁴ of Argonne/Idaho Falls, for both ^7LiF and $\text{CaF}_2\text{:Mn}$. The neutron response functions for the two materials were estimated by Yoon,⁵ based on his own kerma calculations and a survey of the open literature.

A comparison of the calculated and measured absolute gamma-ray heating rates is shown in Table 4.5.1, where the measured values have been corrected for spectral and neutron effects. Also shown in Table 4.5.1 is a comparison of the 4-in. Bonner ball counting rates, which is a relative measure of the total neutron flux in the region 0.1 eV to 100 keV.

An inspection of Table 4.5.1 shows that the corrections applied to the measured TLDs are accurate, and that the TLD responses are calculated as well as those of the ion chamber, which has no corrections. The increasing underprediction of the gamma-ray heating with penetration of the shield is seen to be due to the neutrons, which are the source of the secondary gamma rays. The coarse 51-group structure for the neutrons is apparently responsible for the factor of 2 underprediction in the radiation transport through the complete configuration, since it is unable to sense properly the minima in the total cross sections of stainless steel and sodium.

Table 4.5.1. Values of G/E for both the gamma-ray heating measurements and the 4-in. Bonner balls

	0' cm SS	10' cm SS	20' cm SS	23' cm SS	30' cm SS	40' cm SS	46' cm SS	15' cm Na	18' cm SS	120' cm Na	120' cm SS
CaF ₂ :Mn (TLD)	1.00	0.99	0.95	0.95	0.91	0.85		0.86	0.63		
LiF (TLD)	1.01	0.96	0.93	0.94	.89	0.81		0.80	0.60		
Ion Chamber		1.00		0.92	0.93						
4-in. BB			1.00		.96		0.83	0.77	0.67	1.50	0.67

Detector locations beyond the blanket. The full configuration is 10' cm Na + 9' cm Al + 20' cm SS + 20' cm Blanket + 46' cm SS + 15' cm Na + 18' cm SS + 120' cm Na + 6' cm SS + 10' cm SS.

Summarizing the results, the response functions for both LiF and CaF₂:Mn TLDs are adequately known and the heating rates of these TLDs can be calculated with ENDF/B-IV about as accurately as the neutrons can be calculated, i.e., the gamma-ray-production cross sections are adequate for sodium and the elements in stainless steel.

*Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

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4.6. ANALYSES OF TSF EXPERIMENTS ON RADIATION HEATING IN A STAINLESS STEEL-NORM CRBR RADIAL SHIELD MOCKUP USING A 32-INCH DIAMETER COLLIMATED BEAM SOURCE*

R. E. Maerker

Department of Nuclear Engineering, University of Tennessee

Analyses of the second phase of the radiation heating experiments performed at the Tower Shielding Facility have been made where the emphasis was placed on investigating the accuracy of TLD measurements in mockups of the CRBR radial shield. In particular, it is concluded that with the aid of flux calculations described in this report the corrections to both the LiF and CaF₂ TLD measurements are well enough understood that the TLD measurements can be brought into $\pm 3\%$ agreement with both ion chamber measurements with each other. The accuracy of the gamma-ray calculations is limited by the accuracy of the 51-neutron-group transport calculations; the production cross sections present in ENDF/B-IV are adequate for U²³⁵, sodium, and stainless steel.

*Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

4.7. MEASUREMENTS AND CALCULATIONS OF THE ORNL CRBR UPPER AXIAL SHIELD EXPERIMENT*

R. E. Maerker F. J. Muckenthaler C. E. Clifford

(Abstract of ORNL-5166, June 1977)

The measurements and analysis of the upper axial shield experiment performed at the Tower Shielding Facility using the large beam collimator have been described in a format suitable for eventual adoption as a CSEWG shielding benchmark. Neutron transmissions through various combinations of stainless steel followed by sodium and carbon steel over a range of attenuations of the order of 10^{12} were measured in this experiment. Calculations with a standard transport code using a standard design cross-section set based on ENDF/IV reveal disagreements with measurements behind the thickest shields by at most a factor of about three, but in most of the cases the agreement is within ~30%. One-dimensional scaling of this experiment has also been verified, and fine-group calculations indicate that the basic ENDF/IV cross-section data must be updated for sodium.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

4.8. A MEASUREMENT OF THE NEUTRON STREAMING IN A CRBR PROTOTYPIC COOLANT PIPE CHASEWAY*

F. J. Muckenthaler C. E. Clifford

(Abstract of ORNL/TM-5401, August 1977)

This report summarizes the measurements for a full-scale mockup of the coolant pipe chaseway for the CRBR design. The chaseway contained two 90-deg bends and four chokes, all located in the same horizontal plane. The coolant pipe, 91.4 cm in diameter and surrounded by 22.9 cm of Kaylo insulation, was divided into four separate sections of unequal lengths. Measurements could then be made as a function of pipe length. The sodium coolant was simulated by sodium carbonate. The TSR-II neutron energy spectrum was modified with 45.7 cm of stainless steel and 152.4 cm of sodium to simulate the spectrum incident on the CRBR chaseway. Measurements of the neutron flux, using a series of Bonner balls, indicated an attenuation of about 10^6 for the chaseway with all pipes present.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

4.9. DISCRETE ORDINATE ANALYSIS OF THE TSF EXPERIMENT ON NEUTRON STREAMING IN A CRBR PROTOTYPIC COOLANT PIPE CHASEWAY*

R. E. Maerker

(Abstract of ORNL/TM-6437, December 1978)

This report summarizes the results of calculations of neutrons streaming through a prototypic multilegged CRBR coolant pipe chaseway performed with the two-dimensional discrete ordinates code DOE 3.5, together with a specially written auxiliary routine that transformed the angular fluxes transmitted through one leg into an approximate source for the calculation of the next leg. Comparison with measured fluxes at intermediate points along the duct indicate an approximate factor-of-two accuracy in the S_n calculations using a recommended procedure for geometric modeling of each leg. The accuracy of an alternate geometric approximation was assessed and found to be generally inferior to that of the recommended procedure.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

4.10. MEASUREMENTS AND ANALYSIS OF THE CRBR II INCONEL-STAINLESS STEEL RADIAL SHIELD EXPERIMENT*

R. E. Maerker F. J. Muckenthaler

(Abstract of R.E.-5346, July 1978)

Neutron radial measurements and Bonner ball counting rates were made behind equal thicknesses of Inconel and stainless steel (from 13 cm to 39 cm) to determine which material was the better radial shield for an incident neutron spectrum similar to the one leaking from the proposed CRBR. Calculations of the experiment were performed using a two-dimensional discrete ordinates code, together with a standard design 51-group cross-section set based on ENDF/B-II. Both the calculations and the experiment indicated that Inconel is superior to stainless steel, but the Inconel cross sections are not as accurate in predicting the results of the experiment as are those of stainless steel.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

4.11. FINAL ANALYSIS OF A TSF EXPERIMENT ON NEUTRON STREAMING THROUGH A LATTICE OF GCFR-TYPE FUEL PINS*

C. O. Slater M. B. Emmett† F. J. Muckenthaler

(Abstract of ORNL/GCFR-73/5, February 1978)

This report describes the final analysis of an experiment performed at the ORNL Tower Shielding Facility to investigate neutron streaming through a fuel-pin lattice similar to but not typical of the proposed 300-MW(e) Gas Cooled Fast Reactor. In the experiment 894 enriched UO_2 fuel pins arranged on a triangular pitch inside a 0.459-m-ID steel pipe surrounded by water were positioned in a reactor beam modified to simulate a fast reactor spectrum, and neutron energy fluxes were measured beyond the assembly at various angles from its axis. The purpose of the experiment was to obtain data against which neutron-streaming calculational techniques being developed for the GCFR could be tested. The first calculations tested the adequacy of the discrete ordinates transport method using a two-dimensional cylindrical model for predicting the neutron streaming effects observed in the experiment. The results showed that when the pin lattice was included as a homogenized region in the model, no streaming effects were detected. When the lattice was treated as a heterogeneous region consisting of concentric annuli of fuel and void, streaming through the voids became apparent; however, the effects were considerably overpredicted. When last-flight estimation was used with the heterogeneous results, agreement between calculation and experiment was considerably better, but the calculated results were still higher than the measured results. Thus, the homogeneous and heterogeneous adaptations of the discrete ordinates method bracketed the pin streaming problem. Internal angular fluxes obtained in the discrete ordinates analysis were used as data in subsequent Monte Carlo calculations with semiaccurate-to-accurate geometric representation of the fuel lattice. The Monte Carlo results were still higher than the measured values but were generally much better than the discrete ordinates heterogeneous results. Sensitivity analyses showed which cross sections the calculated results were most sensitive to and also which regions of the experimental configuration had to be included in the Monte Carlo calculations.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

† Computer Sciences Division.

4.12. DESIGN OF A PROTOTYPIC GCFR GRID-PLATE SHIELD CONFIRMATION EXPERIMENT*

D. T. Ingersoll C. O. Slater M. L. Williams

(Reprint of Trans. Am. Nucl. Soc. 28, 650-631, 1978)

Axial neutron streaming in the low-density coolant channels in the core region of the proposed 300-MW(e) gas-cooled fast reactor (GCFR)¹ has stimulated considerable concern regarding localized radiation damage to the core-supporting grid plate. Since there does not currently exist any verified method for accurately calculating streaming in the detailed geometry of a fuel pin lattice, it is highly desirable to provide experimental verification of crucial design calculations such as grid-plate damage. A prototypic GCFR experiment has been designed which will provide a measured effect of neutron streaming from a fuel pin lattice and will provide data on the grid-plate shield effectiveness. To produce a GCFR-type environment in a small-scale experiment, detailed calculations were required for both the GCFR and the experiment design. Analysis showed that the final design satisfied all of the important criteria for a verification-type experiment.

A schematic diagram of the experimental configuration is shown in Fig. 4.12.1. The fuel pin region, consisting of seven hexagonal assemblies filled with Al-clad ²³⁵U-enriched pins, is driven by TSF reactor neutrons which penetrate the concrete shadow shield and stream into the annular void region surrounding a portion of the fuel pin region. The fuel pin assemblies, the B₂C-filled grid-plate shield assemblies, and the Fe grid plate mockup have dimensions and spacings representative of the current GCFR demonstration plant design.

Fig. 4.12.1. Experimental configuration of ORNL TSF experiment for the GCFR grid-plate shield design confirmation.

One- and two-dimensional neutron transport calculations of the experimental configuration were performed throughout the design phase using ANISN² and DOT IV.³ Initial calculations used 7 energy groups and S_8 angular quadrature, while later calculations used 51 groups and S_{12} quadrature. For the 2-D calculations, the experiment was modeled in cylindrical geometry with the fuel pin region represented by concentric rings of void and fuel.⁴ A boundary source was prepared from an existing angular flux distribution at the surface of the TSF reactor. Two-dimensional calculations of the GCFR design used 51 energy groups and both S_{10} and a biased 262-angle quadrature.

To produce a GCFR-type spectrum at the grid-plate shield, it was important that the dominant source of neutrons be from the fuel pin array. Calculations showed that fission and scattered neutrons from the fuel pins provided over 90% of the source incident on the grid-plate shield. Although the radial fission source distribution was significantly peaked at the edge of the pin array, the total flux below the grid-plate shield was nearly flat. Figure 4.12.2 compares neutron spectra at the top of the grid-plate shield. The finite extent of the pin array in the experiment caused the neutron energy distribution to be shifted to higher energy than in the semi-infinite GCFR lattice; however, the spectra were sufficiently similar to yield results that will be applicable to GCFR analysis.

Finally, calculations showed that the TSF experiment and the GCFR design had similar responses to variations in the B_{1C} density for both the lower and upper grid-plate shield sections. This was only true, however, when the experimental assemblies were prototypic of GCFR assemblies.

In conclusion, design of a prototypic GCFR grid-plate shield confirmation experiment has been completed. Results indicate that the experiment will provide a prototypic GCFR environment for the verification of computational methods important to the analysis of GCFR design. The experiment will also provide a direct measurement of neutron streaming from a fuel pin source.

Fig. 4.12.2. Comparison of neutron energy spectra for the TSF experiment and the GCFR design. The 51-group spectra were calculated on the axial centerline at the top of the grid-plate shield.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

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4.13. THE GCFR GRID-PLATE SHIELD DESIGN EXPERIMENT*

F. J. Muckenthaler J. L. Hull J. J. Manning

(Reprints of Trans. AM. Nucl. Soc. 26, 681-682, 1976; also summary of paper to be published as ORNL/TM-2650)

The grid-plate shield design confirmation experiment for the gas-cooled fast breeder reactor (GCFR) called for measurements at strategic locations in a prototypic configuration to provide a verification of the calculational methods used in the design. The experiment, performed at the ORNL Tower Shielding Facility, was designed so measurements could be made behind each of several prototypic segments as they were added to the configuration. The first segment modified the neutron beam from the TSF reactor such that its direct contribution to the flux beyond a region mocking up the GCFR fuel pins (the second segment) would be less than 10%. Calculations indicated that this could be obtained by attenuating the reactor neutrons with a 30.5-m concrete shadow shield. A 15.24-cm void surrounding the fuel pins allowed neutrons to enter the fuel pin region after they had scattered from the concrete wall that enveloped the void and had penetrated through a 0.953-cm Fe liner designed to contain the fuel pin subassemblies. The grid-plate shield and a prototypic grid plate constituted the third and fourth segments. The full configuration, with all mockup segments, is report in ref. 1.

There were two basic variables in the experiment, the number of fuel pins and the spacings between adjacent subassemblies. Most important in the combination of these variables (priority 1A) were measurements for 127 fuel pins per subassembly and a 1.27-cm void spacing between subassemblies. The spacing between subassemblies, both fuel pins and B.C., was changed to 0.635 cm for the second series (priority 1B). For the third priority (1C) measurements the spacing between subassemblies was expanded to 1.27 cm but the number of fuel pins per subassembly was reduced to 91. Next, a check was made of the neutron streaming through a control rod subassembly mockup (priority 2) by removing the central 19 pins from the center subassembly to simulate the void left by a removed control rod. For the lowest ranking priority (3) the number of pins was the same as for 1A but the spacing between subassemblies was reduced to 0.318 cm.

The fuel pins simulating the GCFR were slightly enriched (1.99 ^{235}U) UO_2 clad in aluminum wall tubing having a 1.27-cm o.d. These pins were contained in seven hexagonal-shaped subassemblies. The grid-plate shield contained B.C.-filled (density 76.2 to 77.2 of theoretical) subassemblies of the same shape as the fuel pin subassemblies. The prototypic grid plate consisted of 63.5 cm of Fe containing seven cylindrical holes whose centers were coincident with the axes of the fuel pin and B.C. subassemblies for priority 1A. Removable sleeves, containing B.C., were fabricated to represent the upper grid-plate shield sections found in the GCFR design.

Special interest in this experiment centered on the neutron streaming through the void between the B.C. subassemblies in the grid-plate shield and, hence, a measure of the neutron flux incident upon the grid plate. Since the maximum void to be studied was only 1.27 cm, it was necessary to select a detector whose sensitive volume was reasonably smaller than the void. A 0.635-cm-diam, 0.159-cm-thick Hornyak button, whose response function resembles that from a single-collision dose, was specially fabricated for these measurements. Initial results substantiate this selection as can be seen from the neutron flux peaks obtained with this detector behind the voids shown in the middle curve in Fig. 4.13.1. Placed with these data is a schematic diagram (as seen by the detector) of the three B.C. subassemblies (priority 1A) behind which the data were obtained. A similar traverse made behind the fuel pin subassemblies (accompanying schematic diagram also

THESE CURVES ARE THE FLUX DISTRIBUTIONS OBTAINED DIRECTLY BEHIND EACH OF THE THREE CONFIGURATIONS FOR THE 127-CM SPACING BETWEEN 969 PIN ASSEMBLIES

Fig. 4.13.1. Hornyak button radial transverses beyond GCFR grid-plate shield design configuration.

shown) indicates some slight streaming in the voids between the subassemblies as shown in the top curve. The bottom curve shows the symmetric flux pattern behind the three holes in the prototypic grid plate. Removing the B+C sleeve inserts from the holes in the grid plate increases the emergent flux by about 20%.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

Operations Division.

1. See Paper 4.12, this report.

4.14. SHIELDING EXPERIMENTAL METHODS*

F. C. Maienschein F. J. Muckenthaler

Abstract of paper presented at ANS Meeting, Madison, Wis., November 17-21, 1977

Benchmark, parametric, and design-confirmation shielding measurements are made at the Tower Shielding Facility (TSF) in Oak Ridge. A powerful reactor (to 1 MW thermal) is used with spectral modifiers to provide neutron spectra close to those associated with the various parts of an LMFBR. The source strength allows measurements through shields of full reactor thickness using sensitive detectors. The detectors include spectrometers and dosimeters for both neutrons and gamma rays. A large exclusion area provides much flexibility in arranging shield assemblies to be studied.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

4.15. U.S. INTEGRAL AND BENCHMARK EXPERIMENTS*

F. C. Haenschein R. E. Maerker F. J. Muckenthaler

(Abstract of paper presented at FBR Shielding Seminar, Obninsk, USSR, November 13-17, 1978)

Verification of methods for analysis of radiation-transport (shielding) problems in Liquid-Metal Fast Breeder Reactors has required a series of experiments that can be classified as benchmark, parametric, or design-confirmation experiments. These experiments, performed at the Oak Ridge Tower Shielding Facility, have included measurements of neutron transport in bulk shields of sodium, steel, and Inconel and in configurations that simulate lower axial shields, pipe chases, and top-head shields. They have also included measurements of the effects of fuel stored within the reactor vessel and of gamma-ray energy deposition (heating). This paper consists of brief comments on these experiments, and also on a recent experiment in which neutron streaming problems in a Gas-Cooled Fast Breeder Reactor were studied. The need for additional experiments for a few areas of LMFBR shielding is also cited.

*Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

4.16. INVESTIGATION OF STEEL-SODIUM-IRON SHIELDS*

E. M. Oblow R. E. Maerker

(Abstract of paper presented at FBR Shielding Seminar, Obninsk, USSR, November 13-17, 1978)

An analysis of experimental data from 21 fast reactor shield configurations containing steel, sodium, and iron were made as part of a study of the upper axial shielding needs of the Clinch River Breeder Reactor. The measured data were analyzed using both one- and two-dimensional discrete ordinates transport codes and several cross-section libraries based on ENDF/B-IV data with group structures of 51 and 171 neutron groups. One-dimensional sensitivity studies using the 171-group library and ENDF/B-IV covariance files for sodium and iron data were used to determine the sensitivities of the measured data to multigroup cross sections and to estimate uncertainties in the calculated results. Results indicate that the standard 51-group design cross-section library could be expected to predict the measurements to within 30% over 12 decades of attenuation, although a few of the deepest penetration configurations showed disagreements as large as a factor of three. The sensitivity results revealed very high sensitivity of the measurements to total cross-section minima and cross sections from 5 to 10 MeV in sodium and iron in the deep-penetration configurations. As a result, large uncertainties in the calculated results arose from small uncertainties in the cross-section data. These results indicate the need for better measurements of the total cross-section minima in sodium, especially around 300 keV.

*Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

5. DEVELOPMENT OF METHODS FOR SHIELD AND REACTOR ANALYSIS

5.0. INTRODUCTION

Methods for Shielding Analyses

The Division is continuing in its role of maintaining and improving radiation transport codes based on the discrete ordinates and Monte Carlo methods. Version IV of the discrete ordinates code DOT has now almost entirely replaced DOT-III both at ORNL and among the major LMFBR contractors. DOT-IV has the capability for solving large problem meshes with less fast memory and has several other advantages described in papers given below. For use with DOT-IV and earlier DOT codes, we also now have available a thoroughly documented set of directional quadrature data, as well as new programs for plotting DOT output. Channel-theory techniques that utilize forward and adjoint discrete ordinates fluxes to calculate "contribution" fluxes are also being extended. One extension that uses 2-D DOT fluxes and is known as "effectiveness theory" will provide estimates, based on sensitivity theory predictions, of the effect of placing various shielding materials at each point in the system. Another extension uses 1-D ANISN code fluxes to calculate space-energy fluxes. Other discrete ordinates work is directed toward developing a hybrid 1-D code that will use point and group cross sections in combination.

Our Monte Carlo work has included the generation of a set of doubly differential albedos for four materials of interest in shielding (51 neutron groups, 25 gamma-ray groups). The programs required to utilize the albedos with the MORSE code are available from the Code Center of the Radiation Shielding Information Center.

Methods for Reactor Analyses

Diffusion Theory. The VENTURE code, which applies the finite-difference diffusion theory approximation to neutron transport to 1-D, 2-D, and 3-D multigroup problems, is now routinely used locally for reactor analysis and it is available to other installations for their use locally or at ORNL via remote terminal. This code is a component of a modular code system that allows flexible calculational paths through the modules including global looping. Modules now in the system admit exposure calculations (see below), control rod positioning, and elaborate perturbation calculations. Local applications include fast breeder reactor and thermal converter reactor analyses discussed elsewhere.

The VENTURE code has been subjected to extensive application testing and feedback of results into the implemented iteration procedures. Modifications have been made in the code to improve computer utilization and to produce more auxiliary information for a problem.

Exposure. The code BURNER has been developed to calculate the effects of exposure. Several schemes are incorporated in parallel to solve the chain equations. Enhanced capability includes the provision for a local fine-scale exposure treatment (to produce information about power peaking), the calculation of secondary heat sources (gamma with and energy distribution, etc.) and exposure data, treatment of a steady-state continuous fueling model, and a temperature correlation on the microscopic cross sections.

5.1. THE DOT-IV TWO-DIMENSIONAL DISCRETE ORDINATES TRANSPORT CODE WITH SPACE-DEPENDENT MESH AND QUADRATURE*

W. A. Rhoades D. B. Simpson R. L. Childs† W. W. Engle, Jr.

(Abstract of ORNL/TM-6529, to be published)

DOT IV is designed to allow very large problems to be solved on a wide range of computers and memory arrangements. New flexibility in both space-mesh and directional-quadrature specification is allowed. For example, the radial mesh in an R-Z problem can vary with axial position. The directional quadrature can vary with both space and energy group. Several features improve performance on both deep penetration and criticality problems. The program has been checked and used extensively on several types of computers.

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† Computer Sciences Division.

5.2. A NEW WEIGHTED-DIFFERENCE FORMULATION FOR DISCRETE ORDINATES CALCULATIONS*

W. A. Rhoades W. W. Engle, Jr.

(Reprint of Trans. Am. Nucl. Soc. 27, 776-777, 1977)

One of the shortcomings of the method of discrete ordinates has been a tendency to generate negative estimates of inherently positive fluxes due to overextrapolation.¹ The complexity of practical transport problems is such that mesh refinement is only a partial remedy. This paper shows a new formulation of an old approach which ensures positive results while retaining satisfactory accuracy.

In the two-dimension a (2-D) discrete ordinates transport equation,

$$-a(\phi_{i+1/2}^{n+1} - \phi_{i-1/2}^{n+1}) + b(\phi_{i+1/2}^{n+1} - \phi_{i-1/2}^{n+1}) + (y_{m+1} \phi_{m+1}^{n+1} - y_m \phi_m^{n+1}) + \sigma \phi V = S V, \quad (1)$$

three of the ϕ 's are known from boundary conditions, but assumptions must be made describing the variation of flux across the mesh cell to eliminate three more unknowns. The weighted-difference method for solving Eq. (1) uses three parameters, a , b , and c (ref 2)

$$\phi = \phi_{i+1/2} + (1-a)\phi_i = (1-b)\phi_{i+1/2} + (1-c)\phi_i = \phi_{i+1/2} + (1-a)\phi_m. \quad (2)$$

With these,

$$\phi_{i+1/2} = \frac{a\phi_i + \left(\frac{1-b}{1-a} + 1 + y_m\right)\phi_m + \left(\frac{1-c}{1-a}\phi_{i+1/2} + \phi_i\right)\phi_i + \left[\sigma V - (1-b)\left(\phi_i + \frac{y_m}{1-a} + \frac{\phi_{i+1/2}}{1-a}\right)\right]\phi_i}{\sigma V + \frac{y_m}{1-a} + a\frac{b}{1-a} + \frac{\phi_{i+1/2}}{1-a}}. \quad (3)$$

and ϕ_{i+1} and $\phi_{i+1/2}$ can be found similarly. When $a = b = c = 1/2$, Eq. (3) is equivalent to the diamond-difference model and assumes a linear variation in flux across the mesh cell. The diamond-difference model provides excellent accuracy for small mesh sizes, and values of a , b , and $c = 1/2$ are not considered. Unfortunately, the diamond-difference model generates negatives for mesh sizes which are not uncommon in real problems, even though the true result is positive if a is positive. With $a = b = c = 1$, the step model is obtained, equivalent to assuming $\phi = \phi_{i+1/2} + \phi_{i+1} = \phi_{m+1/2}$. This gives inherently positive fluxes, and values of a , b , and $c = 1$ are not considered.

For many years, a fixup strategy has been used in which the diamond-difference model is used unless one of the extrapolated fluxes is negative. In that case, all of the fluxes are recalculated using the step model.² Another variation involved setting the negative flux to 0 and recalculating the other values.³ Unfortunately, all fixup methods

can lead to spatial flux distortions and model-switching oscillation between iterations, which thwart convergence.

Positivity can also be ensured by adjusting β such that

$$S \phi_S + (w_1 \phi_{S1} + w_2 \phi_{S2}) \theta_S + [\gamma_S - (1 - \beta)(\gamma_S + 2w_1 + 2w_2)] \phi_S \geq 0, \quad (4)$$

where θ_S and θ_T are arbitrary multipliers bounded by 0 and 1.

A solution could be found with $\theta_S = \theta_T = 1$, but this can lead to degeneracy in certain cases, producing numbers near 0 which thwart convergence once again. The DOT III code used $\theta_S = 1$ and $\theta_T = 0$. This provided good results for deep-penetration problems, but failed for criticality calculations due to excessive leakage. Consideration of a difficult reactor cell problem reveals why this happens. Cell problems typically have tall, narrow space cells. In zones having no source, Eq.(4) requires use of the step model along the height of the cell, even though flow across the vertical boundaries would control the actual result of Eq. (3) and produce a satisfactory value.

The DOT IV code includes a θ -weighted option in which θ_S and θ_T are arbitrarily set to 0.9. This allows the accurate linear model to be used when it is adequate and gives a smooth transition to the step model when the linear model actually breaks down. Either the DOE III or θ -weighted method costs roughly 15% more per iteration than the linear model with step fixup, but smooth, rapid convergence is found for problems that would otherwise oscillate. Table 5.2.1 shows the convergence of a cell problem using the methods described, as well as one where the θ -weighted model is used as a fixup to the linear model. The requirement for weighted difference in problems of this class is apparent.

Table 5.2.1. Flux convergence reached using various numerical models^a

Outer iterations	Linear/step	Linear/weighted	DOT III weighted	θ -weighted
1	1.2	1.7	0.3	0.5
2	0.6	0.6	0.1	0.2
3	1.2	1.2	0.7	0.05
4	0.5	0.5	0.03	0.04
5	0.2	0.4	0.03	0.01
6	0.3	0.09	0.01	0.01
7	0.3	0.3	0.006	0.006

^aConvergence for the m'th inner iteration is defined as

$$\frac{\phi_m - \phi_{m-1}}{\phi_m}$$

Table 5.2.2 shows that the θ -weighted model compares favorably with the fixup models in matching the Monte Carlo result, which is assumed accurate. The new model, then, gives the desired smooth convergence without compromising accuracy.

Table 5.2.2. Values of K_{eff} for a difficult cell problem using various numerical methods

Example	Code	Energy groups	Method	K_{eff}
1	DOT IV	11	Linear with step fixup	0.996
2	DOT IV	11	Linear with weighted fixup	0.996
3	DOT IV	11	DOT III style weighted difference	0.930
4	DOT IV	11	θ -weighted difference	0.991
5	DOT IV	51	Linear with weighted fixup	0.998
6	MORSE	51	Multigroup Monte Carlo	0.988 ± 0.015

Note (1): K_{eff} was converged to ±0.001 in the DOT cases.

(2): If 0.002 is taken as an "11-group library correction," based on example 5 vs example 2, the θ -weighted result for 51 groups would be 0.993, compared with the MORSE result 0.988 ± 0.015.

*Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

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5.3. COMPARISON OF REBALANCE STABILIZATION METHODS FOR TWO-DIMENSIONAL TRANSPORT CALCULATIONS*

W. A. Rhoades R. L. Childs[†] W. W. Engle, Jr.

(*Journal of Thermal Anal. Appl. Phys.*, 10, 407-424, 1975)

The use of a space-dependent rebalance method¹ is essential to the economical solution of deep-penetration problems. Reed² showed that the effectiveness of this method can be greatly improved, in one dimension (1-D), by the introduction of fictitious boundary flows. The application of this principle to two dimensions has resulted in various interpretations, however. This paper discusses two interpretations in common use, and the results of numerical comparisons are shown.

With the 1-D plane-geometry transport equation cast into the form

$$\frac{1}{2} (\phi_{i+1/2}^+ + \phi_{i-1/2}^-) + (\sigma_i + \sigma_i') \phi_i = \sigma_i \phi_i + \sigma_i' \phi_i^{k-1} \quad (1)$$

$$\phi_i = \frac{1}{2} (\phi_{i+1/2}^+ + \phi_{i-1/2}^-) \quad (2)$$

an inconsistency is evident, since the removal by σ_i is never equal to the source due to σ_i' until convergence is reached. The equations can be summed over direction and written in terms of boundary flows

$$-[\phi_{i+1/2}^+ + \phi_{i-1/2}^- - \phi_{i+1/2}^- - \phi_{i-1/2}^+] + (\sigma_i + \sigma_i') \phi_i = \sigma_i \phi_i + \sigma_i' \phi_i^{k-1} \quad (3)$$

where the ϕ 's are partial boundary flows leftward or rightward. The self-scatter inconsistency is solved by assigning a rebalance factor, β_i , to each interval

$$-[\phi_{i+1/2}^+ \beta_{i+1/2} + \phi_{i-1/2}^- \beta_{i-1/2}] + [\phi_{i+1/2}^- + \phi_{i-1/2}^+ + \sigma_i \phi_i] \beta_i = \sigma_i' \phi_i^{k-1} \quad (4)$$

Reed assures stability by adding to each β a fictitious current

$$\beta_{i+1/2} = \frac{\beta}{4} \max(\phi_i, \phi_{i+1}) \quad (5)$$

where β is an arbitrary multiplier on the order of 1.

The developers of the DOT 3.5 code observed that $\beta/4$ is approximately the one-sided diffusion-theory boundary flow, and $\beta/4$ is an interval optical thickness, usually on the order of 1. Then, using a user-adjustable parameter Z , the "J-method" generalization to two dimensions is, where $\uparrow$ and $\downarrow$ refer to upward and downward,

$$i_{z+1/2} = Z \max(i_{z+1/2}^S, i_{z-1/2}^S) \quad (6)$$

$$G_{j+1/2} = Z \max(G_{j+1/2}^S, G_{j-1/2}^S) \quad (7)$$

We can also observe, from Eq. (4), that the coefficient of i_z is positive when

$$i_{z+1/2} > \frac{1}{2} \max(\sigma_{z+1/2}^S, \sigma_{z+1/2}^K) \quad (8)$$

and that this is equivalent to Reed's criterion with $\gamma = 2$. Extending this analogue to 2-D,

$$G_{z+1/2} = \frac{Z}{4} \max(\sigma_{z+1/2}^S, \sigma_{z+1/2}^K) \quad (9)$$

$$G_{j+1/2} = \frac{Z}{4} \max(\sigma_{j+1/2}^S, \sigma_{j+1/2}^K) \quad (10)$$

$$Z \approx 1. \quad (11)$$

This method, called the " ϕ -method," is available in DOT IV, together with the J-method. A very similar method is used in the SIMMER code.³

Recent work by Miller⁴ and Alcouffe⁵ allow, in effect, a directionally dependent rebalance. These methods, however, are not available for routine use, as are the methods examined above.

The problems of Table 5.3.1 represent two major areas of application. With the optimum Z , the two methods gave comparable results on P1. The J-method solution was

Table 5.3.1. Results of comparing the J-method with the ϕ -method on two realistic problems

Problem description	Method	Z	Number of flux iterations	Flux convergence	Fission convergence
(P1) Pipe duct streaming with boundary source. Variable space and direction mesh. 28 x 35 maximum space mesh. 1 energy group of a 14-group P ₃ cross-section set is solved. Flux convergence of 1E-2 is sought.	J	4	9	7E-3	
		2	8	8E-3	
		1	8	3E-3	
		0.5	7	7E-3 ^a	
	ϕ or J	0.25	7	9E-3	
		0	15	2E-2	
		0.25	7	9E-3	
		0.5	7	7E-3 ^a	
		1	7	9E-3	
		2	8	6E-3	
		4	8	8E-3	
(P2) Fast critical assembly, 4-group P ₃ cross sections. A K calculation with 20 source iterations and 4 flux iterations per source iteration is solved. The space mesh is 35 x 30. A low value of fission convergence is sought.	J	4	80		5E-5
		2			2E-5
		1			8E-6
		0.5			1E-4
	ϕ or J	0.25			2E-2
		0			1E-2
		0.25			8E-3
		0.5			1E-4
		1			2E-5
		2			3E-5
		4			4E-5

^aOptimum value of Z .

^bIteration limit reached.

somewhat superior on P2. The hypothetical problem of Table 5.3.2 shows that either method can solve the low-dominance-ratio problem. The J-method is clearly superior for high dominance, where the rebalance is most important.

In conclusion, both models are based on plausible generalizations of the same basic research. Both can bring stable convergence to difficult problems, but the J-method showed a clear advantage in the case of high dominance ratio.

Table 5.3.2. Number of iterations to converge a hypothetical test problem

(P₂) classical source-in-a-box problem, 8 × 8 space mesh, uniform source in one quadrant with dominance = 0.5 in source region. Flux convergence criterion = 1E-4. Total cross section = 1.

Dominance ratio outside source region	Δx (mfp)	Δy (mfp)	Method	0	0.125	0.25	0.5	1	2	4
0.5	1	1	J	10	10	9	8	8	9	10
0.5	1	1	c	10	10	9	9	8	8	8
0.5	0.25	4	J	8	8	7	7	8	10	12
0.5	0.25	4	c	8	8	8	8	8	10	12
0.99	1	1	J	39	23	18	13	10	13	16
0.99	1	1	c	39	25	19	15	11	12	16
0.99	0.25	4	J	12	12	12	10	12	17	24
0.99	0.25	4	c	12	12	13	13	19	27	35

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Nuclear Research and Applications.

Computer Sciences Division.

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5.4. A COMPARISON OF CODES FOR DEEP PENETRATION TRANSPORT STUDIES*

W. A. Rhoades E. T. Tomlinson*

Abstract of paper to be presented at American Nuclear Society Symposium on Computer Codes for Neutron Transport, "Computational Methods in Nuclear Engineering," Williamsburg, Virginia, April 22-26, 1978

The authors undertook an investigation of the performance of several computer codes which are intended for neutron transport studies. The emphasis was on deep-penetration problems featuring geometries representative of realistic shielding problems. The TWOTRAN II and DOT IV codes were obtained from files available to the Radiation Shielding Information Center. TWOTRAN DA could not be obtained from LASL, and ANL stated that the transport feature in DIF3D was not suitable for such a study.

The codes were tested on an IBM 360/195 at Oak Ridge, and on a DC CYBER 76 at Idaho Falls. The convergence algorithm in TWOTRAN was changed to the DOT algorithm in order to provide a consistent comparison. Both codes were accompanied by extensive documentation. The TWOTRAN document was thorough in its theory and cataloging of variables. By comparison, the DOT document emphasized programming, application techniques, and adaptation to various computers.

The basic differences between the codes are evident from the documents. While the TWOTRAN provides a single, standard approach to problems, DOT is oriented toward biased quadrature sets, very large problems with anisotropic scattering, 200 or more directions, and intricate geometry. DOT encourages the user to choose among various numerical models and acceleration options based upon experience with a particular class of problems (Table 5.4.1).

Table 5.4.1. Comparison of features

TWOTRAN	DOT
Advantages: <ul style="list-style-type: none"> • Well documented, standard approach • Relatively uncomplicated input • Rapid execution 	Advantages: <ul style="list-style-type: none"> • Runs very large problems with little memory required • Alternate solution modes allow optimizing for difficult problems • Allows biased or multiple direction sets and irregular mesh • Rebalance mesh is arbitrarily variable • Detailed monitoring and control of convergence process is available • Allows variable space mesh
Disadvantages: <ul style="list-style-type: none"> • Relatively little iteration monitoring • Unable to solve large problems • Unable to use biased or multiple direction sets • Rebalance mesh linked to geometric description • More iterations required on some problems • Unable to use variable space mesh 	Disadvantages: <ul style="list-style-type: none"> • Relatively complex input data • Some interface files require re-formatting • Slower execution per iteration

The memory capacity of both codes was limited to about 355 000 words on the IBM computer. On the CDC machine, 40 000 words of fast memory were available to both codes. DOT was set up to use 131 000 words of slow memory, while TWOTRAN allowed 375 000. The memory requirement for several problems is shown in Table 5.4.2. TWOTRAN was able to solve only the smallest P3 problem, and was completely inadequate for realistic shielding problems. DOT was able to solve all but the largest P3 problem on either computer, while the large shielding problem fit easily onto the IBM machine.

Table 5.4.2. Comparison of memory requirements

Problem	Pipe streaming			Sodium steel		Lead compartment		Lead for shield	
Order of scattering	1	3	3	1	3	1	3	1	3
Quadrature sets	25 (3)	10 (25)	(10, 25)	10 (25)	10 (25)	40 (3)	40 (3)	10 (25)	10 (25)
Energy groups	14	14	14	21	21	14	14	14	14
Solution directions	44	44	(16, 44)	4	4	1	1	1	1
Number of unknowns	65x	65x	10x	10x	10x	10x	10x	10x	10x
Flux file size	40x	10x	10x	10x	10x	10x	10x	10x	10x
Blocks per space mesh (DOT only)	1	1	1	1	1	1	1	1	1
Fast memory requirements									
DOT	14x	10x	10x	10x	10x	10x	10x	10x	10x
TWOTRAN	20x	10x	[unstable]	20x	20x	10x	10x	10x	10x
Slow memory requirements									
DOT	10x	10x	10x	10x	10x	10x	10x	10x	10x
TWOTRAN	10x	10x	[unstable]	10x	10x	10x	10x	10x	10x

Number shows applies to DOT, TWOTRAN and compartment for lead for shield.

x = used to denote 1000, TWOTRAN uses somewhat fewer unknowns.

Variable space and directional meshes, etc.

brackets indicate that the requirement exceeded available fast memory.

Table 5.4.3 shows the results of the three smaller test problems. In each case, TWOTRAN executed faster for a given number of iterations. In every case, the more expensive DOT iterations provided better convergence. In some groups of some problems, more iterations would improve the convergence. Some groups were obviously "stuck," however, and would not improve without a more stable method.

Which code is better? TWOTRAN provides a well-documented, standard approach which executed faster than DOT. Users solving relatively uncomplicated problems of no more than P1 scattering or certain users of virtual-memory machines could use TWOTRAN effectively. Users who must solve large problems of P3 or greater scattering expansion, and/or problems with intricate geometry, will probably require the range of solution techniques, convergence controls, and storage flexibility available in DOT.

Table 5.4.3. Convergence and CPU time

Problem	Pipe chase		Sodium-steel		Head compartment	
	TWOTRAN	DOT	TWOTRAN	DOT	TWOTRAN	DOT
Number of outer iterations ^a	3	1	2	1	3	1
Number of inner iterations ^a	8	24	12	24	8	24
Maximum inners per group	24	24	24	24	24	24
IBM CPU time (min)	8.0	11.8	12.8	12.5	11.9	13.0
Total inners; all groups	336	326	502	332	320	208
CDC CPU time (min)	4.7	11.4	6.9	11.4	6.8	13.2
Total inners; all groups	336	328	472	312	321	232
Groups converged to:						
1 to 10 ⁻¹	4	0	0	0	0	0
10 ⁻¹ to 10 ⁻²	3	7	4	0	0	0
10 ⁻² to 10 ⁻³	2	6	4	3	5	3
10 ⁻³ to 0	0	1	13	18	9	11

^aThe number of outer iterations performed by TWOTRAN could not be controlled (a major flaw). Multiple runs were required to obtain a fixed maximum of 24 inners total.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

+ Computer Sciences Division.

5.5. CARP, A COMPUTER CODE AND ALBEDO DATA LIBRARY FOR USE BY BREESE, THE MORSE ALBEDO PACKAGE*

M. D. Emmett⁺ W. A. Rhoades

(Abstract of ORNL/TN-6604, October 1973)

The CARP computer code was written to allow processing of DOT angular flux tapes to produce albedo data for use in the MORSE computer code. An albedo data library has been produced containing several materials.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

+ Computer Sciences Division.

5.6. FORM: PERSPECTIVE PLOTTING CODE FOR ONE- AND TWO-DIMENSIONAL FLUXES*

D. T. Ingersoll

(Abstract of ORNL/TM-6396, May 1976)

FORM (Fluxes Over Region Map) is a perspective plotting code for displaying two-dimensional fluxes projected above the corresponding region boundary map. The code uses the extensive and versatile DISPSA graphics package available at ORNL to provide a number of useful features including the plotting of R-0 fluxes. FORM uses FIDO input format and is designed to read flux tapes generated by the DOT 3.5 code. While providing a few unique features, FORM is intended as a limited alternative to the more extensive GOFRR plotting code.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

5.7. GOFRR: A COMPUTER CODE TO GENERATE GRAPHICAL OUTPUT OF DOT AND ANISN FLUXES AND REACTION RATES*

F. B. Sadler[†] D. L. Selby

(Abstract of ORNL/TM-5963, March 1979)

A computer code identified as GOFRR (for Graphical Output of Fluxes and Reaction Rates) has been developed to calculate and plot reaction rates of fluxes from the discrete ordinates particle transport codes DOT (2-D) and ANISN (1-D). This report discusses the general computer requirements and job control language necessary for running the code. In addition, the data input are described and discussed and a series of sample problems is given.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

[†]Computer Sciences Division.

5.8. THE GENERATION OF A COMPUTER LIBRARY FOR DISCRETE ORDINATES QUADRATURE SETS: PART I: FULLY SYMMETRIC QUADRATURES AND THE DOQDP COMPUTER CODE. PART II: COMMON QUADRATURES AND THE QUADRATURE LIBRARY*

J. P. Jena[†] P. J. Erickson[†] W. A. Rhoades
D. B. Simpson M. L. Williams

(Abstract of ORNL/TM-6023, October 1977)

Although various discrete ordinates quadrature sets are in general use, there exists a need for a standard collection supported by documentation of their origin and derivation. This report attempts to provide this documentation for most of the commonly used sets. Instructions for using DOQDP -- the computer code used to generate these quadratures -- is provided. A library (in standard interface format) containing many of the quadratures has been generated and is documented in this report. Finally, a listing of the fully symmetric, half-symmetric, and several biased sets is provided by XY, RZ, and R₀ geometries.

Part I of the report describes the DOQDP code and the generation of S2-S16 fully symmetric quadratures. Part II describes the generation of S4-S10 half-symmetric and several biased sets, as well as a discussion of the quadrature library.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

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5.9. THE FANG ANGULAR FOLDING CODE FOR CHANNEL THEORY ANALYSIS*

M. L. Williams F. B. Sadler†

(Abstract of ORNL/TM-5228, August 1977)

The development of the FANG code for performing channel analysis is documented. An input description is provided for users, as well as a brief description of the basic channel theory concepts. A sample streaming problem is described, for which FANG input and output are given.

*Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

†Computer Sciences Division.

5.10. SOLID ANGLE SUBTENDED AT A POINT BY A CYLINDER*

S. N. Cramer

(Reprint of Trans. Am. Nucl. Soc. 28, 263-264, 1978)

For the calculation of radiation transport in models of many physical systems, it is often desirable to have a method for evaluating the solid angle of a cylinder. The equations presented here give the solid angle subtended by the sides and end of a cylinder relative to an arbitrary point in space. These equations include complete and incomplete elliptic integrals of the first and third kind, and the evaluation of these integrals is discussed. Also, it is shown how these equations can be applied to Monte Carlo estimation methods.

Consider a right circular cylinder of height $z = L$ and radius $r = R$. Let the z axis be the axis of the cylinder with $z = 0$ at the base. The solid angle subtended by this cylinder relative to an external point at r and z can be conveniently given in terms of the following expression:

$$\Omega(r, z) = \frac{2\pi}{[z^2 + (r + R)^2]^{1/2}} \left\{ \left(\frac{r - R}{r + R} \right) \Pi\left(\frac{r - R}{r + R}, k\right) - F(\phi, k) \right\}, \quad (1)$$

where F and Π are elliptic integrals of the first and third kinds, respectively, as defined in ref. 1. When $\phi = (\pi/2)$, F and Π are complete integrals; otherwise, they are incomplete. The modulus and parameter are

$$k = \left[\frac{4rR}{z^2 + (r + R)^2} \right]^{1/2} \quad \text{and} \quad a^2 = \frac{4rR}{(r + R)^2}. \quad (2)$$

For the disk end of the cylinder, when the sides are not exposed to the point in question, the solid angle is

$$\Omega_{\text{disk}}(z = L, r < R) = 2\pi + \Pi\left(\frac{\pi}{2}, a\right). \quad (3)$$

Continuing with the disk end only, for $r = R$,

$$\Omega_{\text{disk}}(z = L, r = R) = \pi + \Pi\left(\frac{\pi}{2}, a\right), \quad (4)$$

and for $r > R$, the solid angle of the disk could be written as

$$\Omega_{\text{disk}}(z = L, r > R) = \pi + \Pi\left(\frac{\pi}{2}, a\right). \quad (5)$$

However, before an actual computation of Eq. (3) is attempted, the π term in Eq. (1) must be removed since $\pi[(z/2), 1, k]$ is not defined.

Turning to the cylinder side, the solid angle relative to a point not exposed to either disk end is

$$\Omega_c(0 \leq z \leq L, r > R) = \Omega\left(\frac{\pi}{2}, z\right) - \Omega(\phi_1, z) + \Omega\left(\frac{\pi}{2}, L - z\right) - \Omega(\phi_1, L - z), \quad (6)$$

where

$$\phi_1 = \sin^{-1}\left(\frac{r + R}{2z}\right)^{1/2}. \quad (7)$$

Finally, for a point exposed to both an end and the sides, one has

$$\Omega_c(z > L, r > R) = \Omega\left(\frac{\pi}{2}, z\right) - \Omega(\phi_1, z) + \Omega(\phi_1, z - L). \quad (8)$$

By altering the method of integration in deriving the preceding equations or by using the appropriate identities and addition formulas in ref. 1, many and possibly more efficient computational forms of these equations can be obtained. Various expressions for the solid angle of a circular disk equivalent in some form to Eqs. (3), (4), and (5) have been given by many authors (see bibliographies in refs. 2 and 3). Although the development of the equations for the cylinder side follow directly from that for a circular disk, no expressions equivalent to Eqs. (6) and (8) have been found in the literature. If such equations have been given previously, their use has probably been restricted due to the inability, until the last few years, to obtain efficient evaluations of the incomplete elliptic integral of the third kind. A routine to evaluate this expression, based on an algorithm by Bulirsch,⁴ is available from the University of California at San Diego Computer Center. This routine has been converted for use on the IBM 360/91 machine at ORNL and has a maximum computation time of 100 μ s. The other elliptic integral routines for the preceding equations are available from most computer center libraries and have even faster computation times. Evaluation of these solid-angle equations yields results which agree with the tabulations in ref. 2 to within the numerical accuracy of the computer.

An example of a special adaptation of the solid-angle equations is the use of a disk boundary crossing scheme as an alternative to the Monte Carlo flux-at-a-point estimator. The procedure is to estimate the flux as a random estimation trajectory within Ω crosses a disk. This disk of radius R is fixed in the system under investigation about the point in question. The estimator will include Ω/ω , where ω is the cosine of the angle of incidence of the trajectory onto the disk, and Ω is from Eqs. (3), (4), or (5). To eliminate the grazing angle problem for $r \leq R$, R can be made arbitrarily small (within computational limits) so the combined probability of both $r \leq R$ and $\omega \approx 0$ is very small. For $r > R$, Ω/ω is always finite and it can be shown that for estimation from a point at r and z to a disk at $z = L$,

$$\lim_{z \rightarrow L} \frac{\Omega_c(z > L, r > R)}{z} = 2\pi \left\{ \frac{\Omega\left(\frac{\pi}{2}, z\right)}{(r - R)} - \frac{\Omega\left(\frac{\pi}{2}, z\right)}{(r + R)} \right\}, \quad (9)$$

where

$$\Omega = \frac{E(k) - E(k')}{[(z - L)^2 + r^2]^{1/2}} \quad (10)$$

and E is the elliptic integral of the second kind.

The equations presented here for the solid angle of a cylinder are exact and completely general. This incorporation into standard calculational procedures is facilitated by the availability of efficient routines which evaluate the elliptic integrals. In addition to the example given, these equations are directly applicable to Monte Carlo estimation methods for including the effect of a cylindrical detector in the calculation of an experimental mockup.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

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5.11. APPLICATION OF THE FICTITIOUS SCATTERING RADIATION TRANSPORT MODEL FOR DEEP-PENETRATION MONTE CARLO CALCULATIONS*

S. N. Cramer

(Abstracts of ORNL/TM-4930, December 1977, and of *Trans. Am. Nucl. Soc.*, 45, 127-131, 1978)

The fictitious scattering radiation transport model, suitable for Monte Carlo applications in geometrically complex systems, has been extended for use in deep-penetration calculations by the development of an appropriate next-flight estimator. Mathematical derivations are given, and it is shown that the estimation scheme is actually a one-dimensional version of the general model. Sample problems are solved to illustrate the use of the next-flight estimator, its variance characteristics, and the time saving features of the model. Other items discussed are coupling techniques with standard methods, systems with large cross sections, and inclusion of the fictitious scattering model in multigroup cross-section structure.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

5.12. METHODS FOR U.S. SHIELDING CALCULATIONS -- APPLICATIONS TO FFTF AND CRBR DESIGNS*

W. Engle, Jr. F. R. Mynatt R. K. Disney*

(Abstracts of paper presented at FBR Shielding Conf., Chicago, U.S.A., November 14-15, 1975)

The primary components of the U.S. reactor shielding methodology consist of: (1) powerful computer code systems based on discrete ordinates or Monte Carlo radiation transport calculational methods; (2) a data base of neutron and gamma-ray interaction and gamma-ray-production cross sections used as input in the codes; (3) a capability for processing the cross sections into multigroup or point energy formats as required by the codes; (4) large-scale integral shielding experiments designed to test cross-section data or techniques utilized in the calculations; and (5) a "sensitivity" analysis capability that can identify the most important interactions in a transport calculation and assign uncertainties to the calculated result that are based on uncertainties in all of the input data. The required accuracy for the methodology is to within 5 to 10% for responses at locations near the core to within a factor of 2 for responses at distant locations. Under these criteria, the methodology has proved to be adequate for in-vessel LMFBR calculations

of neutron transport through deep sodium and thick iron and stainless steel shields, of neutron streaming through lower axial coolant channels and primary pipe chaseways, and of the effects of fuel stored within the reactor vessel. For ex-vessel LMFBF problems, the methodology requires considerable improvement, the areas of concern including neutron streaming through heating and ventilation ducts, through the cavity surrounding the reactor vessel, and through gaps around rotating plugs in the reactor head, as well as gamma-ray streaming through plant shield penetrations. In these types of streaming problems, complex geometrical features of the system play an important role and necessitate the development of special calculational techniques. Several such problems encountered in design calculations for the Fast Flux Test Facility revealed generic areas requiring improved techniques. In the meantime, work on the primary components of the methodology continues, with significant progress having been made in recent years in the particular areas of cross-section processing and multigroup library production, in sensitivity analysis methods, and in the execution of the radiation transport code systems.

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† Westinghouse Advanced Reactors Division.

5.13. ANALYTICAL CALCULATIONS OF NEUTRON SLOWING DOWN AND TRANSPORT IN THE CONSTANT-CROSS-SECTION PROBLEM*

D. G. Cacuci

Abstract of DOE-NE, April 1984, vol. 1, no. 1, pp. 1-10. This is a preliminary report, DOE-NE-84-1, May 1984.

Some aspects of the problem of neutron slowing down and transport in an infinite medium consisting of a single nuclide that scatters elastically and isotropically and has energy-independent cross sections have been investigated. The method of singular eigenfunctions has been applied to the Boltzmann Equation governing the Laplace transform (with respect to the lethargy variable) of the neutron flux. A new sufficient condition for the convergence of the coefficients of the expansion of the scattering kernel in Legendre polynomials has been rigorously derived for this energy-dependent problem. Formulae have been obtained for the lethargy-dependent spatial moments of the scalar flux that are valid for medium to large lethargies. In deriving these formulae, use has been made of the well-known connection between the spatial moments of the Laplace-transformed scalar flux and the moments of the flux in the "eigenvalue space." The calculations have been greatly aided by the construction of a closed general expression for these "eigenvalue space" moments. Extensive use has also been made of the methods of combinatorial analysis and of computer evaluation, via FORMAC, of complicated sequences of manipulations. For the case of no absorption it has been possible to obtain for materials of any atomic weight explicit corrections to the age-theory formulae for the spatial moments of the scalar flux that are valid through terms of the order of u^{-1} . Higher order correction terms could be obtained at the expense of additional computer time. The evaluation of the coefficients of the powers of u , as explicit functions of the nuclear mass, represent one of the end products of this investigation. In addition, an exact expression for the second spatial moment, $M_2(u)$, valid for arbitrary (constant) absorption, has been derived. Combined with the previously known exact expression for the zeroth spatial moment, it is now possible to calculate analytically and rigorously the "age" for the constant-cross-section problem for arbitrary (constant) absorption and nuclear mass.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

5.14. TEST OF THE DIFFUSION THEORY DIFFERENCE EQUATIONS IN SLAB GEOMETRY (COMPUTER CODE VANCER)*

D. R. Vondy T. B. Fowler

(Abstract of ORNL-5881, June 1979)

The difference equations for representing the diffusion theory approximation to neutron transport depend on the method of formulation. A parameterized set of equations were implemented for slab geometry with mesh points located at material intersections. Such formulations as the mesh edge with four nearest neighbors on planes are included by the equations, as well as the higher coupling of eight nearest neighbors on planes for such formulations as linear finite element. The formulations of primary interest were applied to two dimensional problems, and to three dimensional problems with coupling to nearest neighbors on planes. Solutions were obtained to multigroup problems over a wide range of problem type. Selected results are reported and conclusions regarding the formulations.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

5.15. VENTURE: A CODE BLOCK FOR SOLVING MULTIGROUP NEUTRONICS PROBLEMS APPLYING THE FINITE-DIFFERENCE DIFFUSION-THEORY APPROXIMATION TO NEUTRON TRANSPORT, VERSION II*

D. R. Vondy T. B. Fowler G. W. Cunningham*

(Abstract of ORNL-6026, P. 1, December 1979)

This report documents the computer code block VENTURE designed to solve multigroup neutronics problems with application of the finite-difference diffusion-theory approximation to neutron transport (or alternatively simple P₁) in up to three-dimensional geometry. It uses and generates interface data files adopted in the cooperative effort sponsored by the Reactor Physics Branch of the Division of Reactor Research and Development of the Energy Research and Development Administration. Several different data handling procedures have been incorporated to provide considerable flexibility; it is possible to solve a wide variety of problems on a variety of computer configurations relatively efficiently. The programming in Fortran is straightforward, although data is transferred in blocks between auxiliary storage devices and main core, and direct access schemes are used. The size of problems which can be handled is essentially limited only by cost of calculation since the arrays are variably dimensioned and several data handling modes are programmed; the memory requirement is held down while data transfer during iteration is increased only as necessary with problem size.

The more common orthogonal coordinate systems arising in reactor analysis applications have been treated in from one through three dimensions. These include the slab, the cylinder, C-R, R-R-Z, and hexagonal and triangular coordinate systems in two and three dimensions. Only the mesh-centered finite difference formulation has been programmed. There is provision for the more common boundary conditions including the repeating boundary, 180-deg rotational symmetry, and the rotational symmetry conditions for the 90-deg slab and the 60- and 120-deg triangular grids on planes.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

* Computer Sciences Division.

5.16. ROOMOD - A CODE FOR CONTROL ROD POSITIONING*

D. R. Vondy T. B. Fowler

Abstract of ORO-6002, November 1969

This report documents a computer code which has been implemented to position control rods according to a prescribed schedule during the calculation of a reactor history. Control rods may be represented explicitly with or without internal black absorber conditions in selected energy groups, or fractional insertion may be done, or both, in a problem. There is provision for control rod follower, movement of materials through a series of zones in a closed loop, and shutdown rod insertion and subsequent removal to allow the reactor history calculation to be continued. This code is incorporated in the system containing the VENTURE diffusion theory neutronics and the BURNER exposure codes for routine use. The implemented automated procedures cause the prescribed control rod insertion schedule to be applied without the access of additional user input data during the calculation of a reactor operating history.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

5.17. EXPLOITING THE STEADY STATE, CONTINUOUS FUELING REACTOR MODEL*

D. R. Vondy G. W. Cunningham T. B. Fowler

Abstract of ORO-6002, November 1969
 Abstract of ORO-6002, November 1969
 Abstract of ORO-6002, November 1969

Calculational procedures were developed and implemented and are used routinely to solve the steady state, continuous fueling reactor model problem. The global iteration procedure involves successive neutronics, exposure calculations to effect the steady state. The fuel composition is adjusted as the flux distribution and reactor contents move toward a solution for a critical condition. The development was done to support the analysis of the gas cooled pebble bed reactor concept, which has continuous fueling, but many other applications are likely.

The exposure calculation done in the BURNER code is a simple extension of the application of the same equations used to treat fixed fuel. For the latter we determined the nuclide concentrations after a period of exposure to a region averaged neutron flux spectrum.

$$\boxed{\bullet} \rightarrow \phi(t) = \phi(t + \Delta t)$$

For flow of material, the composition leaving a region is determined for exposure during the residence time in the region, and this composition is the feed for the next zone along the path. A simple estimate of the average region concentrations is returned to the neutronics code.

$$\begin{array}{c} \phi(t) \\ \downarrow \\ \boxed{\bullet} \rightarrow \bar{\phi} = \frac{1}{2} [\phi(t) + \phi(t + \Delta t)] \\ \downarrow \\ \phi(t + \Delta t) \end{array}$$

With this model, the flow of pebbles is followed at a constant velocity.

Along each flow path and dependent only on the local residence time, not directly on volume fractions,

$$\dot{X}(t) = A X(t) \quad (1)$$

where t refers to the residence time, and symbolically $A = A(x, y, z, E, \dots)$, where the functional dependence of the operator matrix A is shown on the cross sections (nuclide and possibly location dependent) and the space, energy flux ϕ . The neutronics code determines, symbolically

$$X(t + \Delta t) = \exp[A \Delta t] X(t) \quad (2)$$

for the geometric description of the problem. Given an estimate for λ , the nuclide concentrations are solved by Eq. (1). The nuclide concentrations in the reactor depend only on λ and on the feed concentrations. Adjusting the feed moves the iterate approximation toward a solution for the problem.

The direct criticality search procedure is used when the neutronics problem is solved to adjust the fissile concentration on each iteration to effect a critical reactor. The feed concentrations are also changed as if located in the reactor.

Use of the VENTURE finite-difference diffusion theory code³ allows the scheme to be used for one-, two-, or three-dimensional problems. The implemented procedures allow material which moves and other material which is fixed (as in blankets), any number of streams along each flow path through the regions with different residence times by stream, counter flow, packing fraction changes in discrete steps, and mixing of streams (as of fertile pebbles from an inert blanket mixing with fueled pebbles introduced below the blanket), within a few limitations.

Application experience will be discussed with attention to methods found to accelerate the iteration process and stabilize the tendency of the flux to oscillate wildly in certain multi-dimensional problems. Typical results of calculations are shown in Table 5.17.1. With the assumptions made, these results are for a point in the operating history after steady state has been effected with continuous feed (and discharge) at a fixed composition.

Table 5.17.1. Comparison of results for three reactor types obtained with four groups and thirteen groups
(1,200 MWe, blanketed, one-dimensional steady state throwaway)

Neutron groups	Burner reactor C/HM = 250		Prebreeder C/HM = 175		Breeder C/HM = 115	
	4	13	4	13	4	13
Conversion ratio	0.637	0.635	0.700	0.698	1.000	0.996
Peak axial power density (W_{th}/cm^2)	13.1	12.9	7.31	7.64	6.63	6.66
Fissile inventory (kg)	1,493	1,508	2,212	2,201	2,882	2,816
Core leakage fraction	0.0594	0.0604	0.0398	0.0411	0.0184	0.0191
Fission product absorption fraction	0.1071	0.1069	0.0781	0.0782	0.0439	0.0445
Average exposure ($MW_{th}d/kg$ heavy metal)	101.0	100.9	33.0	33.0	25.4	25.5
Fissile feed (kg/d)	2.060	2.073	3.212	3.210	3.953	3.855

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

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2. D. R. Vondy and G. W. Cunningham, "Exposure Calculation Code Module BURNER," ORNL-5180 (to be published).
3. D. R. Vondy, T. B. Fowler, G. W. Cunningham, "VENTURE: A Code Block for Solving Multigroup Neutronics Problems Applying the Finite-Difference Diffusion-Theory Approximation to Neutron Transport, Version II" (1977).

5.18. RESTRAINED ACCELERATION IN ITERATION*

D. R. Vondy T. B. Fowler

(Abstract of Technical Note, Nucl. Sci. Eng. 68, 416-418, 1974)

A procedure is described that is used to restrain the iterate estimates of a dependent variable (such as the local neutron flux) when acceleration is applied. This scheme allows negative values but prevents extrapolation from positive values to a negative result.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

5.19. THE CODE PERTUBAT FOR PROCESSING NEUTRON DIFFUSION THEORY NEUTRONICS RESULTS FOR PERTURBATION ANALYSES*

D. R. Vondy T. B. Fowler

(Abstract of ORNL-EE76, March 1976)

This report documents a code which is used for perturbation analyses. The PERTUBAT code is a module in the local computation system which contains the VENTURE neutronics code for applying multigroup diffusion theory. First order effects are calculated for changes in macroscopic cross sections. These may be due to changes in nuclide densities and microscopic cross sections. A fractional change may be made in macroscopic cross sections or the effects of temperature changes may be accounted for directly given the temperature distributions and associated cross section data. Regular, adjoint flux integrals are made available for other use. These integrals and the perturbations may be obtained for selected conditions with multiple sets of the reference data files available in the run of the modular code system. Thus the regular flux may be generated for a reference state, the adjoint flux for a specific perturbed state, and the individual components to the net reactivity change may be obtained for the change between the two states in this computation system. Nuclide importance data may be obtained.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

5.20. REFERENCE TEST PROBLEMS FOR THE VENTURE NEUTRONICS AND RELATED COMPUTER CODES*

D. R. Vondy T. B. Fowler

(Abstract of ORNL/OR-487, August 1976)

Sixteen problems were selected which collectively use many of the options and different calculational paths implemented in the VENTURE diffusion theory neutronics computer program. The input data requirements and key results are presented. This set of problems can be used as an essential part of validation testing of an implementation of, or modification to, the VENTURE and related codes.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

5.21. TRANSPORT/DIFFUSION COMPARISONS FOR SMALL-CORE LMFBR DISRUPTIVE ACCIDENTS*

E. T. Tomlinson²

(Reprints of Issues are \$2.00. For a list of authors, see: Reprints and Orders at JHEI/ISS/TV-88, November 1975.

The determination of situations for which diffusion theory and first-order perturbation theory fail to predict the reactivity state of a disrupted LMFBR core is a prime concern of the reactor safety analyst. To assess the extent to which these methods provide accurate results, five "snapshot" disrupted core configurations similar to those predicted by the SAS3A accident analysis code, but somewhat exaggerated, were considered. These configurations were defined as follows:

- Case 1 - normal BOL 350 Mw(e) core, Fig. 5.21.1.
- Case 2 - initial sodium voiding phase of a transient undercooling (TUC) accident
- Case 3 - post-cladding motion/pre-fuel motion phase of a TUC
- Case 4 - post-fuel motion/pre-shutdown phase of a TUC
- Case 5 - post-fuel motion phase of a transient over-power (TOP) accident.

The reactivity state of each configuration was determined using a number of numerical techniques and a seven-energy-group cross-section library. Direct k_{eff} difference calculations were performed using discrete ordinates transport theory (S_2P_1 , S_4P_1 , S_8P_1 , S_NP_1), diffusion theory, and transport Monte Carlo. The discrete ordinates transport and diffusion calculations were performed with the DOT-4 code and the KENO code was used for the Monte Carlo calculations. Reactivity changes were also calculated using first-order and exact perturbation techniques using both transport diffusion theory fluxes. A hybrid P_1 perturbation technique was investigated in which a transport perturbation calculation was performed using scalar fluxes and moments from diffusion calculations.

Fig. 5.21.i. The calculational model for a 300-MWe LMFR core.

That is, the flux moments (currents) are folded with the change in the P₀ scattering cross sections instead of with the change in the diffusion coefficient as in diffusion theory perturbation.

The calculated reactivity change associated with each configuration calculated by the majority of aforementioned techniques are presented in Table 5.21.1. Diffusion theory consistently underpredicts the positiveness, compared to transport theory, of the disrupted configurations with the exception of configuration 4 in which voids are present. There also does not exist a single bias factor that can consistently be determined and applied to the diffusion theory results to obtain an equivalent transport result.

Table 5.21.1. Reactivity changes calculated using various methods for four disrupted configurations

$\rho_0(\%)$					
Configuration	Direct transport	Direct diffusion	First-order transport	First-order diffusion	First-order hybrid P_1
2	0.034	-0.256	0.193	-0.743	0.151
3	4.619	3.064	5.512	-1.587	5.549
4	7.287	7.393	9.150	1.458	9.139
5	-2.184	-2.416	-2.316	-2.439	-2.412

Similar results were found for the first-order diffusion perturbation calculations. The magnitude of the positive reactivity change was constantly underpredicted, missing the sign in some cases. First-order transport perturbation, on the other hand, consistently overpredicted the magnitude of the reactivity changes, but yielded results that were superior to the diffusion theory results.

The hybrid P_1 perturbation technique⁴ yielded results that were equivalent to first-order transport perturbation in accuracy with the computing expense of diffusion theory. The improvement in results obtained through the use of the hybrid method was attributed to the implicit manner that is used in the transport perturbation techniques to calculate the leakage component of the reactivity change. Similar effects were seen in the recent work by Cahalan⁵ at Argonne National Laboratory.

The main conclusion of this study was that diffusion theory, first-order diffusion perturbation theory, and low-order S_P transport were not valid for the calculation of the reactivity state of severely disrupted LMFBR cores. There does exist, however, a method that yields first-order transport perturbation theory results, which may be of sufficient accuracy for some applications, with diffusion theory computational expenses. It should be noted that this study was meant to be bounding in nature and, as such, does not reflect the real accident situation.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

¹ Computer Sciences Division.

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2. W. A. Rhoades, Comments on the DOT3.5 and DOT4 Codes.
3. L. M. Petrie and N. F. Cross, "KENO-IV on Improved Monte Carlo Criticality Program," ORNL-4938 (1973).
4. R. L. Childs, Oak Ridge National Laboratory, Personal Communication (Apr. 1, 1977).
5. J. E. Cahalan, "On Interactive Diffusion Theory Method for the Incorporation of Transport Effects in Fast Reactor Safety Neutronics," ANL/RAS 76-17, Argonne National Laboratory (1976).

5.22. DEVELOPMENT OF DEPLETION PERTURBATION THEORY FOR COUPLED NEUTRON/NUCLIDE FIELDS*

M. L. Williams

(Abstract of ORNL/TM-5958, September 1974, and of paper to be published in Nuclear Science & Engineering)

A perturbation formulation is developed for the space-energy dependent burnup equations describing depletion and transmutation of nuclide densities in a coupled neutron-nuclide field, such as a reactor core. The formulation is developed in a form consistent with the computational methods used for depletion analysis. The analysis technique currently employed in most burnup calculations is first reviewed as a method for describing the nonlinear coupling between the flux and nuclide fields. It is shown that based on the present formulation three adjoint equations (for flux shape, flux normalization, and nuclide density) are required to account for the coupled variations arising from variations in initial conditions and nuclear data. The adjoint equations are derived in detail using a variational principle, and an algorithm is suggested for solving the coupled equations backward through time.

Perturbation expressions are used to define sensitivity coefficients for responses that depend on the coupled interaction between the neutron and nuclide fields. The relation between coupled and noncoupled sensitivity theory is illustrated. Finally, two analytic example problems are solved that determine the sensitivity of some final nuclide concentration to changes in initial conditions. Results obtained from direct calculation and from the coupled perturbation theory are compared.

*Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

5.23. THE COMPUTATION OF HOMOGENIZED GROUP PARAMETERS*

J. A. Worley A. F. Henry

(Reprints of Trans. Am. Nucl. Soc. 17, 641-644, 1977; similar paper presented at IAEA Technical Committee Meeting on Homogenization Methods in Reactor Physics, Vienna, Austria, November 12-16, 1976; also reported in MIT Report No. OR-1671-16, NINE-419, September, 1977)

To obtain the subassembly-homogenized few-group parameters needed when nodal or finite element methods are applied to reactor analysis, a two-stage homogenization process is carried out. First, equivalent diffusion theory parameters representing fuel cells, control rods, etc. are found. Then a second homogenization is performed over "nodes" consisting of (usually) symmetric arrangements of fuel cells, control elements, structure and burnable poison material. This is done by isolating the node in question and performing a group-diffusion-theory calculation based on a zero net current boundary condition and using (heterogeneous) group-diffusion parameters obtained from the first stage of homogenization.

This conventional flux-weighting procedure is a bit suspect - particularly with regard to the diffusion constants. We propose an alternative method which we feel comes closer to providing "ideal" homogenized nodal parameters capable of predicting the correct average reaction and leakage rates of the nodes.

If $\phi_n^{(g)}(r)$ and $\Sigma_{n,n}^{(g)}(r)$ are the group- g flux and interaction cross section for process- n node- n in the heterogeneous reactor and $\phi_n^{(g)}(r)$ and $\Sigma_{n,n}^{(g)}(r)$ are the corresponding quantities when node- n is homogenized, the "ideal" $\phi_n^{(g)}(r)$ obey

$$\frac{\int_V \phi_n^{(g)}(r) \Sigma_{n,n}^{(g)}(r) dV}{\int_V \phi_n^{(g)}(r) dV} = \Sigma_{n,n}^{(g)}(r) \quad (1)$$

where the integrals are over the volume V_n of node- n .

Rather than assume that

$$\int_{V_n} \phi(r) dV = \int_{V_n} \bar{\phi}_g(r) dV$$

(as is done for the flux-weighting method), we determine the right-hand side of Eq. 1 by using response matrices. Thus we define for each heterogeneous node response matrix elements $R_{gg'}^{ii'}$ (the group- g partial neutron current passing out of nodal face- i due to a flat, unit incoming current of group- g' neutrons across face- i') and $I_{gg'}^{ii'}(\alpha)$ (the interaction rate within the node for group- g neutrons undergoing interaction- α due to a flat unit incoming current of group- g' neutrons across face- i') so that the total partial current of group- g neutrons emerging from face- i of the node is given by

$$J_g^{(i)}(\text{out}) = \sum_{i', g'} R_{gg'}^{ii'} J_{g'}^{(i')}(\text{in}) \quad (2)$$

and the interaction- α due to group- g neutrons in the node is given by

$$\int_{V_n} \Sigma_{\alpha g}(r) \phi_g(r) dV = \sum_{i', g'} I_{gg'}^{ii'}(\alpha) J_{g'}^{(i')}(\text{in}). \quad (3)$$

To use Eq. 3 to determine the numerator of Eq. 1, we need to know the $J_{g'}^{(i')}(\text{in})$'s that the node will experience in its reactor environment. To find these we solve an eigenvalue problem for node- n using Eq. 2 along with further relationships between the $J_g^{(i)}(\text{out})$'s and the $J_g^{(i)}(\text{in})$'s determined by the albedos of the nodes adjacent to node- n . If Eq. 2 is written in matrix form as

$$\underline{J}(\text{out}) \equiv \{\text{Col } J_g^{(i)}(\text{out})\} = \underline{R} \underline{J}(\text{in}) \quad (4)$$

and the albedo relationships are written as

$$\underline{J}(\text{in}) = \underline{A} \underline{J}(\text{out}), \quad (5)$$

this eigenvalue equation is

$$\underline{I} \underline{J}(\text{in}) = \underline{A} \underline{R} \underline{J}(\text{in}), \quad (6)$$

where, if the reactor is critical and the albedos are correct, the eigenvalue γ will equal unity.

To estimate the elements of A without solving the full core problem, we apply Eq. 2 to each of the neighbors of node- n making the assumption that for all faces of those neighbors except the one adjacent to node- n , $J_g^{(i)}(\text{out}) = J_g^{(i)}(\text{in})$.

With Eq. 6 solved, we apply Eq. 3 to determine the numerator of Eq. 1. The denominator of Eq. 1 is found using the same $J_g^{(i)}(\text{in})$'s applied to the homogenized node. This may be done iteratively as improved values of the $\bar{\Sigma}_{\alpha g}$'s are found or, since the $J_g^{(i)}(\text{in})$'s and $J_g^{(i)}(\text{out})$'s yield values of the average group fluxes and their normal derivatives for the nodal surfaces, $\phi_g(r)$ (for the homogenized node) may be estimated by

XX	Reference percent total power*
XX	Percent error using FWC
XX	Percent error using EGDC

CL		
+		+
2.288	3.257	3.025
1.02	-4.57	2.83
-0.74	0.03	-1.18
3.426	3.393	4.424
-4.33	5.08	-3.64
0.15	-0.84	0.14
+		+
3.448	4.906	4.473
3.76	-2.15	6.29
-0.97	0.32	-0.64
4.238	5.164	5.593
-1.39	0.80	0.39
0.17	0.60	0.75
3.841	4.699	5.167
-1.03	0.25	0.36
-0.05	0.38	0.41

fitting to products of cubic Hermite polynomials. Finally, the β_i are found by requiring that the largest response matrices computed for the heterogeneous node equal the corresponding matrices computed using the homogenized parameters. This last step requires iteration.

The method requires little more computer time than the standard flux weighting procedure and reduces to the rigorous answer in one dimension.²

The figure shows results of an application to a two-dimensional light-water-moderated and -reflected reactor model composed of fuel assemblies containing either cross-shaped control rods (indicated by the small crosses in the upper left-hand corners) or cross-shaped water holes. A two-group, fine-mesh solution with the control rods and water holes represented explicitly was taken as standard. The same mesh structure was used to analyze the core described by homogenized parameters and to perform the assembly calculations yielding the standard flux-weighted (FWC) parameters and the response matrices needed for the present (EGDC) scheme. The assembly powers to the reference calculation are shown along with errors for the analogous quantities computed using homogenized parameters. The homogenized parameters obtained by the approach described above yield more consistently accurate power values.

	Reference	FWC Percent Error	CL EGDC Percent Error
K_{eff}	0.842530	-0.572	-0.197

* Reference case is a heterogeneous, fine-mesh calculation.

Fig. 5.23.1. BWR geometry β_i -core solution.

* Research sponsored by the U.S. Department of Energy.

† Massachusetts Institute of Technology.

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6. ANALYSES FOR SPECIFIC SYSTEMS OR APPLICATIONS

6.0. INTRODUCTION

The analyses performed by the Division for specific systems or applications continue to be many and varied, as would be expected by the diversified interests of the sponsoring agencies. In addition to the analyses of experiments discussed in Chapter 4, and to a lesser extent in Chapter 3, we have performed calculations for specific core and/or shield designs corresponding to several fission and fusion reactor concepts. This work is sponsored by the programs for the Liquid-Metal Fast Breeder Reactor, the Gas-Cooled Fast Breeder Reactor, the Pebble Bed Reactor, and the programs for various proposed fusion devices. We have also carried out studies of alternate nuclear fuel cycles and accelerator breeders and continued to respond to the requests for high-energy calculations tailored for particular accelerators. Other work has consisted of a continuation of theoretical studies of radiation detectors associated with these various reactor and accelerator systems and of weapons radiation transport calculations for the Defense Nuclear Agency.

Liquid-Metal Fast Breeder Reactor Program

The Division remains involved in several areas of research in the LMFR program. In the area of shielding, we are still performing calculations for the Fast Flux Test Facility as requested by the Hanford Engineering Development Laboratory during the construction phase, especially when deviations from the original design occur that could affect the radiation doses previously calculated. We are also continuing to examine designs for the Clinch River Breeder Reactor SLFM (Source Level Flux Monitor). In this effort we are attempting to predict the background count at the location of the SLFM due to fuel stored in the stored fuel transport location for various SLFM shadow shield designs and to calculate the absolute count rate at the SLFM position during full shutdown. Finally, we are addressing the effect of the CRBR Alternative Fuel Management System (heterogeneous core) on doses at the core barrel, head, etc.

In a CRBR supporting analysis program we are looking at the performance of the SLFM under various conditions, particularly during refueling periods. The intent is to combine the uncertainties associated with its use in the proposed applications with the uncertainties associated with our calculations and thereby to predict the maximum uncertainty on the SLFM reading. This work is being performed for the heterogeneous core. In a related effort we are planning a ZPPR experiment, together with the Instrumentation and Controls Division, to investigate the initial loading problems associated with the heterogeneous core.

We have also participated in the LMFR Iteration-Resistance Preconceptual Core Design Study (IRPCDS), along with a number of other organizations. Our effort has concentrated on examining low-density designs which might result from using the sphere-pow fuel fabrication technique under study at ORNL. Specific questions to be answered were associated with the effect of the separation of the fissile and fertile materials in the fuel pins and also the effect of the reduced density of the fuel.

In an LMFR safety program, we are comparing transport and diffusion methodologies in analyses of distorted heterogeneous LMFR cores such as might result from a core disruptive accident. We are also using generalized sensitivity techniques to study thermal-hydraulic problems in fast reactors (see Section 3.0), and we have explored the possibility of developing a transient quasi-static Monte Carlo code. Because of convergence difficulties, however, the decision was made that the Monte Carlo method probably was not suitable for space- and time-dependent kinetics problems. A study of the reactivity associated with the collapse of bubbles produced by the vaporization of stainless steel in a medium of molten fuel is also being continued. And finally, the safety program includes the CREO and SACRO data bases described in Chapter 7.

Gas-Cooled Reactor Programs

The Division is engaged in studies both for the Gas-Cooled Fast Breeder Reactor and the gas-cooled Pebble Bed Reactor. The GCFR shielding program which has been under way for some time is continuing with emphasis on verification of the methods used for designing the reactor's grid plate shield. This program involves confirmation experiments that are discussed in Chapter 4. We are also performing neutronic core analyses for the GCFR and are studying the use of alternate fuel cycles with the reactor, both with homogeneous and heterogeneous core designs. The fuels being considered are the classical Pu/U fuel, a denatured U fuel, and Pu/In fuel. Some preliminary studies of GCFRs operated in symbiosis with thermal reactors have also been performed.

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The purpose of Pebble Bed Reactor effort is to examine the design and mode of operation of the high-temperature, gas-cooled, graphite-moderated PBR concept and to project its performance. Ore utilization and fuel cost are of primary interest, with design constraints satisfied. This work is under a technical information exchange umbrella agreement with the Federal Republic of Germany. The concept could be used in electrical energy and process heat production, or a combination. Of special interest is the high-energy conversion efficiency possible, especially with a direct gas cycle. With continuous fueling, the concept should effect and show advantage of a higher fuel conversion ratio and availability than other concepts that require the use of control rods to offset the reactivity swing accompanying periodic refueling. An interim report was issued on the early assessment of breeding capability (see Paper 6.9), and a draft of the report documenting recent effort has had limited distribution (see Paper 6.10).

Performance calculations have been performed with emphasis on designs for the breeder or near-breeder using ^{235}U fuel, the prebreeder using ^{235}U fuel to produce ^{239}Pu fuel, the non-proliferation operation (throwaway) with medium or low-enriched uranium feed, and the burner (throwaway) or converter (recycle) with fully enriched uranium feed. Selected results (see Table 6.0.1) show the performance in some of the service roles under consideration. Not shown are the significant effects on performance of design variations and possible limitations and practical considerations, nor the gain possible by selective recycle of pebbles without reprocessing. These aspects are under study.

The effort on analysis of this concept has included the modification of the local computation capability to obtain solutions of problems specific to the continuously fueled core. A steady-state, continuous-fueling exposure model has been implemented that allows the VENTURE diffusion-theory neutronics code to be used for assessing the core performance at a point in time; procedures are used with the modular code system to determine the fuel feed enrichment required for a critical reactor. A point multibatch neutronics model is used to follow the history of a plant and generate mass balances and perform fuel cost calculations. Effort has also gone into the methods for processing cross section data, thermal hydraulics calculations, and fuel balance/cost symbiosis analysis.

Table 6.0.1. Comparative performance of the Pebble Bed Reactor concept

Fuel	Burner		Converter,	Prebreeder,	Breeder,
	20% ^{235}U	40% ^{235}U	40% ^{235}U	40% ^{235}U	40% ^{235}U
Carbon/heavy metal	450	250	250	175	100
Pebble residence time (full-power years)	2.5	4.2	4.2	1.5	1.5
Conversion ratio	0.56	0.56	0.63	0.70	1.02
Peak power (kW/pebble)	3.8	3.4	3.4	2.8	2.3
Fissile feed (kg/MWe-yr)	0.75	0.66	0.36	1.3	-
Plant fissile inventory (kg/MWe)	0.85	1.3	1.4	1.9	2.8
System fissile inventory (kg/MWe)	1.1	1.5	2.7	3.6	6.0
Fissile commitment (kg/24 MWe-yr)	19.0	12.0	12.0	35.0	6.0
Ore commitment (kg/24 MWe-yr)	4,383	3,922	2,768	6,675	-
Fuel cost* (mill/kWh-hr)	6.0	5.4	4.1	6.2	7.0

*Highly enriched unless specified otherwise. Preliminary.

Ore enrichment tails = 0.002. No credit for breeder feed production.

Alternate Fuel Cycle Program

The Division has been participating in a series of studies of the effects of using alternate fuels in various reactors, with our studies concentrating on fast reactors. In addition, we have during the past year assumed responsibility for the editing of a report on the denatured ^{235}U fuel cycle which is currently being published as OPR-4386.

Magnetic Fusion Energy Program

The Division's Magnetic Fusion Energy Program is divided into three areas: integral experiments; radiation damage calculations; and neutronics calculations in support of the design of different fusion energy devices. The integral experiments, reported on in Chapter 4, are designed to provide data against which ORNL and all other organizations involved in shielding calculations for fusion devices can test their methods and interaction data. The principal problem is the prediction of neutrons streaming through the huge ducts that are designed into fusion reactor blankets and shields for neutral particle injection into the plasma and for vacuum pumping, diagnostics, etc. In order to ensure that the experiments will be sensitive to the most important interactions, cross-section sensitivity calculations are performed by the Division's MFE analysis group to aid in the design of the experiments.

The radiation damage calculations are also carried out to aid experimental design. The ORNL Metals and Ceramics Division is attempting to develop experiments that can be performed at fission reactors to yield damage data for metals that will be exposed in fusion reactors. Our role is one of support: we perform the calculations they need as their designs progress. In addition, we have aided in the design of a Li(D,T) neutron radiation damage facility at Hanford Engineering Development Laboratory by providing them with calculated multigroup neutron-photon cross sections in the 30- to 40-MeV range.

We are also continuing to perform neutronics calculations for the various proposed fusion energy devices. At present our most important calculations are for the neutral beam injectors of the TFTR (the Tokamak Fusion Test Reactor), a reactor which will be built at Princeton University. Calculations are also performed for the TNS (The Next Step) device, and the EBTR (Elmo Bumpy Torus Reactor). In addition we have investigated the concept of a fusion-fission hybrid reactor to produce fissile fuel.

High Energy Physics Program

The Division's calculations of the transport of high-energy particles continue to be done on request from other groups. With the methods and interaction data we have developed and the experience we have gained over the years, we can now perform calculations routinely that were not possible a few years ago. During this reporting period we have carried out calculations to determine the shielding requirements for the proton storage rings at the 400-GeV proton accelerator (ISABELLE) at Brookhaven National Laboratory. We also performed calculations to aid in the design of the uranium target to be hit by 800-MeV protons at Argonne National Laboratory's IPNS (Intense Pulse Neutron Source) being built for solid state studies.

Our other work in the high-energy program continues to be directed at aiding in the design of instrumentation (ionization calorimeters) to measure the energy of various high-energy particles. The calorimeters that have been built and tested at accelerators have all been found to be satisfactory, which has encouraged the development of more sophisticated (and more complicated) devices, each design being for a different purpose. The agreement of our calculations with the experimental tests is especially gratifying since the instruments are very large and expensive, and "testing" design variations with a series of calculations rather than with numerous mockups is much more economical.

Accelerator Breeding Studies

Although it is not one of our major programs, studies of accelerator breeding concepts have continued within the Division. These concepts are based on the production of an intense neutron source by the bombardment of a target with high-energy particles, the neutrons being used to fission and/or breed fissile isotopes in a surrounding blanket of fissile-fertile material. The design calculations are performed in two parts to cover the high- and low-energy regions. The high-energy calculations carried out by ORNL, and also by other organizations working on accelerator breeder concepts, were possible because of our early experience in developing techniques (methods and cross sections) for high-energy accelerator shielding calculations. Similarly, the low-energy calculations have utilized discrete ordinates and Monte Carlo methods developed in earlier reactor shielding programs.

LIQUID-METAL FAST BREEDER REACTOR PROGRAM**6.1. VENTURE CODE RESULTS FOR THE LARGE-CORE
SODIUM-VOIDING BENCHMARK PROBLEMS***

D. R. Vondy T. B. Fowler

Notes: The following is an abstract of a paper which will be included in a forthcoming DOE report. It reports ORNL solutions to problems for a benchmark analysis of a homogeneous fast reactor in the DOE-sponsored large-core code evaluation effort. These results will be published along with those from other participating installations.

Results for the DOE Large-Core Sodium-Voiding Benchmark Problems were obtained with the VENTURE Diffusion Theory Neutronics Code using 21-group General Electric macroscopic cross sections and R-Z geometry. In the VENTURE Code meshpoints are located at the centers of equal volume elements. Tight convergence of 5×10^{-5} maximum relative flux change on outer iteration was satisfied, which should produce five-digit accuracy in integral properties. Core leakage was calculated as that from the core including the rod zone which extends into the blanket. Normalization was to one source neutron (unit loss). The case 'B' results were obtained with equal mesh spacing to supplement the other results (within short-word precision of mesh layout and calculation of equation coefficients). The principal results of the calculations are given in Table 6.1.1 and as follows:

Sodium Void Worth (Δk)

Mesh	78 x 40	156 x 80	Extrapolated
Inner core voided, rod in	0.016495	0.016479	0.01647
Inner core voided, rod out	0.017506		
Inner, outer core voided, rod in	0.020590		
Inner, outer core, axial blanket voided, rod in	0.019418		

Central Control Rod Worth (Δk)

Sodium in	0.002816	0.00295	0.00299
Sodium voided inner core	0.003827		

Comparison with results obtained by others indicates that the uniform mesh produces a superior result for the control rod worth: the near neighbor is apparently too far away with volume-centered points for an accurate representation of neutron leakage and its change.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

Table 6.1.1. Principal results for VENTURE calculations of large-core sodium-voiding benchmark problems

Case	Sodium Voided	Central Rod Position	RZ Mesh Points	IBM 360/195 Processor Time (min)	Core Leakage Fraction	Total Blanket Absorption Fraction	Peak to Average Core Power Density	k_{eff}
0	No	In	Uniform 78 x 40	7.7 ^a	0.173057	0.185626	1.67756	1.006497
1	No	In	78 x 40	4.5	0.172943	0.185686	1.67365	1.006718
1a	No	In	156 x 80	27.2	0.172935	0.185877	1.67982	1.006410
1 ^b	No	In	(-)		0.17293	0.18594	1.6819	1.00631
2	Inner core	In	78 x 40	5.7	0.179247	0.190889	1.63144	1.023213
2a	Inner core	In	156 x 80	27.3	0.179220	0.191059	1.63602	1.022889
2 ^b	Inner core	In	(-)		0.17921	0.19112	1.6375	1.02278
3	No	Out	78 x 40	5.0	0.171987	0.184204	1.63616	1.009551
3a	No	Out	156 x 80	27.9	0.171924	0.184318	1.64095	1.009356
3 ^b	No	Out	(-)		0.17190	0.18446	1.6425	1.00930
4	Inner core	Out	78 x 40	5.8	0.178722	0.189776	1.59477	1.027040
5	Inner, outer core	In	78 x 40	5.7	0.192932	0.205971	1.59372	1.027308
6	Inner, outer core, axial blanket	In	78 x 40	5.2	0.194207	0.203511	1.60064	1.026136

^aIBM 360/91.^bExtrapolated on the basis of the error being proportional to the mesh spacing squared, we note that the inaccuracy of short-word precision in mesh layout and calculation of equation coefficients causes the extrapolation to lose reliability for small changes.

6.2. ANALYSIS OF STREAMING THROUGH THE PRIMARY COOLANT PIPE CHASEWAY OF THE CLINCH RIVER BREEDER REACTOR*

M. L. Williams W. W. Engle, Jr.

(Reprint of Trans. Am. Nucl. Soc. 27, 779-780, 1977)

One of the most difficult problems in shielding analysis is that of determining radiation streaming through multileg ducts. The piping ducts for the heat transport system (HTS) in a reactor provide especially large streaming paths of this variety. This paper summarizes design support analysis done at ORNL to ascertain the extent of neutron streaming through the primary sodium HTS pipe chaseway in the CRBRP.

Figure 6.2.1 is a plan view of one of the three primary coolant loops in the CRBRP. The coolant loop exits from the reactor cavity at an elevation of ~238 in. above the core centerline. The pipes then make two 90-deg bends and enter the intermediate heat exchanger vault (IH XV). It had been thought that neutrons streaming through the chaseway of the 36-in. hot-leg pipe could possibly enter the IH XV and activate the secondary sodium loop, thus producing a radiation source external to the reactor containment shield. The Oak Ridge analysis of this effect included both numerical computation and a reasonably prototypic experiment at the ORNL Tower Shielding Facility (TSF); however, this paper will focus mainly on the analytical portion of the study.

Fig. 6.2.1. CRBRP primary pipe chaseway.

Although the pipe chaseway geometry is very three-dimensional (3-D) in nature, conventional Monte Carlo techniques were ruled out for the initial analysis because the attenuation was too great. Instead, it was decided to apply 2-D discrete ordinates to each of the three pipe chaseway legs, in a coupled sequence. Thus, the calculational assumption was that (a) each leg could be adequately represented in R-Z geometry, and (b) the calculations for the three legs could be coupled in a reasonable manner. The procedure was as follows:

1. An initial boundary source was entered along the lower surface of leg I of the pipe chaseway.
2. A radial boundary source at the surface between legs I and II was found from discrete ordinates calculation of the leg I geometry.
3. The leg I radial source was transformed to an axial source for leg II.
4. Steps (2) and (3) were repeated for the coupling between leg II and leg III.
5. A discrete ordinates calculation for leg III determined the leakage into the IHXV.

Transforming the angular flux boundary sources from one coordinate system to another required development of the Discrete Angle Transformation Technique. This method essentially rotates the quadrature direction vectors defined in one system to correspond to the directions in the coupled system by using a transformation matrix. Simultaneously, the angular flux is interpolated over spatial coordinates to transform from the axial mesh of the coupled leg. Total leakage was conserved by normalizing the transformed source to agree with an inexpensive Monte Carlo calculation of the pipe intersection.

Discrete ordinate calculations were performed with DOT, using a P₁ Legendre expansion of the scatter cross section and a biased 166 direction quadrature set. Cross

sections were a 51 multigroup set, generated by the AMPX² code for LMFBR studies. The initial boundary source entering leg I was from an earlier calculation of the reactor vessel region.

The computation revealed that the first leg of the pipe chase attenuates about a factor of 900; leg II attenuates about 14,000; and the shorter leg III attenuates less than a factor of 50. These values correspond to an overall source degradation of approximately 2×10^{-9} . Preliminary analysis of the prototypic experiment of the TSF also gives values on the order of 10^{-9} for the effective source attenuation.³ The peak sodium activation in the IHXV was found to be less than 2×10^{-1} reactions/cm³-s, without considering dispersion throughout the vault or attenuation by heat exchanger components. This value is well within the value required by dose constraints. Channel theory⁴ results and other experimental comparisons have also been obtained.

The result of this analysis reduced previous estimates of the secondary sodium activation in the IHXV and indicated that the chaseway design model was conservative. The study used new calculational methods (e.g., DATT, channel theory), which appears to give good results for the complex pipe chase geometry. The computed values for the effective pipe chase attenuation agree well with preliminary experimental analysis.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

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6.3. USE OF VARIATIONAL TECHNIQUES FOR THE ESTIMATION OF NEUTRON DETECTION EFFICIENCY*

Sheng-Chi Lin[†] J. C. Robinson[†] D. L. Selby

(Abstract of Nucl. Sci. Eng. 87, 61-73, 1978)

The Lewins variational functional was employed to formulate approximations to the neutron detection efficiency, which is a parameter required in the measurement of reactivity by the modified source multiplication technique. In particular, a conventional variational method, a variational extrapolation method, a variational interpolation method, and a multi-reference-state variational method were developed for estimating neutron detection efficiency. Results obtained using the various approximate techniques in one and two dimensions were compared with results from exact formulations. The results obtained using the multi-reference-state variational method in all cases and the variational interpolation method in most cases compared very favorably (discrepancies <5%) with results from the exact calculations. The approximate techniques can be cast in a form where very simple calculational capabilities are all that are required to obtain detection efficiency for any given (but arbitrary) subcritical state. Therefore, we conclude that the methodology developed herein would be applicable to on-line applications using minimal computer capabilities.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

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6.4. EVALUATION OF A VARIATIONAL METHOD FOR ESTIMATION OF NEUTRON DETECTION EFFICIENCY*

D. L. Loyd[†] J. C. Robinson[†] D. L. Selby

(Reprint of Trans. Am. Nucl. Soc. 26, 773-774, 1978)

The purpose of this work is to evaluate a multi-reference-state variational method introduced by Lin^{1,2} for the estimation of the neutron detection efficiency for an external detector in a fast reactor system.

The neutron detection efficiency, η_n , is a parameter required for the application of the modified source multiplication (MSM) technique for subcriticality determination.³ Neutron detection efficiency is defined by

$$\eta_n = \frac{\langle \Sigma_d \phi \rangle}{\langle \Sigma_f \phi \rangle} \quad (1)$$

where Σ_d and Σ_f are the macroscopic detector and fission cross sections, and $\langle \rangle$ represents integration over phase space. Application of the MSM technique requires that η_n be known for each reactor configuration for which the technique is applied. Depending on the complexity of the system, the number of different system configurations, and the position of the detector in question, determination of the necessary detection efficiencies by direct solution of the neutron transport equation could involve substantial computing time (cost). Therefore, there is potential for substantial savings in the use of an approximating technique such as Lin's^{1,2} for accurately estimating the neutron detection efficiency of an ex-core detector.

For the work reported here, Lin's variational method was used to estimate η_n for the ZPPR-5 source level flux monitor for a number of different subcritical configurations. In particular, the subcritical configurations chosen for calculating the η_n 's using the variational method are the same as those that had previously been calculated using direct methods.⁴ Results along with the percent difference between the variational and direct methods are presented in Table 6.4.1. The subcriticality of the states ranges from 1.266 to 34.72 dollars subcritical. The reference states[†] used for calculation of the variational results presented in Table 6.4.1 were 34.40 dollars subcritical.

The differences in the η_n 's calculated by the two methods ranged from about 1 to under 6% for perturbations other than large source changes. The last five states listed in Table 6.4.1 involve very large changes in the source at specific intervals; i.e., source changes which involve adding one-tenth of the total fixed source of the core into a small interval. Even with the large source perturbations, the differences in η_n range from a maximum of 5 to 10%.

A comparison was made regarding the overall computing time (cost) of determining the η_n 's for the ex-core detector with ZPPR-5 in these 13 different configurations. Specifically, the direct solution method required computing time which was a factor of 3 more than that required by the variational method, thus establishing a 75% savings for the variational method over the direct solution method. The required computation time (cost) for applying the variational method exceeds the direct method for the first four η_n 's calculated. But while the time requirement for the direct method increases linearly, additional calculations by the variational method are insignificant.

On the basis of the above comparisons, it is concluded that acceptable estimates of the neutron detection efficiency for ex-core detectors can be obtained employing the multi-reference-state variational method of Lin at a substantial saving in computing time relative to a direct solution method.

On the basis of these conclusions, it is recommended that this variational method along with the MSM technique be extended to on-line applications. For example, complex computations for reference states can be made and used to find components of η_n for all possible cross section and source changes. An on-line small computer system can then be

Table 6.4.1. Detection efficiency-comparisons of variational estimates with direct solutions for ZPPR-5 source level flux monitor

Configuration ^a Number	Subcritical Reactivity (ρ)	k_{eff} , Direct Calculation ^b	k_{eff} , Estimate	% Difference ^c
23	-1.266	1.9180-7	1.8073-7	5.8
24	-1.327	1.9125-7	1.8274-7	4.5
25	-1.138	1.8436-7	1.7753-7	3.8
26	-1.403	1.8762-7	1.8486-7	1.5
34 ^d	-34.40	2.5495-7		
35	-34.60	2.6237-7	1.5840-7	1.5
36	-34.72	2.6175-7	2.5894-7	1.1
37	-34.59	2.6113-7	2.5471-7	2.5
38 ^e	-33.70	2.4695-7	2.3351-7	5.4
39 ^e	-33.70	2.4976-7	2.2764-7	8.9
40 ^e	-33.70	2.5494-7	2.3706-7	7.0
41 ^e	-33.70	2.6332-7	2.3866-7	9.4
42 ^e	-33.70	2.6321-7	2.4003-7	8.8

^aThe reference states and perturbed states correspond to different ZPPR-5 configurations (e.g., different amounts of fuel and control material and some source changes) and are described in ref. 4.

^bResults of direct calculation of k_{eff} obtained from ref. 3.

^c% Difference = $\left(\frac{\text{direct calculation} - \text{estimate}}{\text{direct calculation}} \times 100 \right)$.

^dThe solution to the forward and adjoint transport equation for this state was used as reference for the variation estimates.

^eThese states involved very large changes in the fixed source.

used for the relatively simple calculations necessary to estimate the k_{eff} 's for each successive configuration. In this way, each step of a loading (or reloading) procedure may be followed.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

† University of Tennessee.

‡ Those states chosen for solving the forward and adjoint transport equations, the solutions of which are used in the variational method.

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6.5. ANALYSIS OF THE ZPPR-5 SOURCE LEVEL FLUX MONITOR (SLFM) EXPERIMENT*

D. L. Selby G. F. Flanagan D. L. Loyd†

(Reprint of Trans. Am. Nucl. Soc. 27, 905-906, 1977; also published as ORNL-5314, February 1978)

This paper summarizes the ORNL analytical analysis of the source level flux monitor (SLFM) phase of the zero-power plutonium reactor (ZPPR) experimental program designated as Assembly 5. All steps of the experiment were analyzed using modified source multiplication (MSM) methods. The basic MSM equation is¹

$$S_i = S_0 \frac{S_i W_i \bar{v}_0 CR_0 B_0}{S_0 W_0 \bar{v}_i CR_i B_i},$$

where

 S = effective source, W = detection efficiency, $\bar{v}$ = average number of neutrons per fission, CR = detector count rate, B = effective delayed-neutron fraction.

The subscripts 0 and i refer to reference state 0 and configuration i , respectively. Of the above parameters, the detector count rate is obtained experimentally and S , $\bar{v}$, and B are integral quantities that can be obtained with little difficulty since they involve core fluxes only. The parameter that is not easy to calculate is the quantity W , called the detection efficiency. This represents the number of reactions in the detector per fission reaction in the core. If the ratio W_i/W_0 can be calculated accurately, the level of subcriticality can easily be obtained provided a method of calibration is available. The difficulty involved in this calculation is calculating the neutron flux at the detector location which is located several mean-free-paths from the core boundary.

Several three-dimensional (3-D) calculations were made using the 3-D Monte Carlo code KENO (ref. 2) to calculate the energy-dependent axial leakages. These leakages were then used to form fictitious cross sections. These fictitious cross sections were determined using the equation

$$\Sigma_{g,i} = \frac{L_{g,i}}{\phi_{g,i}},$$

where $\Sigma_{g,i}$ is the fictitious cross section for group g and region i ; $L_{g,i}$ is the net axial leakage for group g between the region i and the axial blanket; and $\phi_{g,i}$ is the g -group volumetric flux associated with region i . These fictitious cross sections were then used in the 2-D X-Y transport calculations to account for axial leakage.

Using the computer code DOT 3.5 (ref. 3), P-S₄, and 44 energy groups, transport calculations were made to obtain the neutron flux. Heterogeneity effects in the core were accounted for by cell weighting the core cross sections. The 44-energy-group cross sections were collapsed from a fine-group cross-section structure using zone weighting. The resulting reaction rates at all detector locations were calculated using GOFRR (ref. 4). These results were then utilized to determine the detection efficiency to be used in the MSM equation. At each step of the analysis the reactivity inferred by each detector was compared to a reference reactivity.

Two types of comparisons were made between calculated and experimental results. One comparison is made between the experimental and calculated ratio of the SLFM detector count rate to an in-core detector count rate and the other comparison is made between calculated and experimentally obtained reactivities. In the former comparison it was found that with graphite in the SLFM zone, the calculated-to-experimental (C/E) ratios ranged from 1.09 to 1.175. This agrees favorably with the results of the ²³⁵U foil measurements where a similar C/E of 1.137 was found. With polyethylene in the SLFM zone, the C/E ratios range from 1.20 to 1.24. These results imply there is a systematic error associated with the use of graphite in the SLFM zone and a different systematic error associated with the use of polyethylene in the SLFM zone. This comparison is summarized in Table 6.5.1 for selected steps of the experiment. Steps 23 to 44 contain polyethylene in the SLFM zone, while the other have graphite in the SLFM zone. In the latter comparison (between calculated and experimental reactivities), all systematic errors canceled and the C/E's obtained ranged from 0.951 to 1.04.

Table 6.5.1. Comparison of calculated count rate ratios with experimentally obtained count rate ratios

Case Number	Ratio: $\frac{\text{SLFM Detector Response}}{\text{In-Core Detector Response}}$		C/E
	Calculated	Experimentally	
1	0.0343	0.0301	1.140
2	0.0378	0.0330	1.145
23	0.1433	0.1182	1.212
24	0.1436	0.1170	1.227
25	0.1415	0.1107	1.278
26	0.1402	0.1137	1.233
27	0.1388	0.1159	1.198
28	0.1406	0.1165	1.207
29	0.1411	0.1162	1.214
30	0.1417	0.1171	1.210
31	0.1419	0.1166	1.217
32	0.1414	0.1163	1.216
33	0.1406	0.1163	1.209
34	0.2162	0.1820	1.199
35	0.2287	0.1854	1.234
36	0.2299	0.1868	1.231
37	0.2284	0.1841	1.241
38	0.1953	0.1615	1.209
39	0.2045	0.1814	1.127
40	0.2240	0.1860	1.204
41	0.2347	0.1939	1.210
42	0.2340	0.1983	1.180
43	0.2136	0.1935	1.104
44	0.2192	0.1873	1.170
61	0.0528	0.0483	1.093
62	0.0425	0.0390	1.090

From these comparisons it was determined that reactivity levels could be inferred by the SLFM detector provided detection efficiency ratios can be calculated.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

² University of Tennessee.

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6.6. ASSESSMENT OF GEL-SPHERE-PAC FUEL FOR FAST BREEDER REACTORS: REACTOR PHYSICS CONSIDERATIONS*

D. L. Selby

(Excerpt from ORNL-5185, compiled by W. J. Loekey and J. E. Selle, October 1976)

With the present interest in remote fabrication and sphere-pac fuel, several reactor physics questions have arisen regarding this fuel. Two such questions deal with the Doppler coefficient: (1) What effect does the sphere-pac arrangement have on the Doppler coefficient for a fast reactor system? (2) What is the change, if any, in the time constant associated with the Doppler effect?

Doppler Coefficient. The currently preferred sphere-pac rod design calls for fissile material in only the coarse and medium-size spheres. Thus, the fine spheres contain only fertile material. This results in a very heterogeneous composition, which might affect the magnitude of the Doppler coefficient.

To determine the effect on the Doppler coefficient, first-flight collision probabilities were calculated for neutrons at various resonant energies. The hypothesis is there will not be any significant difference in the Doppler coefficient for the sphere-pac and pellet fueled systems if there is no significant difference in the first-flight collision probabilities.

The calculated first-flight collision probabilities for five different neutron energy levels are presented in Table 6.6.1. All differences between the heterogeneous (sphere-pac) and homogeneous (pellet) fuel mixtures are less than 5%. These differences are considered to be small. Therefore, it is concluded that the Doppler coefficient for the sphere-pac fuel will be essentially the same as for the pellet material with the same fissile-fertile smear mixture.

Table 6.6.1. Calculated first-flight collision probabilities for five different neutron energies

Energy (eV)	Homogeneous			Heterogeneous			Difference, %		
	Oxygen	²³⁸ U	Pu	Oxygen	²³⁸ U	Pu	Oxygen	²³⁸ U	Pu
6.6	0.00253	0.9963	0.001216	0.00252	0.9964	0.00116	-0.6	0.02	-4.6
7.8	0.05093	0.06871	0.8804	0.05155	0.0697	0.8787	1.2	1.4	-0.19
41	0.1120	0.1889	0.6991	0.1137	0.1924	0.6939	1.5	1.8	-0.75
100	0.02088	0.939	0.04011	0.02083	0.9400	0.03913	-0.2	0.11	-2.5
4.5 x 10 ⁵	0.7727	0.1941	0.03316	0.7727	0.1938	0.03341	0.0	-0.15	0.75

Fuel Time Constant. The time constant associated with the Doppler coefficient may also be affected by the heterogeneous arrangement if the fertile fines sphere-pac process is used. Thus, an analysis was made to determine the effect of this distribution on the time response of the fuel rod to changes in power.

The time-dependent heat conduction equation for a slab fuel rod with uniform thermal conductivity and heat generation rate changes from $\dot{q}_1$ to $\dot{q}_2$ at time zero were used. The results for a bare UO₂ fuel element undergoing a step change in heat generation from $\dot{q}_1$ to $0.9\dot{q}_1$ are as follows:

τ (pellet)	τ (sphere-pac)	τ Difference
0.525 ± 0.007	0.555 ± 0.007	5.7 ± 2.8

The results of these calculations demonstrate that the nonuniformity in fuel distribution in the sphere-pac will have little, if any, effect on the time response of the fuel rod. Certainly such an effect will not be measurable since uncertainties in geometry, thermal properties, heat transfer coefficients, etc. will more than mask the change due to nonhomogeneity.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

6.7. CALCULATIONAL AND EXPERIMENTAL ANALYSIS OF A ²³⁹Pu SAMPLE IRRADIATED IN EBR-II*

M. L. Williams S. Raman*

(Reprints of Trans. Am. Nucl. Soc. 30, 766-768, 1975.)

This paper summarizes the status of a program at Oak Ridge National Laboratory (ORNL) whose ultimate goal is to obtain integral data for actinide transmutation in a fast reactor spectrum.¹ It focuses on comparative calculations performed with preliminary ENDF/B-V data, and on the application of depletion sensitivity theory to the integral experiment analysis.

In 1966, several actinide samples ranging from ²³²Th to ²⁴¹Pu were irradiated for four years in the fast reactor EBR-II at Argonne National Laboratory (ANL), Idaho. The purpose of this research was to experimentally ascertain the isotopic composition of the irradiated sample. However, after one sample had been analyzed, the ANL program was halted until 1977 when interest was revived in obtaining better actinide cross sections in the higher energy range. At that time the other irradiated samples were sent to ORNL for further analysis as part of its cross-section measurement program.

Oak Ridge is presently examining the second sample, which was nominally 94.1-mg ²³⁹PuO₂, with some impurities present. Accompanying the experimental analysis is a calculational program whose objectives are to: (a) provide a comparison with the integral experiment, as a measure of the performance of preliminary ENDF/B-V actinide cross sections, (b) determine cross-section data important for the buildup of various isotopes, and (c) to establish the effect of data uncertainties on the calculated/measured discrepancies.

To accomplish task (a), preliminary ENDF/B-V cross sections were processed with MINX.² Cross sections were collapsed to 14 groups with a ZPR 6/7 spectrum, and then to one group with an experimentally determined EBR spectrum. The cross-section values, along with measured exposure data, were input to the ORIGIN³ code to compute the sample transmutation. A comparison of the calculated and measured plutonium isotopics is given in Table 6.7.1.

Table 6.7.1. Comparison of calculated and measured plutonium concentrations

Pu Isotope	ORNL Measured (%)	ORNL Calculated (%)
238	0.020	0.012
239	93.15	93.01
240	6.56	6.57
241	0.251	0.266
242	0.026	0.030

Depletion sensitivity analysis was performed on the sample, using the ORIGEN-A⁴ code to calculate the "nuclide adjoint functions" required for the perturbation expressions. Table 6.7.2 shows sensitivity coefficients for data important to various "responses" (i.e., nuclide concentrations). The values (relating percent change in the response to a one percent change in data) are an indication of the relative importance of the specified data to the build-up of the specified isotopes. For example, it can be seen that the ²⁴¹Am concentration in this sample is most sensitive to the decay constant of ²⁴¹Pu.

Table 6.7.2. Sensitivity coefficients for various responses to data used in ORIGEN calculation

Parameter α_i	Sensitivity Coefficients			
	R1	R2	R3	R4
P9 ^a σ_c	5.30-3	4.58-2	2.45-1	-1.64-2
P9 σ_f	-2.27-4	-2.75-3	-1.20-3	-1.77-1
P0 σ_c	5.47-2	3.06-1	-2.01-2	0
P0 σ_f	-1.19-3	-1.07-2	-6.09-2	0
P1 σ_c	-3.39-3	-1.83-2	0	0
P1 σ_f	-2.89-3	-1.56-2	0	0
P1 λ	3.91-1	-1.42-1	0	0
A1 σ_c	-6.96-2	0	0	0
A1 σ_f	-4.92-2	0	0	0
P9 N_0	5.31-3	4.62-2	2.48-1	1.0
P0 N_0	5.01-2	2.65-1	7.54-1	0
P1 N_0	3.72-1	6.89-1	0	0
A1 N_0	5.75-1	0	0	0

^aP9 indicates ²³⁹Pu, P0 indicates ²⁴⁰Pu, etc.

R1 = ²⁴¹Am concentration after 1374 days irradiation.
R2 = ²⁴¹Pu concentration after 1374 days irradiation.
R3 = ²⁴⁰Pu concentration after 1374 days irradiation.
R4 = ²³⁹Pu concentration after 1374 days irradiation.

Finally, using the sensitivity coefficients, the uncertainty in the computed nuclide concentrations due to data uncertainties can be computed. Table 6.7.3 shows uncertainties in some of the concentrations, due only to uncertainties in plutonium data.

Table 6.7.3. Computed uncertainties in plutonium concentrations in irradiated sample

Data	$\delta\sigma/\sigma$ (%)	$\delta R1/R1$ (%)	$\delta R2/R2$ (%)	$\delta R3/R3$ (%)	$\delta R4/R4$ (%)
P9 σ_c	6.7	3.6-2	3.0-1	1.6	1.1-1
P9 σ_f	3.0	6.8-4	8.3-3	3.6-3	5.3-1
P0 σ_c	10.0	5.6-1	3.1	2.1-1	0
P1 σ_c	12.4	4.2-2	2.3-1	0	0
P1 σ_f	3.0	8.7-3	4.7-2	0	0
P1 σ_{γ}	2.7	1.1	3.8-1	0	0
Totals		1.2%	3.1%	1.6%	0.54%

P9 indicates ^{239}Pu , P0 indicates ^{240}Pu , etc.

Results are summarized below:

1. Comparison between measured values and values calculated with preliminary ENDF/B-V data is good for most of the plutonium isotopes.

2. Sensitivity analysis on the sample indicates the following nuclear data as being most critical for the given nuclides: ^{241}Am , ^{241}Pu , ^{241}Pu , $\sigma_c^{240}\text{Pu}$, ^{240}Pu , $\sigma_c^{239}\text{Pu}$, ^{239}Pu , $\sigma_f^{239}\text{Pu}$.

3. Uncertainties in nuclide concentrations due to uncertainties in P. data range from 0.5% for ^{239}Pu to 3% for ^{241}Pu .

* Research sponsored by DOE Office of Basic Energy Sciences, Division of Nuclear Sciences, Physics Division.

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6.8. PRELIMINARY NEUTRONIC STUDY OF ACTINIDE TRANSMUTATION IN A FAST REACTOR*

M. L. Williams J. W. McAdoo G. F. Flanagan

(Abstract of ORNL/TM-830, August 1978)

Preliminary results are cited from a neutronic study of actinide recycle in a fast reactor, using an LMFBR model. The work was performed as part of a Chemical Technology

Division program for assessing the merit of actinide transmutation as a means of waste disposal. Calculations were performed by coupling two-dimensional diffusion theory calculations with point-depletion recycle calculations. Cross sections were obtained from a "preliminary" version of ENDF/B-V.

Results are given concerning the recycled-actinide equilibrium condition, and the waste-actinide destruction rate. The effect of adding U-238 diluent to reduce power peaking is examined parametrically. The heterogeneous effects of placing an actinide target assembly in a fuel lattice are also examined.

*Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

†UCC-ND Computer Sciences Division.

GAS-COOLED REACTOR PROGRAM

6.9. INTERIM REPORT ON CORE PHYSICS AND FUEL CYCLE ANALYSIS OF THE PEBBLE BED REACTOR POWER PLANT CONCEPT*

D. R. Vondy

(Abstract of ORNL/TM-8111, December 1977)

Calculations were made to predict the performance of a pebble bed reactor operated in a mode to produce fissile fuel (high conversion or breeding). Both a one-pebble design and a design involving large primary feed pebbles and small fertile pebbles were considered. A relatively short residence time of the primary pebbles loaded with ^{235}U fuel was found to be necessary to achieve a high breeding ratio, but this leads to relatively high fuel costs. A high fissile inventory is associated with a low C/Th ratio and a high thorium loading, causing the doubling time to be long (even though the breeding ratio is high) and the fuel costs of electrical product to be high. Production of ^{235}U fuel from ^{235}U feed was studied and performances of the converter and breeder reactor concepts were examined varying the key parameters.

*Research sponsored by DOE Office of Nuclear Energy Programs, Division of Nuclear Power Development.

6.10. PEBBLE BED REACTOR CORE PHYSICS AND FUEL CYCLE ANALYSIS*

D. R. Vondy B. A. Worley

(Abstract of ORNL-66-1, in preparation)

Calculations were done to analyze the projected performance of the pebble bed reactor concept. Operation as a burner on a throwaway cycle, converter with recycle, pre-breeder, and breeder with ^{235}U recycle fuel were considered. Medium enriched fuel was considered in addition to fully enriched ^{235}U for a denatured cycle and also low enriched uranium feed. The base calculations were done for electrical energy generation application in a 1200-MWe plant. Local methods of analysis were used including the application of a steady-state continuous-fueling model considering one or two dimensions to characterize performance by treating a single point in time.

Results are reported for a wide range of the design and operation parameters. The analysis shows that the flexibility of the concept can be exploited. The pebble bed reactor concept seems to be competitive with other concepts, but breeding of a significant amount of fuel does not appear practical.

*Research sponsored by DOE Office of Nuclear Energy Programs, Division of Nuclear Power Development.

6.11. ANALYSIS OF GAS-COOLED FAST REACTOR SHIELD DESIGNS*

D. E. Bartine L. R. Williams

(Abstract of paper presented at FFR Shielding Seminar, Chalk Point, N.S.W.A., November 13-17, 1978)

In its shielding program for the Gas-Cooled Fast Reactor (GCFR) as conceived by General Atomic, Oak Ridge National Laboratory has developed an advanced shielding analysis system that incorporates the latest analysis techniques for converging to a shield design compatible with other design parameters such as cooling and structural requirements or material compatibility. Basically the system consists in applying the various techniques in a logical sequence to a given design, thereby generating a large body of data to serve as an information base for subsequent redesign.

As an illustration, this system is applied to successive typical models for the GCFR, resulting in a reduction in the thickness of the radial shield and redesign of the lower shield region. In principle, the design-analysis-redesign iterations would continue until they converge upon an acceptable configuration.

*Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

ALTERNATE FUEL CYCLE PROGRAM⁵

6.12. DENATURED FAST REACTOR SYMBIOSIS*

T. J. Burns J. R. White[†] D. E. Bartine J. M. Kallfelz[‡] D. M. Rowland[‡]*(Abstract of Trans. Am. Nucl. Soc. 17, 444-447, 1978)*

The use of denatured uranium (^{235}U denatured with ^{238}U) fuel has been proposed as a means for increasing the proliferation resistance of fresh recycle nuclear fuel.^{1,2} Initial scoping studies have indicated the feasibility of the denatured uranium fuel cycle for LMFBRs;³ however, a significant property of the "standard" denatured LMFBR (i.e., no internal blanket assemblies used) is that although the breeding ratio is greater than unity, the ^{238}U component of the breeding ratio is less than unity. Hence, a makeup source or ^{238}U must be provided. (Use of the plutonium bred from the ^{235}U denaturant in the denatured reactors is disallowed.) To provide the ^{238}U , a symbiotic system is envisioned in which dispersed denatured reactors would operate in conjunction with "transmuters" located in a centralized energy center as indicated by Fig. 6.12.1. The transmuters would be designed to utilize the plutonium produced in the dispersed reactors to produce ^{238}U , the use of the plutonium thus being restricted to the secure energy center after it has been reprocessed. This paper summarizes the results of work addressed at assessing some of the implications of such a symbiosis.

One symbiotic parameter of particular importance is the ratio of power produced in the dispersed reactors to that produced in the energy center transmuters. Since two fissile species are involved in the symbiotic relationship, a "window" of allowable outside/inside ratios is present. The symbiotic power ratio is constrained by ^{235}U production on one side, and by Pu production on the other. One interesting aspect of the symbiosis is the existence of an asymptotic value of the outside/inside power ratio. Assuming that the inventory of each fissile isotope is compounded annually (in a manner similar to that used to calculate a reactor's compound doubling time), the time-dependent behavior of the symbiotic power ratio can be determined. Figure 6.12.2 illustrates the behavior for the two extreme initial values of the power ratio: no denatured reactors, and the maximum

⁵See also Paper 6.44.

Fig. 6.12.1. Schematic diagram of fuel flow for denatured reactor and energy center symbiosis.

Fig. 6.12.2. Symbiotic power ratio for dispersed denatured LMFBRs ($20 \text{ }^{235}\text{U}$ in ^{238}U) coupled to Pu/Th LMFBRs in energy center (equilibrium cycle mass flows, 2% losses, 1 year out-of-reactor time).

number of denatured reactors that can be maintained by the transmuter. As Fig. 6.12.2 indicates, the asymptotic value is approached regardless of the starting configuration. The asymptotic value is also the point where the growth rate of the two fissile inventories is identical. Hence, each component of the system (i.e., dispersed reactors and centralized reactors) is growing at the same rate and a compound symbiotic system doubling time can be determined for this value of the power ratio. Table 6.12.1 indicates the various symbiotic system parameters for a variety of transmuter and dispersed reactor combinations. Also indicated are the corresponding values for a reference (Pu/ ^{238}U)O₂ system operating with no fuel cycle restrictions.

Table 6.12.1. Equilibrium cycle symbiotic parameters
(2% losses, 1 year cut-of-reactor time)

Energy Center Reactor LMFBR	Dispersed LMFBR (% Denatured)	Asymptotic Outside/Inside Power Ratio	Compound System Doubling Time (yr)
(Pu/U)O ₂ + ThO ₂ RB	~12	0.35	14.3
(Pu/U)O ₂ + ThO ₂ RB	20	0.52	16.5
(Pu/U)O ₂ + ThO ₂ RB	40	0.94	21.9
Pu/Th	~12	1.67	24.7
Pu/Th	20	2.83	33.8
Pu/Th	40	6.91	56.0
	(Pu/U)O ₂	∞	11.3

Possible changes in the symbiotic system parameters due to heterogeneous denatured designs and/or heterogeneous transmuters are also considered. Preliminary calculations indicate that use of ThO₂ internal blanket assemblies in the denatured reactors significantly reduces (or for some designs, may eliminate) the ^{233}U makeup requirement.

*Research sponsored by DOE Office of Nuclear Energy Programs, Division of Nuclear Research and Applications.

†Computer Sciences Division.

‡Georgia Institute of Technology.

1. U.S. Department of State, Committee on Atomic Energy (D. E. Lilienthal, Chairman). "... A Report on the International Control of Atomic Energy," U.S. Department of State Publication 2498, 127-198, U.S. Government Printing Office, Washington, D.C. (March 1946).
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3. T. J. Burns and D. E. Bartine, "Feasibility of Denatured LMFBRs," Trans. Am. Nucl. Soc. 26, 285 (1977).

6.13. ALTERNATIVE FUEL CYCLES FOR BREEDER REACTORS*

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Recently accentuated concerns regarding the possible use of reactor fuel as a source of weapons material have led to an overall re-assessment of various alternative fuel cycles. In particular, the recycle of Pu and the development of Pu-producing breeder reactors has been questioned. Long-term viability and growth of fission-based nuclear generating capacity, however, must ultimately rely on both recycle fuel and breeder reactors. Prudence therefore dictates that any alternative cycle which has potential for reducing the attractiveness of breeder reactor fuel as a source of nuclear weapons material be examined. The denatured fuel cycle¹⁻³ is one such option being studied. It extends some of the proliferation characteristics of the current once-through low-enriched uranium cycle to the recycle/breeder scenario. The fresh fuel consists of fissile ²³³U "denatured" with ²³⁸U to preclude direct use of the fissile material as a source for weapons manufacture.

In assessing the impact of the alternate fuel cycles, a reference design⁴ (Pu/²³⁸U) was selected and analyzed. The performance parameters of alternate fissile/fertile fuel combinations were then calculated by replacing the reference core and blanket material by the appropriate alternative material(s). That is, no attempt was made to modify or optimize the designs to account for differences in the thermophysical properties of the alternative fuel materials relative to the reference system. Table 6.13.1 summarizes some of the significant nuclear performance parameters for certain alternate fissile/fertile combinations. It should be noted that the results are for a "classical" or homogeneous LMFBR (i.e., two core enrichment zones surrounded by fertile blanket regions).

Table 6.13.1. Alternate oxide-fueled LMFBR performance parameters
(2% losses, 1 year ex-reactor inventory)

Core		Uranium Enrichment	Axial Blanket	Radial Blanket	Breeding Ratio	Fissile Doubling Time (yr)	Energy Growth Rate (%/yr)
Fissile	Fertile						
Pu	²³⁸ U		²³⁸ U	²³⁸ U	1.272	12.7	5.4
Pu	²³⁸ U		²³⁸ U	²³² Th	1.272	13.0	5.3
²³³ U	²³⁸ U	12	²³⁸ U	²³² Th	1.127	24.3	2.8
²³³ U	²³⁸ U/ ²³² Th	12	²³² Th	²³² Th	1.121	26.6	2.6
²³³ U	²³⁸ U/ ²³² Th	20	²³² Th	²³² Th	1.086	43.2	1.6
²³³ U	²³⁸ U/ ²³² Th	40	²³² Th	²³² Th	1.048	116.4	0.6
²³³ U	²³² Th	100	²³² Th	²³² Th	1.020	-696.4	-0.1

As indicated by the data of Table 6.13.1, the breeding performance of the alternate LMFBR fuel cycles is significantly lower than that of the reference system. The principal cause of this performance degradation is the higher value of the absorption cross-section of ²³⁸U relative to the fissile Pu isotopes. Additionally, the replacement of ²³⁸U by ²³²Th as the core fertile material also has a deleterious effect -- resulting from the markedly lower fast fission effect of ²³²Th relative to ²³⁸U.

A further point which must be addressed regarding the denatured reactors is their self-sufficiency in terms of the fuel material, ^{233}U . Since the denatured systems contain ^{235}U in addition to ^{232}Th , Pu is produced via neutron capture. Thus in evaluating the self-sufficiency of a fast breeder reactor, the ^{233}U component of the overall breeding ratio is of primary importance since the bred Pu cannot be recycled back into the denature system. As illustrated schematically by Fig. 6.13.1, the ^{233}U component of the breeding ratio increases as the allowable denatured enrichment is increased (allowing more Th in the fuel material). More importantly, the magnitude of the ^{233}U component of the breeding ratio is very sensitive to the allowable degree of denaturing at the lower enrichments (i.e., between 12% and 20%). Therefore, even though the overall breeding ratio decreases as the allowable enrichment is raised, the decrease in the required ^{233}U makeup is even more significant and presents a strong incentive from a performance viewpoint to set the allowable enrichment as high as practicable from a proliferation viewpoint.

Fig. 6.13.1. Alternate LMFBF fuel cycle breeding ratio components.

One means of improving the self-sufficiency of a denatured reactor is to utilize a geometrically heterogeneous design.⁶ Although initially conceived as a means to increase the breeding performance (as well as to assuage certain safety related problems), the heterogeneous design, which has fertile fuel assemblies interspersed with fissile driver assemblies, is of particular interest for these alternate cycles because of its greater breeding flexibility. Our calculations show that the use of ThO_2 internal blanket assemblies in the denatured LMFBFs, in addition to increasing the overall breeding ratio, results in a larger ^{233}U to Pu production ratio. In fact, it appears feasible to design a "self-sufficient" denatured heterogeneous LMFBF using ~30% denatured uranium as a fuel.

Since all denatured reactors require an initial inventory of ^{233}U , as well as varying amounts of ^{235}U as makeup material, a second class of reactors must be considered when evaluating the denatured fuel cycle. The purpose of these systems would be to produce the ^{233}U required by the denatured reactors. Possible LMFBF candidates for this role are the Pu/ ^{238}U reactor with a ThO_2 radial blanket, a Pu/Th LMFBF, and a ^{235}U /Th breeder. All three

of these systems, since they are not denatured, must be subject to rigorous safeguards, that is, operated only in nuclear weapon states or in internationally controlled energy centers. A symbiotic system, such as that depicted in Fig. 6.12.1, is envisioned in which the "inside" reactors consume Pu and produce ^{233}U for the denatured reactors and vice-versa. Certain characteristics of three possible candidate reactors for the inside role are given in Table 6.13.2. Clearly, each system has its own unique properties. From the standpoint of ^{233}U production capability, the hybrid Pu/Th LMFBR is clearly superior. However, it does require a large quantity of fissile Pu as makeup since it essentially "transmutes" Pu into ^{233}U . The Pu/ ^{238}U + ThO₂ radial blanket LMFBR generates significantly less ^{233}U but also markedly reduces the required Pu feed. In fact, for the case illustrated, this system actually produces a slight excess of Pu. The third possible LMFBR system considered is the ^{233}U /Th breeder which is characterized by a very small excess ^{233}U production. Moreover, it does not provide a means for utilizing the Pu which is bred

Table 6.13.2. Performance characteristics of some oxide-fueled ^{233}U production LMFBRs

Fuel Material	Blanket Material	Net Equilibrium Cycle Production (kg/GWe-y)		Breeding Ratio
		^{233}U	Pu	
Pu/Th	Th	+582.6	-492.8	1.144
Pu/ ^{238}U	Th	+154.2	+30.4	1.272
^{233}U /Th	Th	+15.2		1.020

in the denatured systems. The use of heterogeneous designs for the inside role has also been addressed. The primary benefit of the heterogeneous designs lies in the increased ^{233}U production attainable through the use of ThO₂ internal blankets. Since the inside reactors represent a higher proliferation risk than the dispersed reactors, there is an obvious incentive to minimize the number of inside reactors. Thus, the amount of power generated by the dispersed reactors relative to that generated inside the energy centers is a crucial symbiotic parameter. The power ratio (i.e., outside power/inside power) is typically characterized by a symbiotic "window," constrained on one side by ^{233}U production and by Pu production on the other. Moreover, once initiated, a system of dispersed denatured LMFBRs and energy center LMFBRs is characterized by an asymptotic power ratio, at which point the growth rate of each fissile inventory is identical. Thus, the asymptotic value can be viewed as the "neutral" symbiotic power ratio. Additionally, a compound system doubling time can be calculated at this value since each component of the system is increasing at the same rate. Table 6.13.3 gives some symbiotic system parameters for certain reactor combinations. As indicated, a trade-off clearly exists concerning fissile doubling time and asymptotic power ratio. Use of heterogeneous designs for one or both components tends to increase the asymptotic power ratio as well as decrease the overall system doubling time.

The impact of the denatured fuel cycle on breeder-based nuclear power will clearly be significant, resulting in reduced power growth capability. Additionally, the implementation of the denatured fuel cycle will require changes in international institutional arrangements due to the requirement for safeguarded energy centers as well as international agreement on what constitutes "adequate" denaturing. Many of these questions, as well as the assessment of the subjective value of denatured fuel from a proliferation standpoint, thus will clearly involve considerations other than the technical aspects addressed here.

Table 6.13.3. Equilibrium cycle symbiotic parameters
(2% losses, 1 year out-of-reactor time)

Energy-Center Reactor LMFBR	Dispersed LMFBR	Power Ratio			CSDT (yr)	Energy Growth Rate (%/yr)
		Min.	Asym.	Max.		
(Pu/U)O ₂ + ThO ₂ RB	~12% denatured	0.00	0.30	0.42	14.5	4.8
(Pu/U)O ₂ + InO ₂ RB	20% denatured	0.00	0.55	1.00	17.3	4.0
(Pu/U)O ₂ + ThO ₂ RB	40% denatured	0.00	1.00	2.63	23.7	2.9
(Pu/Th)O ₂	~12% denatured	1.13	1.32	1.57	29.0	2.4
(Pu/Th)O ₂	20% denatured	2.52	2.83	3.79	40.9	1.7
(Pu/Th)O ₂	40% denatured	6.55	6.92	9.93	88.7	0.8
	(Pu/U)O ₂				12.7	5.4

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Nuclear Research and Applications.

† Computer Sciences Division.

‡ Georgia Institute of Technology.

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2. H. A. Feiveson and T. B. Taylor, "Security Implications of Alternative Fission Futures," Bull. At. Scientist 32, 14 (December 1976).
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6.14. PRELIMINARY EVALUATION OF ALTERNATIVE FUEL CYCLE OPTIONS UTILIZING FAST BREEDERS*

T. J. Burns J. R. White[†]

(Abstract of BNW-8000, March 1978)

Various "alternative" fuel cycles have been proposed to alleviate concerns about reactor fuel being diverted for the manufacture of nuclear weapons. This report addresses the impact of such fuel cycles on the performance of the classical (homogeneous) oxide-fueled Liquid Metal Fast Breeder Reactor. The primary fuel cycle analyzed is the ²³⁵U/²³⁸U/²³²Th fuel cycle in which the fissile component ²³⁵U is "denatured" with ²³⁸U in order to restrict the use of reactor fuel as a source of weapons-usable material. The denatured reactor performance as a function of the fissile enrichment (²³⁵U in U) is evaluated. The associated energy center concept in which dispersed denatured reactors are coupled to ²³⁵U production reactors operating in secure energy centers is described. The symbiotic

system of dispersed/energy center reactors is analyzed both from the standpoint of energy growth supported and dispersed/centralized power production. Lastly, the effects of proposed changes in the nuclear data on both the reactor performance and symbiotic system characteristics are addressed.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Nuclear Research and Applications.

† Computer Sciences Division.

6.15. BENCHMARK EXPERIMENTS WITH THORIUM* IN FAST REACTOR LATTICES*

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(Abstract of paper presented at International Conference on Nuclear Physics Data for Reactors and Other Applied Purposes, Harvard, Oxford Kingdom, September 25-29, 1978)

In order to provide checks on the reliability of nuclear data needed for calculations on thorium reactor fuel cycles, a series of integral measurements in fast reactor lattices has been carried out. Regular lattices containing thorium allow checks on the nuclear cross sections used, and heterogeneous configurations provide tests of both data and methods. The broad conclusion is that, for the parameters we have measured, predictions based on ^{232}Th and ^{235}U data in ENDF/B-IV agree well with experimental values in the light of the stated uncertainties in the library data.

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† Swiss Federal Institute for Reactor Research.

6.16. THE ROLE OF THORIUM FUEL CYCLES*

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(Reprint of Trans. Am. Nucl. Soc. 22, 349, 1977; similar paper presented at AEC-IEEE Joint Power Generation Conference, Long Beach, California, September 19-21, 1977)

An evaluation is given of the role that thorium fuel cycles can play in a nuclear economy which utilizes U_3O_8 and separative work of relatively high cost, and in an economy where proliferation concerns may influence the recycle of bred fissile fuel. In performing this, three fuel cycles are considered: (a) the thorium fuel cycle (with possible recycle of bred ^{233}U); (b) the uranium cycle (with possible recycle of bred Pu); and (c) a mixed uranium/thorium cycle in which uranium having an enrichment less than about 20% is utilized with thorium (termed denatured fuel cycle) with recycle of the bred ^{233}U . Thermal reactors considered are: light-water reactors (LWRs), spectral shift converter reactors (SSCRs), high-temperature gas-cooled reactors (HTGRs), and molten salt reactors (MSRs); fast reactors considered are liquid-metal-cooled fast breeder reactors (LMFBRs) and gas-cooled fast breeder reactors (GCFRs).

For all the thermal converter reactors, the thorium cycle with recycle of bred fuel gives improved fuel utilization over that of the uranium cycle; however, the best economic and fuel utilization performance is obtained in HTGRs and MSRs. The improvement in fuel utilization is dependent on the timing of thorium cycle introduction, the U_3O_8 resource base, and the nuclear power growth scenario. Based on a nuclear power level of 450 GW(e) in the year 2000 with no subsequent converter reactor power growth, a U_3O_8 resource of 3.5 million tons, and recycle of bred fuel, commercial introduction of thorium-fueled reactors in the year 2000 led to an increase (relative to uranium cycle use in LWRs) of 3 to 54% in the energy available from the above uranium resource, with the highest increase associated with MSR use. However, the MSR commercialization date may lag that of the other thermal reactors by five years or more and under this situation the most attractive thorium-fueled thermal reactor appears to be HTGR. Following the HTGR in thorium-use attractiveness are the HWR, SSCR, and the LWR, in that order.

The above thermal reactors can all be operated as low-gain breeder reactors on the thorium fuel cycle; however, it does not appear likely that LWRs, SSCRs, HWRs, or HTGRs will operate economically as breeders because of the high fuel recycle costs associated with the low fuel exposures required for breeder operation. MSRs, on the other hand, have the potential of operating more economically as breeders because the cost of the fuel recycle appears much less dependent on fuel exposures than for the other concepts, but at a delayed reactor introduction date.

Use of denatured fuel cycles in thermal reactors leads to a decrease in fuel utilization performance relative to use of the thorium cycle; also, use of the denatured fuel cycle imposes a larger penalty on fuel utilization performance for those systems having the higher conversion ratios. Thus, use of a denatured fuel cycle in "breeder" MSRs drops the fuel breeding ratio to less than unity.

In fast breeder reactors (FBRs), use of the uranium cycle with recycle of bred fuel gives better fuel utilization and economic performance than does use of the thorium fuel cycle. In LMFBRS, however, use of the thorium cycle provides more favorable sodium void reactivity coefficients. While the GCFR has a significantly higher breeding ratio than the LMFBRS when the uranium cycle is used, the breeding ratios of the two systems appear nearly comparable when using the thorium cycle. Imposition of denatured fuel cycles in FBRs because of proliferation concerns leads to fuel utilization characteristics intermediate between those of the uranium and thorium cycles. Use of denatured fuel cycles, however, will still produce significant quantities of plutonium, and it will be required to recycle that plutonium along with ^{239}Pu to obtain an overall breeding system. Burning plutonium to produce ^{239}Pu "energy centers" while using denatured fuel cycles, which consume ^{239}Pu in FBRs, such centers, will provide an overall breeding system, although more limited than overall use of the uranium cycle. While it would be desirable to have a high ratio of "outside/inside" reactors, the overall equilibrium system fuel doubling time decreases as that ratio increases. Relative to a system doubling time of 12 years for a uranium cycle LMFBRS, the doubling time increases to about 25 years for an "outside/inside" ratio of 1.7, and to about 70 years for a ratio of 7 (ref. 1). A high doubling time indicates that emphasis should be given to relatively high performance systems associated with heterogeneous designs and use of carbide and metal fuels. "Inside" breeders can also be combined with "outside" converter reactors, but such combinations provide little flexibility for overall system growth, except when natural U_3O_8 resources are available at reasonable prices.

Overall, the role of thorium in thermal reactors is one of providing a contingency position in case of a significant delay in FBR commercialization; in FBRs, the role of thorium is to provide fissile fuel for denatured cycles in a significant fraction of FBRs, and to improve the sodium void reactivity coefficient in LMFBRS.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Nuclear Research and Applications.

Directors Division.

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6.17. THORIUM ASSESSMENT STUDY QUARTERLY PROGRESS REPORTS FOR SECOND AND THIRD QUARTERS FISCAL 1977*

I. Spiewak,[†] Program Director D. E. Bartine, Program Manager

Contributors:

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(Releases of ORNL/TM-2949, Nov. 1977, and ORNL/TM-2950, October 1977)

The objective of this program is to contribute to the ongoing assessment of the potential role of thorium fuel cycles for alleviating safeguards concerns. Scenarios include (1) no fuel recycle permitted, (2) fuel recycle permitted only in secure regions ("energy parks") with denatured (chemically nonseparable) fuels only outside these regions,

and (3) no limits on fuel recycle. A further objective is to provide nuclear mass balance data on HTGRs required by ERDA contractors for comparative cost-benefit studies.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Nuclear Research and Applications.

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MAGNETIC FUSION ENERGY PROGRAM[§]

6.18. MACROSCOPIC CROSS-SECTION SENSITIVITY STUDY FOR FUSION REACTOR SHIELDING EXPERIMENTS*

Y. Seki[†] R. T. Santoro E. M. Oblow J. L. Lucius[‡]

(Abstract of OPHL-6467, September 1976)

Sensitivities of the calculated neutron and gamma-ray responses of an NE-213 detector to the macroscopic cross sections for neutron interactions in materials used in an integral experiment for fusion reactor shielding studies are presented. Sensitivities for the interactions in the 14-MeV and 1- to 2-MeV energy ranges are found to be large enough to ensure the derivation of a significant amount of information from the experiment.

* Research sponsored by DOE Office of Fusion Energy, Division of Development and Technology.

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6.19. MACROSCOPIC CROSS-SECTION SENSITIVITY STUDY FOR THE TNS INTEGRAL EXPERIMENTS*

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(Reprint of Trans. Am. Nucl. Soc., 197-204, 1975)

Integral experiments are being carried out in the Oak Ridge National Laboratory fusion energy program in support of the design of the TNS device (The Next Step). In conjunction with the experiments, the sensitivities of the response of an NE-213 liquid scintillator detector to the macroscopic cross sections of iron, air, stainless steel, borated polyethylene, and concrete have been calculated for the proposed experimental configuration. The objectives of this work were to evaluate the effectiveness of the experimental system in measuring the radiation attenuation in a typical fusion reactor inner toroidal shield configuration and to provide information to guide detailed two- and three-dimensional analyses of the experiment.

The one-dimensional (1-D) spherical model used to represent the experimental configuration is shown in Fig. 6.19.1. Five measurements in which the composition and thickness of the shield are varied have been proposed and are identified as the 0'th through 4'th measurements. The cross-section sensitivities were calculated using the FORSS¹ code system, a P_3 scattering expansion, an S_1 quadrature, and a coupled 45 neutron, 16 gamma-ray cross-section library.² The location of an NE-213 detector was taken to be 124 cm from the D-T neutron source. For this study, the detector response function is

[§]See also Paper 6.47.

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Fig. 6.19.1. One-dimensional spherical model of the first attenuation experiment.

1.0 for neutrons and gamma rays with energies >500 and 450 keV, respectively, and 0.0 otherwise. The cross-section sensitivities of the neutron and gamma-ray responses were calculated separately using the appropriate adjoint flux for each response.

The cross-section sensitivities of the NE-213 detector neutron and gamma-ray responses to the macroscopic cross sections of the materials comprising the 0'th (no shield) and 4'th (full shield) measurements are given in Table 6.19.1. The entries labeled R_n and R_γ are the integrated neutron and gamma-ray detector responses, respectively. For the 0'th measurement, $R_\gamma = 0.43 R_n$ and for the 4'th measurement, $R_\gamma = 6.6 R_n$. The larger value of R_γ in the latter case arises from the introduction of borated polyethylene into the shield, which is an excellent neutron moderator but ineffective in attenuating gamma rays. The remaining entries in the table are sensitivities per unit lethargy $[(\delta R_i/R_i)/(\delta \Sigma/\Sigma)]$, $i = n, \gamma$ of the detector responses integrated over all neutron and gamma-ray energies. These data are the percentage changes in the detector response $(\delta R_i/R_i)$ due to a 1% increase in the macroscopic cross section $(\delta \Sigma/\Sigma)$. The sensitivity of the responses to the scattering cross section $\Sigma_{s,0}$ was obtained by subtracting the sensitivity to the capture cross section, $\Sigma_{c,0}$ from the sensitivity to the total cross section $\Sigma_{t,0}$. Negative values mean that an increase in the cross section results in a decrease in the response.

The neutron response for the 0'th measurement is sensitive to the iron and stainless-steel cross sections. The positive values of the sensitivity in stainless steel are due to neutrons backscattered into the detector. The detector gamma-ray response is almost equally sensitive to $\Sigma_{\gamma,t}$ for iron and to both $\Sigma_{n,0}$ and $\Sigma_{\gamma,t}$ for stainless steel. The sensitivities of the gamma-ray response to the gamma-ray production cross sections are included in the sensitivity to $\Sigma_{n,0}$, which results in the positive value (+8.40-2) in the sensitivity of the response to $\Sigma_{n,0}$ of iron.

In the 4'th measurement, the neutron response is extremely sensitive to the neutron cross sections for stainless steel, the value being -6.69 for $\Sigma_{n,t}$. The negative values of the sensitivities of the neutron response to the total cross sections of stainless steel, borated polyethylene and iron are nearly proportional to the thickness of each material between the source and the detector. The large values are due to the significant contributions of uncollided source neutrons to the local flux at the end of the shield, a result seen in previous sensitivity studies for the EPR shield design.³ The gamma-ray

Table 6.19.1. Sensitivity of the neutron and gamma-ray responses to macroscopic cross sections

Material	Cross Sections	[[$(\delta R_i/R_i)/(\delta \Sigma/\Sigma)$], $i = n, \gamma$]			
		0'th Measurement		4'th Measurement	
		Neutron ($R_n = 9.23$)	Gamma Ray ($R_\gamma = 3.97$)	Neutron ($R_n = 3.72-3$)	Gamma Ray ($R_\gamma = 2.44-2$)
Iron (Fe)	$\Sigma_{n,total}$	-3.06-1 ^a	-1.47-2	-1.07	-3.72-1
	$\Sigma_{n,capt}$	-9.63-2	-9.88-2	-1.24-1	-1.12-1
	$\Sigma_{n,scat}$	-2.10-1	+8.40-2	-9.50-1	-2.60-1
	$\Sigma_{\gamma,t}$	0.0	-5.55-1	0.0	-4.83-4
	$\Sigma_{\gamma,c}$	0.0	-5.87-3	0.0	-3.90-7
	$\Sigma_{n,t} + \Sigma_{\gamma,t}$	-3.06-1	-5.70-1	-1.07	-3.72-1
SS-304 (SS)	$\Sigma_{n,t}$	+1.48-1	+4.27-1	-6.69	-1.91
	$\Sigma_{n,c}$	-2.75-2	-1.48-1	-7.99-1	-7.07-1
	$\Sigma_{n,s}$	+1.76-1	+5.74-1	-5.89	-1.20
	$\Sigma_{\gamma,t}$	0.0	-4.77-1	0.0	-2.61
	$\Sigma_{\gamma,c}$	0.0	-5.33-3	0.0	-1.16-2
	$\Sigma_{n,t} + \Sigma_{\gamma,t}$	+1.48-1	-5.04-2	-6.69	-4.52
Borated polyethylene (BP)	$\Sigma_{n,t}$			-1.82	-1.38
	$\Sigma_{n,c}$	No borated polyethylene used in the 0'th measurement		-5.30-2	-5.19-1
	$\Sigma_{n,s}$			-1.77	-8.61-1
	$\Sigma_{\gamma,t}$			0.0	-3.88-1
	$\Sigma_{\gamma,c}$			0.0	-1.54-5
	$\Sigma_{n,t} + \Sigma_{\gamma,t}$			-1.82	-1.77

^aRead -3.06 x 10⁻¹.

response for the 4'th measurement is most sensitive to $\Sigma_{\gamma,t}$ and $\Sigma_{n,t}$ for stainless steel, and to $\Sigma_{n,t}$ for borated polyethylene. The sensitivities of the detector responses to the cross sections of air and concrete were found to be very small so they have not been included in the table.

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†Visiting scientist from the Japan Atomic Energy Research Institute, Tokai Research Establishment.

‡Computer Sciences Division.

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6.20. CROSS-SECTION SENSITIVITY ANALYSIS OF A STREAMING EXPERIMENT WITH A TWO-DIMENSIONAL MODEL*

Y. Seki[±] R. T. Santoro E. M. Oblow J. M. Barnes[‡] J. L. Lucius[‡]

(Abstract of ORNL/TM-8688, in press)

Cross-section sensitivity analysis of a proposed streaming experiment for a typical penetration in a fusion reactor shield has been performed using a two-dimensional sensitivity method. The streaming neutrons were shown to be determined mostly by the 4.5-cm-thick duct liner made of iron and the concrete immediately surrounding the liner. The neutrons scattered near the surface of the liner were found to be very important. A recommendation was made to move the proposed detector position to improve spatial resolution in the measurements without changing the sensitivity of the measurements to the nuclear characteristics of the duct materials.

*Research sponsored by DOE Office of Fusion Energy, Division of Development and Technology.

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6.21. COMPARISON OF ONE- AND TWO-DIMENSIONAL SENSITIVITY CALCULATIONS FOR A FUSION REACTOR SHIELDING EXPERIMENT*

Y. Seki[±] R. T. Santoro E. M. Oblow J. L. Lucius[‡]

(Abstract of ORNL/TM-8689, in press)

Cross-section sensitivities of a neutron detector response for a fusion reactor shielding experiment calculated in 1-D and 2-D are compared. The effectiveness of the 2-D calculation in the accurate modeling of the experiment and the detector configuration are demonstrated. At the same time the validity of the 1-D sensitivity study as a pre-analysis is shown and a revision of the 1-D calculational model for a better simulation of the actual experiment is presented.

*Research sponsored by DOE Office of Fusion Energy, Division of Development and Technology.

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6.22. NEUTRONICS AND RADIATION DAMAGE CALCULATIONS FOR FUSION REACTORS*

R. G. Alsmiller, Jr. T. A. Gabriel R. T. Santoro

(Abstract of paper presented at second ANTI-NUCLEAR Symposium on Fusion Reactors, Palo Alto, California, October 3-7, 1977, and published in ORNL/TM-8690, Vol. II, 1978)

Some of the neutronics calculations that have been carried out at the Oak Ridge National Laboratory to assess radiation damage problems in fusion reactors are presented and discussed.

*Research sponsored by DOE Office of Fusion Energy, Division of Development and Technology.

6.23. THE SENSITIVITY OF THE FIRST-WALL RADIATION DAMAGE TO FUSION REACTOR BLANKET COMPOSITION⁺

J. M. Barnes⁺ R. T. Santoro T. A. Gabriel

(Abstract of ORNL/TM-6106, December 1977; also Nucl. Tech. 40, 446-451, 1979)

The atomic displacement and hydrogen and helium gas production rates in a 1-cm-thick type-316 stainless steel first wall have been calculated as a function of blanket composition in a typical one-dimensional fusion reactor model. For a 50-cm-thick blanket, variations in the rates of atomic displacement and hydrogen and helium gas production of factors of 2.7, 1.3, and 1.2, respectively, were obtained. The dependence of the radiation damage responses on the thickness of the first wall and blanket are also given.

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⁺ Computer Sciences Division.

6.24. THE EFFECTS OF Fe-Ni-Co-V STRUCTURAL ALLOYS ON FUSION REACTOR NEUTRONIC PERFORMANCE⁺

J. M. Barnes⁺ B. L. Bishop⁺ R. T. Santoro T. A. Gabriel

(Abstract of ORNL/TM-6647, to be published)

Calculations have been carried out to determine the effect of Fe-Ni-Co-V alloys on fusion reactor neutronic performance. The radiation damage in the first structural wall and the tritium breeding ratio were calculated as a function of alloy composition using the tritium breeding blanket module of the Experimental Power Reactor as the reference design. The atomic displacement rate was found to be approximately constant for all alloy combinations. However, the gas production and tritium breeding exhibit reasonably large variations (~20%) as the cobalt concentration is increased in the alloy.

* Research sponsored by DOE Office of Fusion Energy, Division of Development and Technology.

⁺ Computer Sciences Division.

6.25. SENSITIVITY OF PRIMARY KNOCK-ON ATOM SPECTRA AND DISPLACEMENT PER ATOM CROSS SECTIONS TO DIFFERENT SECONDARY NEUTRON ENERGY AND ANGULAR DISTRIBUTIONS AND "IN-GROUP" WEIGHTING SCHEMES⁺

T. A. Gabriel B. L. Bishop⁺

(Abstract of Nucl. Sci. Eng. 66, 84-90, 1978, and of ORNL/TM-6108, January, 1979)

The sensitivity of primary knock-on atom (PKA) spectra and displacement per atom (dpa) cross sections to different secondary neutron energy and angular distributions and "in-group" weighting schemes is investigated. It is shown that the sensitivity of the PKA spectra and dpa cross sections for the (n,n') unresolved and $(n,2n)$ reactions in iron to different angular distributions and the same secondary neutron spectrum is reasonably large (~15%). For grossly different secondary neutron spectra and the same angular distribution, the change in the dpa cross section is smaller than one would initially expect. It is also shown that for aluminum, the sensitivity of dpa cross sections to different in-group weighting schemes is, for the most part, small.

* Research sponsored by DOE Office of Fusion Energy, Division of Development and Technology.

⁺ Computer Sciences Division.

6.26. TWO- AND THREE-DIMENSIONAL NEUTRONICS ANALYSES OF THE TFTR NEUTRAL-BEAM INJECTORS*

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**Presented at the 1981 American Nuclear Society Meeting, Albuquerque, New Mexico, September 1981.*

Two- and three-dimensional neutronics calculations have been carried out to aid in the design of the neutral-beam injectors for the tokamak fusion test reactor (TFTR). Neutral deuterium is injected into the tritium plasma through a duct that passes through a concrete pillbox-shaped shield (igloo) that surrounds the reactor. The D-T fusion neutrons will stream through this duct into the injector and interact in the components producing heat and other detrimental responses. The fusion neutrons will also impinge on the igloo wall and along with secondary gamma rays leak through the wall and contribute to these responses. The magnitude of the nuclear effects will influence both the design of the injector and a variety of operational and maintenance constraints.

The principal purpose of this study was to estimate the nuclear performance of the injectors. The motivation for adopting two calculational procedures was to obtain a two-dimensional (2-D) calculational model which allows for more economical analysis of the injectors, particularly to study the effects of design changes, and which gives results that are consistent with those obtained from the more detailed 3-D analysis. The 2-D discrete ordinates code DOT¹ using an S₄ angular quadrature and the 3-D Monte Carlo code MORSE² were each used to calculate the spatial dependences of the nuclear heating rates and neutron and gamma-ray scalar flux distributions in the injector. In the 3-D analysis, these data were estimated at several locations in a detailed model of the injector using accurate neutron and gamma-ray energy, spatial, and angular distributions to describe the radiation sources incident on the injector. However, in Monte Carlo calculations, economic constraints limit the number of locations at which the responses and spectra can be estimated since the computation time is a function of the number of detector locations. In 2-D calculations, these responses can be obtained simultaneously at all spatial locations, but the injector model and the source are represented more approximately.

The D-T neutron source distribution was taken to be that at strong-post compression of the plasma³ and was transported using DOT with the TFTR and the igloo modeled in r-z geometry with toroidal symmetry about the z-axis. The outward directed angular flux at the surface of the igloo wall from DOT was processed by the interface code DOMINO,⁴ which generates normalized probability distribution functions in energy, space, and angle for input to MORSE. Distribution functions that account for the radiation streaming through the injector port and the radiation leaking through the igloo wall were obtained separately. The source term for the 2-D radiation transport was obtained by using a separate MORSE analysis to obtain the radiation current through a 200-cm-radius disk located between the igloo and the injector.

The neutral-beam injector was modeled in the detail appropriate for each calculational method. The configurations and compositions of the injector were taken from engineering drawings and all of the important components were modeled. These included the cryogenic pumping panels, calorimeter, bending magnet, ion dump, ion neutralizers, and source guns. For the 2-D analysis, the injector was represented in cylindrical r-z geometry with symmetry about the axis of injection.

All of the calculations were carried out using a 35n, 21k energy group, P₀ expanded transport cross-section library. These data were obtained by collapsing the BLC-37 (ref. 5) [ENDF/B-IV] data set. The nuclear heating rates were estimated using neutron and gamma-ray flux-to-kerma factors generated by MACK⁵ and SMUG,⁶ respectively.

The neutron, gamma-ray, and total heating rates at four locations in the cryogenic pumping panels that line the sides of the neutral-beam injector are given in Table 6.26.1. Locations A and B are at the front of the injector and closest to the duct opening and locations C and D are toward the rear of the assembly in the shadow of the bending magnet. The heating rate data are given for stainless-steel type 316 and are normalized to 7×10^{17} n/s. The entries from the 3-D analysis correspond to the contributions to the heating rates from the radiation streaming from the injector port, from the radiation leaking through the igloo wall, and the sum of these. The values given in parentheses below each entry are the fractional standard deviations associated with these data. The

TABLE 6-1
Calculated instantaneous heating rates in SS-316 at all four locations in the cryopanel

Location	DOT	MORSE	DOT/MORSE
Point A	1.2E-01	1.1E-01	1.1
Point B	1.2E-01	1.1E-01	1.1
Point C	1.2E-01	1.1E-01	1.1
Point D	1.2E-01	1.1E-01	1.1

Notes: 1. The heating rates are in units of 10^4 W/m².
2. The DOT and MORSE results are for the 2-D analysis.
3. The DOT/MORSE ratio is the range of agreement among the results obtained from the two calculations.

results given for the 2-D analysis include both contributions. The symmetry of the 2-D representation does not allow for the proper orientation of the cryopanel surfaces relative to the igloo wall, thus the DOT results for points A and B and for points C and D are the same. The MORSE/DOT ratios indicate the range of agreement among the results obtained from the two calculations.

The calculated instantaneous heating rates in SS-316 at all four locations in the cryopanel are small. The agreement among the results obtained using the two calculational procedures is reasonable considering the approximation used in the 2-D analysis and the inability to account for the asymmetry of the geometry in the 2-D procedure. Data obtained for the heating rates in aluminum, copper, and zeolite, which are also constituents of the cryopanel, indicate similar trends. The heating rates were also calculated in the injector superstructure, the bending magnet, and in the ion guns at the rear of the injector.

The neutron and gamma-ray energy distributions obtained at the various locations were also calculated and will be presented.

*Research sponsored by DOE Office of Fusion Energy, Division of Development and Technology, Computer Sciences Division.

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6.27. DESIGN CALCULATIONS FOR A NEUTRON COLLIMATOR FOR THE TFTR*

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Report of ORNL Am. Nucl. Soc. 17, 137-138, 1977; also published in: ANL-77-10, August 1976, and submitted for Journal publication.

The tokamak fusion test reactor (TFTR) being constructed at Princeton University will produce neutrons by D-T reactions and measurements of the energy distribution of these neutrons will be used as a plasma diagnostic. That is, the measured energy distribution of the neutrons will be used to infer such quantities as the temperature and drift motion of the interacting ions. To be most useful, the neutron diagnostic information must refer to a reasonably small region of the overall plasma volume and this means that a neutron collimator that defines the volume from which the measured neutrons are emitted must be available. Such a collimator for TFTR application is being designed at the Oak Ridge National Laboratory and in this paper some of the design calculations are presented and discussed.

The collimation problem considered here is somewhat different from the usual collimation problem in that a very large extended 14-MeV source rather than a small (point) source is used, and source rather than field definition is of primary concern. Because of these differences, the information in the literature is not very directly applicable but some information is available. Experimental measurements have been obtained for 14-MeV neutrons by Green and Major,¹ for 15-MeV neutrons by Maruyama and Bouts,² and for a variety of energies by Glasgow et al.³ Extensive numerical investigation of collimator design for fission reactor core mapping has been performed by Straker,⁴ and a multislot collimator has been designed and tested at the TREAT reactor by De Volpi et al.⁵ A variation of the Glasgow collimator has been used for experiments with 2.5-MeV neutrons at the Princeton Large Torus.⁶

Discrete ordinates transport calculations have been completed for several right circular cylindrical stainless collimators using a 14-MeV neutron isotropic 30-cm radius disk source to approximate a plasma volume source. The collimators considered in these survey calculations consisted of 3-cm-diam air columns centrally located in 60-cm-diam stainless-steel cylinders that varied in length from 30 to 180 cm. The source collimator-detector configurations consisted of the uniform disk source located 60 cm from the entrance of each collimator and a 6-cm-diam x 6-cm-long right circular cylinder detector located at various distances from the exit of each collimator. The configurations were analyzed with the first collision source code GRTUNCL⁷ and the two-dimensional (2-D) discrete ordinates code DOT III⁸ using a highly biased (150 angle) angular quadrature. The cross sections used in this study were obtained from a 37-neutron 21-gamma-group F₄ scattering library.⁹

The portion of the source on which measurements are desired is hereinafter called the "resolved" source. The determination of the direct neutron flux (uncollided flux from the resolved source) and the background neutron flux (scattered flux from the resolved source and total flux from the remainder of the source) and the prompt photon flux at various detector locations requires four separate calculations. The required calculations are: a GRTUNCL calculation using only the resolved source (the resolved source was taken to be the 5-cm-diam central portion of the disk source) to obtain the uncollided flux and first-collision source distribution, a DOT calculation using the GRTUNCL first-collision source to obtain the scattered neutron flux and total photon flux due to the resolved source, a second GRTUNCL calculation using the remaining source, and a final DOT calculation using the second GRTUNCL first collision source to obtain the background neutron and photon fluxes due to the remaining source.

The ratio of the source group background flux to the direct flux from the resolved source is shown in Fig. 6.27.1 as a function of detector location. Ratios > 2 can be obtained for collimators 120 cm thickness as long as the detector is located more than 350 cm from the source. Both the scattered and uncollided components of the 14-MeV background are high for the 30-cm collimator. Its length is insufficient to attenuate 14-MeV neutrons that originate from the unviewed portion of the source. The 60-, 120-, and 180-cm collimators are effective attenuators as exhibited by the exponential behavior for detectors located near their exits. This behavior is due to the attenuation of uncollided neutrons that originate from the "viewed" portion of the source that lies

Fig. 6.27.1. Ratio of source group background to direct flux as a function of detector distance for various thickness collimators.

outside of the resolved source area. However, as the source-detector distance increases, the 14-MeV background becomes dominated by neutrons that scatter in the material immediately adjacent to the air column. This scattered component of the background renders the 60-cm collimator unacceptable. An increase in collimator length decreases this component of the background. However, this scattered contribution is also due to neutrons that originate primarily from the "viewed" portion of the source that lies outside of the resolved source area and thus depends heavily on the size of the collimator opening and the source-detector distance rather than collimator length. In addition, further increases in collimator length reduce the direct flux from the resolved source by increasing the minimum source-detector distance and thus do not significantly lower the background ratio.

* Research sponsored by Princeton University.

† Instrumentation and Controls Division.

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6.28. NEUTRON-PHOTON MULTIGROUP CROSS SECTIONS FOR NEUTRON ENERGIES ≤ 60 MeV*

R. G. Alsmiller, Jr. J. Barish†

(Abstract of ORNL/TM-6466, August 1978; also paper to be published in Nuclear Technology)

Multigroup cross sections (47n-groups, 21γ-groups) in ANISN format for neutron energies from thermal to 60 MeV and for the elements H, ^{10}B , ^{11}B , C, O, Si, Ca, Cr, Fe, and Ni are described. A P₅-Legendre expansion is used at energies ≥ 14.9 MeV and a P₃-Legendre expansion is used at energies ≤ 14.9 MeV.

Below 14.9 MeV the cross sections are from the Radiation Shielding Information Center's fusion energy cross section library. Above this energy differential elastic scattering cross section data from optical model calculations are used and differential nonelastic scattering data from the intranuclear-cascade-evaporation model are used.

Calculated results of the dose equivalent vs depth in the shield from a point isotropic source at the center of a spherical shell (366 cm thick) heavy-concrete (density = 3.6 g cm^{-3}) shield are presented. The energy distribution of the source neutrons is approximately that from a Li(D,n) neutron radiation damage facility.

*Research sponsored by Westinghouse Hanford Engineering Development Laboratory.

†Computer Sciences Division.

6.29. THE NUCLEAR PERFORMANCE OF MOLTEN SALT FUSION-FISSION SYMBIOTIC SYSTEMS FOR CATALYZED DD AND DT PLASMAS*

M. M. H. Ragheb‡ R. T. Santoro J. M. Barnes‡ M. J. Saltmarsh‡

(Abstract of ORNL/TM-6560, to be published)

The nuclear performance of a fusion-fission hybrid reactor having a molten salt composed of Na-Th-F-Be as the blanket fertile material and operating with a catalyzed DD plasma is compared with a similar system utilizing a Li-Th-F-Be salt and operating with a DT plasma. The production of fissile fuel via the ^{232}Th - ^{233}U fuel cycle was considered on the basis of its nonproliferation aspects. The calculations were performed using one-dimensional discrete ordinates methods to compare neutron balances, fuel production rates, energy deposition rates, and the radiation damage in the reactor structure. The results indicate that the Na salt in conjunction with the catalyzed DD plasma represents a viable alternative to the Li salt and DT plasma. In a reactor consisting of a 42-cm-thick salt compartment followed by a 40-cm-thick graphite reflector, the Na salt-catalyzed DD system exhibits a higher fissile nuclide production potential via Th(n,γ) reactions (0.880 reactions/source neutron) than the Li salt-DT system (0.737 reactions/source neutron) without the additional complication of tritium production. Also the blanket energy multiplication ratios and radiation damage rates are more favorable in the Na salt-catalyzed DD system.

*Research sponsored by DOE Office of Fusion Energy, Division of Development and Technology.

†University of Wisconsin.

‡Computer Sciences Division.

§Fusion Energy Division.

6.30. NUCLEAR PERFORMANCE CALCULATIONS FOR THE ELMO BUMPY TORUS REACTOR (EBTR) REFERENCE DESIGN*

R. T. Santoro J. M. Barnes†

(Abstract of ORNL/TM-6026, December 1977)

The nuclear performance of a fusion-fission hybrid reactor having a molten salt composed of Na-Th-F-Be as the blanket fertile material and operating with a catalyzed DD

plasma is compared to a similar system utilizing a Li-Th-F-Be salt and operating with a DT plasma. The production of fissile fuel via the ^{232}Th - ^{233}U fuel cycle was considered on the basis of its potential nonproliferation aspects. The calculations were performed using one-dimensional discrete ordinates methods to compare neutron balances, fuel production rates, energy deposition rates, and the radiation damage in the reactor structure. The results indicate that the Na salt in conjunction with the catalyzed DD plasma represents a viable alternative to the Li salt and DT plasma. In a reactor consisting of a 42-cm thick salt compartment followed by a 40-cm thick graphite reflector, the Na-salt catalyzed-DD system exhibits a higher fissile nuclide production potential via $\text{Th}(n,\gamma)$ reactions (0.880 reactions/source neutron) than the Li salt-DT system (0.737 reactions/source neutron) without the additional complication of tritium production in the blanket. A 1000 MW(e) hybrid reactor for a once-through cycle is estimated to be able to support 14 fission reactors of the same power for a DD system and can support about 8 reactors for a DT system.

* Research sponsored by DOE Office of Fusion Energy, Division of Development and Technology.

[†] Computer Sciences Division.

6.31. MONTE CARLO ANALYSIS OF THE EFFECTS OF A BLANKET-SHIELD PENETRATION ON THE PERFORMANCE OF A TOKAMAK FUSION REACTOR*

R. T. Santoro J. S. Tang[†] R. G. Alsmiller, Jr. J. M. Barnes[†]

(Abstract of ORNL/TM-5874, May 1977, and Nucl. Tech. 37, 65-72, 1976)

Adjoint Monte Carlo calculations have been carried out using the three-dimensional radiation transport code, MORSE, to estimate the nuclear heating and radiation damage in the toroidal field (TF) coils adjacent to a 28 x 68 cm² rectangular neutral beam injector duct that passes through the blanket and shield of a D-T burning Tokamak reactor. The plasma region, blanket, shield, and TF coils were represented in cylindrical geometry using the same dimensions and compositions as those of the Experimental Power Reactor. The radiation transport was accomplished using coupled 35-group neutron, 21-group gamma-ray cross sections obtained by collapsing the DLC-37 cross-section library. Nuclear heating and radiation damage rates were estimated using the latest available nuclear response functions. The presence of the neutral beam injector duct leads to increases in the nuclear heating rates in the TF coils ranging from a factor of 3 to a factor of 196 depending on the location. Increases in the radiation damage also result in the TF coils. The atomic displacement rates increase from factors of 2 to 138 and the hydrogen and helium gas production rates increase from factors of 11 to 7630 and from 15 to 9700, respectively.

* Research sponsored by DOE Office of Fusion Energy, Division of Development and Technology.

[†] Computer Sciences Division.

HIGH-ENERGY PHYSICS PROGRAM

6.32. PARTICLE FLUXES AND ENERGY DEPOSITION IN INFINITE HOMOGENEOUS AIR DUE TO PROTONS OF ENERGY 1-, 5-, 10-, and 20-GeV*

T. A. Gabriel B. L. Bishop[†] J. D. Amburgey[†] J. Barish[†]

(Abstract of ORNL/TM-8107, December 1977)

Calculations have been carried out to determine the spatial dependence of the energy deposition and the spatial and energy dependence of particle fluxes in infinite homogeneous air without hydrogen due to protons of energy 5-, 10-, and 20-GeV. The fluxes are broken down into nine particle types. The energy deposition is divided among

seven categories. The spatial dependence of the energy deposition is also presented for 1- and 10-GeV protons incident on infinite homogeneous air with hydrogen. The spatially integrated production of various residual nuclei is included.

*Research sponsored by DOE Office of High Energy and Nuclear Physics, Division of High Energy Physics.

+Computer Sciences Division.

6.33. PHOTON DOSE RATE FROM INDUCED ACTIVITY IN THE BEAM STOP OF A 400-GeV PROTON ACCELERATOR*

R. G. Alsmiller, Jr. T. A. Gabriel J. Barish⁺

(Abstract of ORNL/TN-6959, February 1976, and of Nucl. Instrum. Methods 155, 399-406, 1978) C

Calculated results are presented of the photon dose rate from the induced activity produced in the beam stop of a high-energy proton accelerator. Incident proton energies of 400 GeV, 200 GeV, and 30 GeV are considered. In each case the photon dose rate is given as a function of radius in the beam stop and as a function of time after the beam is turned off for assumed operating times of 90 days and 10 years.

*Research sponsored by DOE Office of High Energy and Nuclear Physics, Division of High Energy Physics.

+Computer Sciences Division.

6.34. CALCULATED HADRONIC TRANSMISSION THROUGH IRON ABSORBERS*

T. A. Gabriel B. L. Bishop⁺

(Abstract of ORNL/TN-6182, March 1976, and of Nucl. Instrum. Methods 166, 91-92, 1978)

The transmission of hadrons through iron absorbers is investigated for a variety of experimental configurations, for two incident particle types (protons and charged pions) and for a range of incident energies (0.9 GeV/c to 100 GeV). The results obtained are consistent with available experimental data.

*Research sponsored by DOE Office of High Energy and Nuclear Physics, Division of High Energy Physics.

+Computer Sciences Division.

6.35. URANIUM LIQUID-ARGON CALORIMETERS: A CALCULATIONAL INVESTIGATION*

T. A. Gabriel

(Abstract of ORNL/TN-6769, June 1977, and of Nucl. Instrum. Methods 160, 146-168, 1978)

A detailed calculational study has been performed to determine the response of uranium liquid-argon calorimeters to high-energy (<20 GeV) incident protons. The results obtained are consistent with the recently published experimental data of Fabjan et al. This experimental data and the calculations indicate that "nuclear fluctuations," which limit the resolution of calorimeters containing ordinary materials, can be effectively reduced by the fissioning and capture characteristics of ²³⁸U.

*Research sponsored by DOE Office of High Energy and Nuclear Physics, Division of High Energy Physics.

6.36. CALOR: A MONTE CARLO PROGRAM PACKAGE FOR THE DESIGN AND ANALYSIS OF CALORIMETER SYSTEMS*

T. A. Gabriel J. D. Amburgey[†] B. L. Bishop[†]

(Abstract of ORNL/TM-5619, April 1977)

CALOR is a Monte Carlo program package, written in FORTRAN IV, that is designed to assist experimentalists in evaluating and analyzing different types of calorimeter systems that are used in many high-energy physics experiments to determine the energy and direction of incident hadrons, leptons, and photons. This code package is intended to be used with the code package HETC which supplies needed information on the transport of hadrons and on the spatial location of the electromagnetic source energy. Input and output for a sample problem are presented.

*Research sponsored by DOE Office of High Energy and Nuclear Physics, Division of High Energy Physics.

[†]Computer Sciences Division.

6.37. HIGH ENERGY ($E < 1000$ GeV) INTRANUCLEAR CASCADE MODEL FOR NUCLEONS AND PIONS INCIDENT ON NUCLEI AND COMPARISONS WITH EXPERIMENTAL DATA*

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[Abstract of Phys. Rev. C 17(4), 1382-1384, 1978; also ORNL/TM-4710, August 1977]

An intranuclear cascade model for reactions of pions and nucleons with complex nuclei that should cover the energy range from about 50 MeV to about 1000 GeV has been developed. The model includes the effect of a diffuse nuclear surface, the Fermi motion of the bound nucleons within the nucleus, the exclusion principle, a local potential for nucleons, a localized reduction in the density of the nucleus during the development of the cascade, and the sequencing of the events correctly with time. Theoretical results from the model are compared with experimental data over the energy range ~ 3 -1000 GeV. Within the cascade model, the equations that are used to represent the pion multiplicity from the reactions of nucleons and pions with the individual nucleons of the nucleus underestimates the number of shower particles produced by these individual nucleon interactions. In spite of this the predicted number of shower particles escaping from the nucleus, for 100-GeV nuclear interactions, is overestimated by about 25% for light nuclei to about 60% for heavy nuclei. The agreement for the number of escapes from the nucleus is somewhat better at lower interaction energies (~ 10 -20 GeV). Some of the major trends in the observed high energy data are predicted quite well. These include the energy independence of (a) the number of black tracks produced at interaction energies above 5 GeV, (b) the number of shower particles produced above 100 GeV, and (c) the radionuclides produced above 10 GeV. Other trends that are predicted fairly well are the small mass dependence of the shower particle multiplicity (factor of 2 increase predicted from carbon to lead; factor of 1.5 measured), and the change in the angular dependence of the multiplicity with mass. In absolute comparisons, the predicted total reaction cross sections are in good agreement with experimental data, and the cross sections for the production of radionuclides are in fair agreement.

*Research sponsored by DOE Office of High Energy and Nuclear Physics, Division of High Energy Physics.

[†]Computer Sciences Division.

6.38. THE INTRANUCLEAR-CASCADE-EVAPORATION MODEL FOR NONELASTIC NUCLEAR COLLISIONS AND ITS APPLICATION IN BIOMEDICAL AND MFE RESEARCH*

T. A. Gabriel R. G. Alsmiller, Jr.

(Abstract of paper presented at Symposium on Neutron Cross Sections from 10 to 40 MeV, Brookhaven National Laboratory, Upton, New York, May 3-5, 1977; Brookhaven National Laboratory Report BNL-NCS-50681, 515-532, July 1977)

The intranuclear-cascade-evaporation model (ICEM) of nuclear reactions is a semi-classical approach used to describe in absolute terms continuum state transitions induced by a variety of incident particles including neutrons, protons, pions, heavy ions, etc. The emphasis here will be on incident protons and neutrons. The assumptions of the model are most applicable to high-energy (>100 -MeV) particle-induced nuclear reactions, but comparisons between calculated and experimental nonelastic total and differential cross-section data at lower energies (~ 20 -100 MeV) have shown reasonable agreement. Due to this agreement and to the fact that above 20 MeV there is a lack of experimental data, this model has often been used to generate the differential nuclear reaction data for a variety of applications. For example, ICEM is currently incorporated into the Monte Carlo high-energy transport code (HETC) that has been and is being used extensively and successfully for a variety of radiation transport problems, and data obtained from ICEM has often been used in the discrete ordinates codes ANISN and DOT. This paper gives a brief description of ICEM and compared results obtained with the model with experimental cross-section data. Transport calculations carried out using differential cross-section data obtained with ICEM are also presented and compared with experimental data. These include a variety of calculations of biomedical interest and radiation damage calculations of interest in magnetic fusion energy research.

*Research sponsored by DOE Office of Fusion Energy, Division of Development and Technology.

ACCELERATOR BREEDING STUDIES

6.39. MEDIUM-ENERGY NUCLEON-MESON TRANSPORT CALCULATIONS AND ELECTRONUCLEAR FUEL PRODUCTION*

R. G. Alsmiller, Jr. J. Barish D. E. Bartine T. J. Burns T. A. Gabriel

(Reprint of Trans. Am. Nucl. Soc. 28, 756-757, 1977)

The concept of converting fertile material to fissile material by using medium-energy (~ 1 GeV) protons incident on a suitable target-blanket assembly has recently been reconsidered and a variety of target-blanket designs have been proposed.¹⁻³ In this paper the results of medium-energy transport calculations carried out to evaluate the nuclear performance of some of these designs are presented and discussed.

All of the results presented here were obtained by coupling the high-energy transport code HETC⁴ with the low-energy transport code MORSE.⁵ All charged particles and neutrons with energy >15 MeV are transported with HETC, and neutrons with energy <15 MeV are transported with MORSE. The differential particle production data from nucleon-nucleus and pion-nucleus collisions in HETC are obtained from the intranuclear cascade model of nuclear reactions as implemented by Bertini.⁶ Calculated results obtained with HETC have been compared with a variety of experimental data, and, in general, good agreement has been obtained.⁷ In particular, calculations have recently been carried out and compared with experimental data⁸ for the number of neutron captures in water when 960-MeV protons are incident on one face of a cylindrical depleted uranium target (61 cm long, 10.2 cm in diameter) that is surrounded by a large water region. The experimentally measured and calculated number of neutron captures in the water is 32.3 ± 1.6 and 30.4 ± 0.6 , respectively; thus, the calculated and experimental results are in good agreement.

In Fig. 6.39.1 one of the target-blanket configurations that has been considered at ORNL is shown. The 1-GeV proton beam is incident on a sodium target surrounded by a sodium-cooled blanket. Regions III, IV, V, and VI are composed of 35 vol % metal, 50 vol % sodium, and 15 vol % cladding. Region VII is composed of 52.5 vol % metal, 37.5 vol % sodium, and 10 vol % cladding. In Regions III, IV, V, VI, and VII the density of the homogenized metal and cladding is 13.3 g/cm^3 . The metal in Region III is composed of 78 at. % ^{232}Th , 15 at. % ^{238}U , and 7 at. % ^{239}Pu . The metal in Region IV is composed of 78 at. % ^{232}Th , 12 at. % ^{238}U , and 10 at. % ^{239}Pu . The metal in Regions V, VI, and VII is all ^{232}Th . The cladding is composed of 80 at. % V and 20 at. % Ti.

Fig. 6.39.1. Schematic diagram of target-blanket configuration. The composition of the various regions is given in the text.

In Table 6.39.1 the calculated number of neutron captures and fissions in various materials and regions is given for the target-blanket configuration shown in Fig. 6.39.1. Also shown in the table is the energy deposition in the various regions. Note that the results in the table have been integrated over the various regions and the volumes of the regions are not equal. The number of neutron captures in ^{232}Th in Regions III and IV is comparable, but because of the higher number of fissions in Region III the energy deposition in Region III is considerably larger than that in Region IV and in Region I. It is also interesting to note that since 538 MeV is deposited in Region I, a large fraction of the 1-GeV incident proton energy is carried by particles that leave Region I.

Table 6.39.1. Neutron captures, fissions, and energy deposition in various elements and regions of the target blanket configuration shown in Fig. 6.39.1

(All results are given per incident 1-GeV proton)

	Region								Sum Over All Regions
	I	II	III	IV	V	VI	VII	VIII	
Neutron captures in ^{232}Th			12.8	13.7	4.4	2.3	4.0		37.4
Neutron captures in ^{238}U			2.1	1.7					3.8
Neutron captures in ^{239}Pu			1.5	0.2					1.7
Neutron captures in all elements other than ^{232}Th , ^{238}U , and ^{239}Pu	0.5	0.0	1.1	1.2	0.3	0.2	0.1	0.7	4.1
Total captures in all elements	0.5	0.0	17.5	17.0	4.7	2.5	4.1	0.7	47.0
Fissions in ^{232}Th			0.77	0.55	0.26	0.16	0.07		
Fissions in ^{238}U			0.38	0.30					
Fissions in ^{239}Pu			5.8	1.0					
Total fissions in all elements			7.0	1.8	0.26	0.16	0.07		9.3
Energy deposition (MeV)	538	3	1457	509	100	74	19	14	2714

Calculated results similar to those shown in Table 6.39.1 have been obtained for a variety of target-blanket configurations and will be presented.

* Research sponsored jointly by DOE Office of Basic Energy Sciences, Division of Nuclear Sciences and DOE Office of Nuclear Energy Programs, Division of Nuclear Research and Applications.

† Computer Sciences Division.

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6.40. NUCLEON-MESON TRANSPORT CALCULATIONS AND THE NUCLEAR PERFORMANCE OF TARGET-BLANKET ASSEMBLIES FOR ELECTRONUCLEAR FUEL PRODUCTION*

R. G. Alsmiller, Jr

(Abstract of seminar paper, Chalk River Nuclear Laboratories, Chalk River, Ontario, July 28, 1977)

Many, but certainly not all, of the problems in the nuclear design of the target-blanket assemblies for electronuclear fuel production arise from the fact that high-energy (≥ 1 GeV) particles must be considered. There is relatively little experimental information available on particle production from nucleon-nucleus and pion-nucleus collisions at energies ≥ 20 MeV and thus much reliance must be placed on theoretical differential cross section data. The only readily available source of such data is the intranuclear-cascade-evaporation model of nuclear reactions. This model will be discussed and comparisons between calculated results and experimental data will be given.

The intranuclear-cascade-evaporation model has been incorporated into a nucleon-meson transport code HETC. The physical interactions included in this transport code will be described and a variety of comparisons between calculated results and experimental data will be presented. The code HETC has been used to study the nuclear performance of several target-blanket assemblies that have been proposed for electronuclear fuel production and the results of these calculations will be discussed.

* Research sponsored by DOE Office of Basic Energy Sciences, Division of Nuclear Sciences.

6.41. PRELIMINARY REPORT ON THE PROMISE OF ACCELERATOR-DRIVEN NATURAL-URANIUM-FUELED LIGHT-WATER-MODERATED BREEDING POWER REACTORS*

Ehud Greenspan

(Abstract of ANS/NE-1181, February, 1978)

A new concept for a power breeder reactor that consists of an accelerator-driven subcritical thermal fission system is proposed. In this system an accelerator provides a high-energy proton beam which interacts with a heavy-element target to produce, via spallation reactions, an intense source of neutrons. This source then drives a natural-uranium-fueled, light-water-moderated and -cooled subcritical blanket which both breeds new fuel and generates heat that can be converted to electrical power. This report presents a general layout of the resulting Accelerator Driven Light Water Reactor (ADLWR), evaluates its performance, discusses its fuel cycle characteristics, and identifies the potential contributions to the nuclear energy economy this type of power reactor might make.

A light-water thermal fission system is found to provide an attractive feature when designed to be source-driven: The equilibrium fissile fuel content that gives the highest energy multiplication is approximately equal to the content of ^{235}U in natural uranium. Consequently, natural-uranium-fueled ADLWRs that are designed to have the highest energy generation per source neutron are also fuel-self-sufficient; that is, their fissile fuel content remains constant with burnup. This feature allows the development of a nuclear energy system that is based on the most highly developed fission technology available (the light water reactor technology) and yet has a simple and safe fuel cycle. ADLWRs will breed on natural uranium, have no doubling time limitation, and be free from the need for uranium enrichment or for the separation of plutonium.

It appears that ADLWRs could also be efficiently operated with thorium fuel cycles and with denatured fuel cycles. In addition, fissile-fuel-producing ADLWRs might work in tandem with LWRs to provide a nuclear power system that is fuel-self-sufficient and is free from the need for uranium enrichment or plutonium separation. It may also be possible to fuel ADLWRs with depleted uranium.

This preliminary evaluation of the performance of ADLWRs indicates that the net overall efficiency for the conversion of the fission energy into electricity of a power system that is based on ADLWRs might be comparable to that of a power system based on the symbiosis of accelerator-driven fuel factories and conventional LWRs but smaller than that of a power

system based on the symbiosis of accelerator fuel factories and advanced converter reactors. Compared with the blankets of other accelerator-driven fission systems, the blankets of ADLWRs will have significantly lower radiation damage rates, lower power density gradients, and lower rates of variation of energy multiplication with burnup.

It is concluded that it is possible, but quite uncertain at this point, that ADLWRs could be developed to provide viable power reactors. The major uncertainties are associated with the successful development of the accelerator and target assemblies, as well as with the attainment of a high enough overall plant efficiency. In view of the attractive fuel cycle characteristics offered by the ADLWRs and the useful options they may provide for the development of the nuclear energy economy, it is recommended that the feasibility of source-driven subcritical thermal power breeder reactors be thoroughly investigated.

* Research sponsored by DOE Office of Basic Energy Sciences, Division of Nuclear Sciences.

† Visiting scientist from Nuclear Research Center-Negev and Ben-Gurion University of the Negev, Israel.

MISCELLANEOUS STUDIES

6.42. RADIATION TRANSPORT IN EARTH FOR NEUTRON AND GAMMA RAY AIR-OVER-GROUND SOURCES*

R. A. Lillie R. T. Santoro

(Abstract of ORO-71-0686, to be published)

Two-dimensional discrete ordinates methods have been used to calculate instantaneous dose rates and neutron and gamma ray fluences as a function of depth in earth for air-over-ground point sources at heights of 1, 67 and 800 yards. The nuclear responses due to primary neutrons and gamma rays and secondary gamma rays are presented as line integrals at a distance of 433 yards from the source-ground axis. The radiation incident on the earth's surface was transported through an earth-only and an earth-concrete model containing three feet of borated concrete beginning two feet below the earth's surface to obtain fluence distributions to a depth of 10 feet. The inclusion of borated concrete does not significantly reduce the nuclear responses, and, in all cases, the total instantaneous dose rates and fluences are dominated by secondary gamma rays for depths greater than two feet.

* Research sponsored by Department of the Army.

6.43. DEVELOPMENT OF A MID-HEAD TISSUE DOSE RESPONSE FUNCTION*

D. K. Trubey D. E. Bartine J. V. Pace III[†] J. R. Knight[†]

(Reprint of Trans. Am. Nucl. Soc. 30, 662-663, 1970; also summary of ORO-71-0684, to be published)

The objective of this investigation was to define realistic tissue neutron and gamma-ray dose response functions for the mid-head position for military applications.

Accurate response functions are required for the evaluation of weapons effects in tactical situations since they directly influence the effective radius of the weapon. Tissue dose response may be divided into two types, free-in-air kerma (kinetic energy deposited in tissue due to a single collision), and dose-in-phantom (multiple collision energy deposition in a tissue phantom). Free-in-air kerma does not account for the interaction of the radiation field with the phantom, while conventional dose-in-phantom responses¹⁻³ score the maximum energy deposition within the phantom, which varies with the

energy of the incident particle so that the resulting response does not correspond to any position within the phantom. The technique used in this investigation was to calculate the radiation transport in a two-dimensional (2-D) adjoint model by the discrete-ordinates code DOT 3.5 (ref. 4). An SeP₂ approximation was used. The adjoint source was taken to be the tissue kerma response in a 1-cm-radius region in the center of the head. The procedure was to then calculate in the adjoint mode the current on the surface of the phantom model. The adjoint leakage gives the response function for scoring an isotropic incident radiation field, and the exit adjoint angular flux gives the response for angular-dependent incident radiation. The head models employed were basically those suggested by Warshawsky⁵ for use in correlating radiation dose received in monkeys during experiments at the Armed Forces Radiological Research Institute with personnel dose received in a nuclear weapons environment.

The geometric model chosen to represent the human head and torso was two coaxial cylinders with radii of 7.17 and 15 cm and heights of 14.34 and 50 cm. A layer of bone, 0.8 cm thick, enclosed the head. The head was assumed to consist of brain tissue, and the torso consisted of tissue called "standard man." The cross sections were those of DLC-31 (DPL-1)/REWG1, a coupled cross-section set with 37 neutron groups and 21 gamma-ray groups⁶ based on ENDF/B-IV. The values of kerma for the necessary nuclides are also available in DLC-31. These were used as the adjoint source to calculate "dose" to whole brain, the mid-brain, and the medulla oblongata. For each case, both neutron and gamma-ray adjoint runs were made. The neutron runs provide the energy deposition from neutron reactions. The gamma-ray runs provide energy deposition data for incident gamma rays and for incident neutrons; i.e., energy deposition from secondary gamma rays produced in the phantom.

The resulting neutron response function values for mid-head dose are given in Fig. 6.43.1. Also included in Fig. 6.43.1 is the Snyder-Auxier tissue dose response for comparison. The latter gives larger values primarily because the function represents the maximum dose at locations near the surface. For energies above about 1 MeV, the dose is due mainly to neutron interactions. Below about 0.1 MeV, the dose is due mainly to secondary gamma rays and therefore does not vary much with incident-neutron energy.

To determine the difference in dose, compared to the Snyder-Auxier function, the two responses were used to calculate dose in an air-over-ground geometry for fission and fusion sources. Calculation of the flux was performed by Pace et al.,⁷ and the results were available on tape. The calculations were performed using a 22/18 group structure, and therefore, the response function had to be converted to this structure. The dose based on the Snyder-

Fig. 6.43.1. Neutron conversion factor for fluence to dose (rad·cm²) as a function of energy.

Auxiliary response and the dose based on the response function of this study, and their ratio, were calculated as function of distance from the source. The values of the ratio are almost constant to a range of about 1500 m. The value is 4.0 ± 0.1 over almost the entire range. The secondary gamma-ray dose (gamma rays from neutron interactions in the air) ratio is 1.48 ± 0.02 . In the case of a fusion source, the neutron dose ratio varies more. It is about 1.9 at a close range, about 2.5 at 600 m and about 3.0 beyond 1300 m. The gamma-ray dose ratio is nearly a constant equal to 1.46 ± 0.02 .

A report giving additional detail and comparisons of other related response functions is in preparation.

* Research sponsored by Defense Nuclear Agency.

* Computer Sciences Division.

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6.44. MOLTEN-SALT REACTORS FOR EFFICIENT NUCLEAR FUEL UTILIZATION WITHOUT PLUTONIUM SEPARATION*

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(Research of ORNL-TM-4842, August 1977)

Molten-salt reactors (MSRs), because of the fluid nature of the fuel, appear to provide an attractive approach to efficient fuel utilization in the Th-U cycle as well as a means for limiting the availability of plutonium and the general proliferation risks associated with nuclear power generation.

High-enrichment ^{235}U systems could, in principle, be operated with positive breeding gains to effectively eliminate plutonium as a nuclear fuel. However, such systems would be proliferation sensitive. Concept modifications (short of denaturing the uranium fuel) can be conceived to enhance the proliferation resistance of high-enrichment MSRs, but it is doubtful that sufficient enhancement could be achieved to make the systems suitable for deployment other than at "secure" sites.

Denaturing the uranium in an MSR introduces some plutonium into the fuel cycle and generally degrades its breeding performance. Nevertheless, a denatured MSR with full-scale on-site fuel reprocessing appears to be capable of break-even breeding. In addition, the plutonium (most of which is consumed in situ) would be of poor quality and would never be isolated from all other undesirable nuclides. Thus, such systems would provide for efficient utilization of uranium resources in a proliferation-resistant environment while limiting the amount of plutonium (and transplutonium actinides) that would have to be handled as waste.

The development of commercial MSRs by early in the 21st century appears to be technologically feasible.

*Research sponsored by DOE Office of Nuclear Energy Programs, Division of Nuclear Research and Applications.

†Engineering Technology Division.

‡Chemical Technology Division.

6.45. MONTE CARLO SIMULATION OF TURBULENT ATMOSPHERIC TRANSPORT: COMPARISON WITH EXPERIMENTAL DATA*

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(Reprint of *Trans. Am. Nucl. Soc.* 27, 117-119, 1977; also summary of ORNL/TN-6291, March 1976, and of paper to be published in *Journal of Applied Meteorology*)

The Monte Carlo computer code SPOOR, written by C. W. Watson and S. Barr,¹ permits sample trajectories of individual tracer particles in the atmosphere to be followed. Each particle velocity is taken to be the sum of a stochastic turbulent component and the atmospheric mean wind velocity. The mean wind velocity field can, in principle, be an arbitrarily complex function of position and time. As originally formulated, SPOOR provides the motion of the centroid of a puff of airborne tracers released instantaneously from a fixed source and the time evolution of the puff shape. Following a suggestion of F. A. Gifford,² SPOOR has been modified to allow for continuous particle emission from a smokestack and comparisons between results obtained with the model and experimental data reported by Peterson³ have been obtained. These experimental data extend over distances of the order of 300 km and are therefore very suitable for testing the validity of the model.

If subscript j numbers the time step ($\Delta t = 15$ min) and t_0 is the starting time for a given particle, the basic equations of SPOOR for the particle velocity component, u_j , in any direction during the time step j may be written

$$u_j = u_0 + \delta u_j \quad (1)$$

$$u_j(r_j, t_j) = \bar{u}_j(r_j, t_j) + u'_j(r_j, t_j) \quad (2)$$

$$u'_j = u'_0 + \sum_{i=1}^{j-1} \delta u_i u'_i \quad (3)$$

$$R_j(n\Delta t) = \langle (u_j - \bar{u}_j)(u_{j+n} - \bar{u}_{j+n}) \rangle \quad (4)$$

$$= \sum_{j=1}^{j-1} \rho_{j,j} R_j[(n + j - j)\Delta t] \quad (5)$$

where

r_j = position vector of the particle at time t_j ,

$\bar{u}_j(r_j, t_j)$ = mean velocity component at r_j and t_j ,

$u_i(r_{ij}, t_{ij})$ = turbulent velocity component at r_{ij} and t_{ij} .

Δu_{ij} = random velocity component chosen at each time step from a Gaussian distribution with standard deviation $\sigma_{\Delta u}$.

$R_2(\xi)$ = Lagrangian correlation function for lag time ξ .

$\langle \rangle$ = average over all of the particles in a puff.

Equation (5) follows from Eqs. (1) through (4) and the fact that the expectation value of Δu_{ij} , i.e., $\langle \Delta u_{ij} \rangle$, is zero. The δ_{ij} 's are coefficients that are determined from Eq. (4) and the Lagrangian correlation functions. In SPOOR, the sums on the right of Eqs. (3) and (5) are terminated at three previous time steps. The δ_{ij} 's are independent of the mean wind field and the normalization of the turbulence power spectrum used in computing the correlation functions. In the work reported here the "medium" turbulence power spectrum of ref. 1 was used in the horizontal plane and no velocity correlations were considered in the vertical direction.

The boundary conditions used in SPOOR are reflections of particle velocities at two horizontal planes designated by $Z = Z_{\max}$ and $Z = Z_{\min}$. The experimental data show conditions of neutral stability and very few tracer particles reached altitudes of >1 km. With the customary assumption of zero vertical mean wind and $Z_{\max} = 1$ km, $Z_{\min} = 0.006$ km, it was found possible to reproduce the data at least approximately by using the standard deviation, $\sigma_{\Delta u}$, of the Gaussian velocity distribution in vertical direction to be 0.0012 km h^{-1} . The standard deviations of the random velocity distributions in the horizontal plane, $\sigma_{\Delta u_x}$ and $\sigma_{\Delta u_y}$, were taken to be 1.6 km h^{-1} as in ref. 1.

The horizontal mean wind fields, $\bar{u}$ and $\bar{v}$, were obtained from the hourly surface wind readings reported by 12 airports in the vicinity of the region over which the experimental data were taken. Interpolation in time and inverse square weighting with distance from the wind stations were used. Variations of speed with altitude were computed by use of the power law adjustment factor reported by Peterson.⁴ No change of wind direction with altitude was assumed.

Peterson made measurements on the gamma-ray intensities from the ^{41}Ar particles emitted into the atmosphere from a nuclear reactor at the Brookhaven National Laboratory. From these measurements he inferred the ^{41}Ar concentrations at various distances from Brookhaven. At a given distance x (measured east from Brookhaven) and at a given time t , Peterson reports the y -position (north), $y_{\max}$, of the maximum ^{41}Ar concentration and the lateral standard deviation, σ_{yy} , of the concentration with respect to $y_{\max}$.

The first moment, $\bar{y}$, and the second moment, σ_y^2 , of the calculated distributions are compared with the experimental data in Table 6.45.1. In obtaining the calculated results, the decay of the ^{41}Ar particles was taken into account. All of the calculated values in the table have been averaged over a time interval of 0.5 h, an x interval of 10 km, and a z interval of 0.33 km. The statistical errors of the calculated values are not given in the table because they are quite small ($<$ a few percent of the calculated values). The calculated $\bar{y}$'s are in fair agreement with the measured $y_{\max}$'s at most of the data points. The largest disagreement occurs at data points 9, 10, and 11, where the measured $y_{\max}$'s have rather large negative values. These large negative values correspond to positions over the ocean that are far from any wind stations and thus the discrepancy may be due to an inadequate knowledge of the mean wind field in the vicinity of these data points. The calculated σ_y^2 's are, in all cases, in very good agreement with the measured σ_{yy} 's. The agreement is very good even at those data points 9, 10, and 11, where the calculated $\bar{y}$ is in poor agreement with the measured $y_{\max}$. Comparisons between measured and calculated maximum concentrations as a function of position and time have also been obtained.

Table 6.45.1. Comparison of measured and calculated first and second moments in the north direction

Data Point	t^a (h)	x^b (km)	y_M (meas) (km)	$\bar{y}$ (calc) (km)	σ_{yM} (meas) (km)	$\sigma_{\bar{y}}$ (calc) (km)
1	12.5	15	7.4	1.1	0.85	0.75
2	12.6	82	17	6.9	1.9	3.9
3	12.9	106	15	8.6	7.0	6.3
4	13.2	205	24	23	9.8	10
5	15.0	290	23	20	5.7	3.7
6	15.5	268	5.5	15	9.2	9.5
7	16.3	266	28	22	15	11
8	16.8	197	41	36	8.5	8.3
9	17.6	158	-18	10	7.0	6.5
10	17.7	148	-20	3.0	6.0	5.9
11	17.8	122	-21	-5.7	4.0	4.7
12	18.0	94	0.0	-12	3.6	3.5

^at is the time measured from midnight.

^bx is distance measured east from the Brookhaven National Laboratory.

*Research sponsored by ORNL Exploratory Studies Program.

[†]Engineering Technology Division.

[‡]Computer Sciences Division.

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6.46. NUCLEAR ACCIDENT DOSIMETRY: CALCULATIONS AND COMPARISON WITH EXPERIMENTAL DATA*

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(Abstract of ORNL/TM-8401, Revised, August 1978, and of paper to be published in Health Physics Journal)

Calculated results, carried out by the method of discrete ordinates, of the absorbed dose received by personnel at various locations relative to the source in a simulated reactor criticality accident are presented and compared with experimental data. The approximate geometry of the room in which the "accident" took place is included in the calculations. The calculated and experimental data are in good agreement in all cases.

*Research sponsored by ORNL Exploratory Studies Program.

[†]Computer Sciences Division.

6.47. CALCULATED ATOM DISPLACEMENT AND GAS PRODUCTION RATES OF MATERIALS
USING A FUSION REACTOR FIRST WALL NEUTRON SPECTRUM*

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(Abstract of Nucl. Techn. 38, 427-433, 1978, and of ORNL/TM-5356, June 1977)

The displacement per atom and gas production rates have been calculated for a number of alloys and elements using a design neutron spectrum at the first wall of a fusion reactor. These rates can be combined for most alloys to yield the defect production rates, the parameters currently used to extrapolate available irradiation effects data to fusion reactor conditions. Calculated rates of atom displacement and hydrogen generation in stainless steels are relatively insensitive to recent changes in the nuclear data files and to neutron spectrum differences produced by slight reactor design changes. In contrast, the helium production rate is sensitive to these changes and to the exact alloy composition. Composition variation within the specification range for type 316 stainless steel can produce variations of $\pm 9\%$ in the helium generation rate.

*Research sponsored by DOE Office of Fusion Energy, Division of Development and Technology.

7. INFORMATION ANALYSIS AND DISTRIBUTION

7.0. INTRODUCTION

Since it was established in 1962, the Radiation Shielding Information Center (RSIC) has become a model of technology transfer, especially as a result of its technical activities associated with computer code and data exchange. In addition to RSIC, the Division now has several information analysis and data centers. The Biomedical Computing Technology Information Center (BCTIC) was started in 1975, and since then the Safety Analysis Computerized Reactor Data Base System (SACRD), the Central Reliability Data Organization (CREDO), the Reactor Safety Research Data Repository (RSRDR) (formerly Measured Data Repository), and a Technical Data Management Center (TDMC) have been established under DOE or NRC sponsorship to validate and standardize computing technology and to organize, manage, and disseminate selected data bases for use in NRC licensing and regulatory processes. Management of the work of building the SACRD and CREDO data bases resides in the Reactor Analysis, Safety and Reliability Group of the Division, while RSIC, BCTIC, RSRDR, and the new TDMC work is managed under the Technology Resources Group (see p. A-26).

Radiation Shielding Information Center (RSIC)

RSIC, now in its 17th year, continues to be our largest and most widely used information center. Established to promote the exchange of shielding technology, it acts as a resource base for both government and civilian agencies in the United States and in foreign countries and performs such diverse functions as providing bibliographic information; testing, assembling, and distributing computer codes; preparing, testing, and distributing multigroup cross-section libraries, both fine-group and broad-group; collecting and distributing other types of data bases; holding seminars to educate the shielding community on particular techniques, especially computer techniques; helping to establish shielding benchmark problems and shielding standards; and generally providing problem assistance to requestors.

The RSIC scope includes the physics of interaction of radiation with matter; radiation production, protection, and transport; radiation detectors and measurements; engineering design techniques; shielding materials properties; computer codes useful in research and design; and nuclear data compilations. The goals of RSIC are to function as a technical institute to provide information (bibliographic and other data, computer codes, technical advice) upon request; collect, evaluate, enrich, distill, and repackage information to extend the state of the art, bringing into the public domain technology more usable and more valuable than the sum of the input; and to initiate and effect research and development in areas of need that can be managed by RSIC personnel.

Results from some of the RSIC activities during the current reporting period have already been cited in Section 3.0, in particular, the maintenance of evaluated neutron cross-section files for the Defense Nuclear Agency and the development and validation, in collaboration with other Division staff members, of the 171-neutron and 36-gamma-ray group VITAMIN-C data library and of others derived from it. Other data activities have also been mentioned. RSIC has continued to participate in the work of the Cross Section Evaluation Working Group (CSEWG), with the Shielding Subcommittee presently chaired by an RSIC staff member. Other staff members are actively involved with the American Nuclear Society in developing radiation protection and shielding standards and standards for computer programming and computer code documentation.

In the area of computing technology, RSIC now makes available 450 computer code packages and 150 data packages. The RSIC Newsletter is currently mailed monthly to 1600 people. During FY 1978, more than 3000 letters/telephone calls of requests were processed by RSIC, resulting in 7600 separate activities required to satisfy the requests. A total of 133 (54 foreign) persons came for an orientation visit and/or to use the center's facilities.

Important highlights for the year include two seminar-workshops. These were the RSIC Seminar-Workshops on "Multigroup Nuclear Cross Section Processing," March 1978 and "Theory and Application of Sensitivity and Uncertainty Analysis," August 1978. The proceedings of the first (ORNL/RSIC-41) has been published and the proceedings of the second (ORNL/RSIC-42) are in press. Both seminar-workshops were well attended and this contributed materially to an advancement in the state of the art at a number of institutions.

To help in processing and filling the large number of requests received by RSIC and the other TRG centers and to better analyze our constituents, a computer-based record system (ADES) was initiated on the central DEC System 10. It is expected that this function will move to a minicomputer to be installed in early 1979.

CRNL-DWG 78-13254

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Fig. 1.0.1. RSIC information analysis, processing, and dissemination model.

The open code/data package concept (Fig. 1.0.2) as developed by RSIC, which promotes scientific scrutiny of the package and feedback of user revisions or improvements, has received wide recognition. With this concept, RSIC has been an effective technology

exchange center, with users providing much of the testing, validation, and additional improvements which continuously advance the state of the art. Users may find errors, make revisions and extensions appropriate to new applications, or implement the codes on new hardware or software systems. They may also write auxiliary codes or produce new data bases which enhance the usefulness of the package. Each new development is reflected in

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RSIC CODE / DATA PACKAGE

- (1) Packaging - Auxiliary routines, data libraries, and sample problem are collected and written on to magnetic tape (the master tape) - the code is run on local computers. Sample problem output is captured and entered on a separate file on the master tape. The tape is scanned, and described on a tape list which accompanies each package (see example attached). Documentation is added to complete the package.
- (2) A package is extended to include versions which have been converted to run on hardware different from that for which the code was designed - or is converted to a language different than that which the code was developed.
- (3) Availability is announced through the RSIC Newsletter, which is currently being mailed to more than 1600 persons, 74% of whom are in countries outside the U.S. (29 different nations).

Fig. 7.0.2. Open code data package concept.

the "open" code/data package in RSIC and is announced in the RSIC Newsletter which is distributed worldwide.

Biomedical Computing Technology Information Center (BCTIC)

BCTIC is sponsored by the Division of Human Health Studies of the DOE Office of Health and Environmental Research to provide a focal point for sharing computing technology in biomedicine and to assess and promote the state of the art. The Center serves biomedical research and development and medical clinical practice by collecting, organizing, processing, evaluating, packaging, and disseminating computing technology. The goal is the more effective use of computers with consequent cost savings.

Specifically, BCTIC serves as a clearinghouse for algorithms, computer programs, data, and interface designs pertinent to clinical and/or research biomedicine. The "open code package" concept is used - continual updates and revisions in programs are made as errors and incompatibilities are detected or as programs are extended, modified, or translated into other languages. Twenty-four code packages and seven data bases are currently available. These include clinical applications and research tools for statistical and other data analysis studies, for internal dosimetry, and for various simulations.

In support of its code development and exchange activities, BCTIC maintains an information storage and retrieval system containing bibliographic citations and abstracts accumulated through review of pertinent books, journals, and reports. Currently, the BCTIC data base contains over 7300 bibliographic references on biomedical computing.

The BCTIC Newsletter, which serves as a communication medium for the biomedical computing community, is currently issued bimonthly to 1500 people. The Newsletter highlights important developments in the field, surveys existing biomedical computer installations, notes upcoming meetings and events of interest, and provides a bibliography of recent biomedical computing literature.

A significant BCTIC accomplishment during the year was the publication of a "Review of Information Processing in Medical Imaging," ORNL/BCTIC-2, the work of many specialists in the field under the sponsorship of BCTIC, Vanderbilt University School of Medicine, and the FDA Bureau of Radiological Health. In addition, work continued on updating the directory of computer users in nuclear medicine, ORNL/BCTIC-1. A seminar-workshop, conducted in collaboration with the National Institutes of Health, in which the National Cancer Institute's SAAM Code System was featured, was well attended.

The BCTIC personnel actively promote hardware and software standards in biomedical computing including general computing standards activities of the American National Standards Institute's Technical Committee on Computer Interfaces (X3T9), the Task Group on Minicomputer Interfaces (X3T9.2), and the ANSI-10 Standards Subcommittee of the American Nuclear Society (computer programming and documentation practices).

Since it was established in 1975, BCTIC has logged 1877 communications. A review of the last six months gives the following statistics. A total of 427 communications were logged into BCTIC, resulting in several activities. Thirty-two requests for code/data packages were processed. In addition, responses were made by letter or telephone to 255 persons with inquiries concerning specific problems in biomedical computing; 36 documents were mailed; 5 updates were made to existing code/data packages; and 54 miscellaneous inquiries were processed. A total of 240 directory changes were made in the period and 5 visitors were received. Of the 427 requesters, 346 were domestic and 81 were foreign. The U.S. requesters represented 122 university medical schools and hospitals, 43 general medical centers and private practitioners, 41 national laboratories, 40 vendors, 26 veterans administration hospitals, 25 public and private hospitals, 7 armed forces medical groups, and 4 medical consultants. In addition, 17 requests were received from the national institutes of health, 10 from MJC officials/MUMPS development committee, 3 from the society of nuclear medicine, and 3 from the bureau of radiological health.

BCTIC, together with the activities preceding its founding, has exercised an influence within the national biomedical computing community considerably out of proportion to its size or its budget. There is some evidence to indicate a direct relationship between the existence of BCTIC and improved computing practices and software development

in nuclear medicine. This specific community has organized a Council on Computers within the Society of Nuclear Medicine, at BCTIC instigation, through which to share information and technology.

Although a DOE-sponsored review of the BCTIC operation gave positive findings and a strong recommendation for its continuance, the Division continues to question its relevance to its other programs. A decision on BCTIC's future will be made during this fiscal year.

Reactor Safety Research Data Repository (RSRDR)

A repository and distribution center for reactor safety research data, sponsored by the Division of Reactor Safety Research of the U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Regulatory Research, established in FY 1978, is now fully functioning. The Reactor Safety Research Data Repository (RSRDR) serves as the archival repository and response center of the NRC/RSR Data Bank System, which also includes Data Base Managers and a Data Bank Processing System located at the Idaho National Engineering Laboratory (INEL).

The NRC/RSR Data Bank System is designed to manage data pertinent to research investigating the loss-of-coolant accident for both pressurized water reactors and boiling water reactors. The data is in the form of raw measured data that has not been processed with the exception of conversion to engineering units and application of calibration factors. Data sources include experimental programs sponsored by NRC, other government agencies, private industry, and foreign governments. Specific examples include data from Semiscale, Loss of Fluid Test (LOFT), Two Loop Test Apparatus (TLTA), Thermal Hydraulic Test Facility (THTF), Full Length Emergency Cooling Heat Transfer Facility (FLECHT), Other Heat Transfer Loops, and Two Phase Pump Performance Facilities.

The repository maintains archival records and provides a distribution service. The stored data (currently 34 Semiscale and LOFT test data tapes) is available on demand to NRC contractors, to contributors to the Data Bank, and to other individuals or organizations working in the field of reactor safety.

Procedures and computer routines have been developed to make RSRDR information quickly available in hardware-independent, computer-readable form. A directory of persons interested in the data and its use is being compiled and a mechanism for sharing feedback from use of the data is being established.

A descriptive brochure, written guidelines for use of those requesting data, and an inventory of the data currently packaged for dissemination, including abstracts of reports describing the tests and resultant data, have been prepared. RSRDR was officially offered as an international technology resource at the NRC Sixth Water Reactor Safety Research Information Meeting in November 1978.

Technical Data Management Center (TDMC)

A Technical Data Management Center (TDMC) has been established to serve NRC licensing and regulation programs with goals and priorities determined by an advisory committee of NRC staff members chaired by the contract monitor within the Division of Technical Information and Document Control of the NRC Office of Administration. The objectives of the program are to define, develop, and maintain a center for the acquisition, analysis, packaging, documenting, and dissemination of technical data (numeric, methodology, computing technology) for use by NRC staff, licensees, contractors, and others engaged in related work. A further objective is to develop the methodology necessary to process the data for accurate analysis and for efficient retrieval of documented data upon request.

Current work in progress (FY 1979) includes program enhancement and documentation for radiological assessment (GASPAR and LADTAP, which calculate dose factors, and GALE, which computes effluents from PWR and BWR radwaste systems) and for the meteorological evaluation (XOQDOQ code) of effluent releases from nuclear power plants. The work also includes application of the "open code/data package concept" (Fig. 7.0.2) to the SCALE

(Standardized Computing-Analysis for Licensing Evaluation) System, methodology developed at ORNL to provide analytical capabilities (criticality safety, radiation shielding, heat transfer) to be applied to safety analyses of equipment and facilities proposed for licensing by NRC.

In response to an early NRC request, stand-alone computer codes expected to form a base of the SCALE System were collected, tested, packaged, and made available by RSIC. This series will be terminated with the release of standardized modules of the SCALE system.

The initial version of the automated SCALE system to be released in early F/ 1979 consists of a system driver, functional modules, a numeric nuclear data base, a materials information processor, and system control modules for out-of-core criticality safety studies. Advanced versions of SCALE will include radiation shielding and heat transfer analysis, more sophisticated control modules, and CRT interactive input capabilities. It is also expected that, in the long term, a compilation of the card image input and a retrieval code for all the critical experiment analyses used to validate the data libraries in the system will also be made available.

Under discussion are additional tasks associated with technical data acquired in support of NRC regulation and licensing processes, a continuous activity of the Commission. The lack of good documentation of the acquired data, the lack of data management procedures for analyzing, storing, and retrieving the data when needed, and the lack of an efficient mechanism for the dynamic maintenance of the data bases inhibit reliance on existing data. Consistent, accurate collection of the relevant data with the necessary methodology for analysis is clearly indicated. There is a perceived need for a center (a clearing-house) to systematize the organization and handling of the data, analyze it, and develop methodology for efficient retrieval in support of NRC regulation and licensing processes.

SACRD: Safety Analysis Computerized Reactor Data Base System

The Safety Analysis Computerized Reactor Data Base (SACRD), which was initiated in the Division in 1975, has now released Version 1 of the base to the Safety Analysis Data Coordinating Group (SADCG) and is releasing to the public through the publication of two forthcoming reports (ORNL-5477, Volumes 1 and 2). As reported in our last progress report, SACRD will provide input data for approximately 100 fast reactor safety analysis codes, and the Version 1 release includes data on thermodynamics, heat transfer, fluid mechanics, structural mechanics, aerosol transport, meteorology, neutronics and dosimetry. Other data will be added in future versions. In addition, SACRD will contain bibliographic and reference materials, files for commonly used correlations and variables, benchmarks for data testing, and files containing new but nonevaluated data.

The SACRD Coordinating Group is composed of representatives from industry, universities, regulatory bodies, national laboratories and DOE, and it is responsible for the evaluation and approval of the data released. The management system for the data base is the computer system JOSHUA, a modular code system developed at Savannah River Laboratory. Interactive terminal display capability under time-sharing operations is being implemented to facilitate easy user access to the data base.

CREDO: Central Reliability Data Organization

A program plan for CREDO, the recently established Central Reliability Data Organization that compiles statistics on the reliability of various nuclear plant components as determined from experience with operating reactors, has been distributed throughout the LMFBR community and is being revised on the basis of the feedback received. Currently CREDO is receiving historical data from EBR-II experience, from the Energy Technology Engineering Center at Atomic International (formerly the Liquid Metal Engineering Center) and from special sodium facilities at Argonne National Laboratory, General Electric, Westinghouse (WARD), and Hanford Engineering Development Laboratory. Discussions have been held with the British Systems Reliability Service (SRS) on the exchange of LMFBR reliability data, and the sodium component data on hand at SRS have been provided for CREDO. In addition, a member of the CREDO staff has presented an international paper on sodium valve failure rates (see Paper 7.8).

7.1 RECENT TRENDS IN RADIATION SHIELDING: AN RSIC PERSPECTIVE*

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(Abstract of paper to be presented at Third National Symposium on Radiation Physics, Waltair, A. P., India, February 22-24, 1978)

In attempting to compile and analyze recent trends in radiation shielding, one realizes that any description is highly dependent on one's perspective. This paper will be based on observations from within the Radiation Shielding Information Center (RSIC), where information has been gathered from the international shielding community for 16 years. The following trends, which are not listed in order of importance, are partly technical and partly socioorganizational. They are in both the research and development sector, principally for fission and fusion reactor radiation transport problems, and in the commercial nuclear power sector.

- (1) There is: diminishing use of empirical and approximate computational methods in favor of discrete ordinates (one and two dimensions) and Monte Carlo (up to three dimensions). Increasing sophistication in the use of these methods, for example, successive problems with different geometry description, coupling discrete-ordinates results to a Monte Carlo treatment of more detailed geometry. Increasing use of systematic variance reduction in Monte Carlo. Development of "channel" theory to follow only "contributors," i.e., particles which contribute to the sought effect. Greater availability of reflection data for use in albedo methods which can be applied to streaming problems.
- (2) We observe: great advances in cross-section availability and application, including use of common format, greater emphasis on elaborate and sophisticated evaluations guided by sensitivity analysis, greater attention to photon production and high neutron energies, interagency cooperation, data centers for cross-section processing and distribution, common reference libraries, integral experiments for data testing, coupling of neutron and gamma-ray cross-section sets for routine application, and generation and exchange of energy-group cross-section libraries. Group cross-section specifications guided by sensitivity analysis. Increased emphasis on fine-group libraries which are nearly application-independent and which facilitate user-produced problem-dependent libraries. Increased use of temperature-dependent self-shielding factors.
- (3) There is an increasing technical sophistication in the nuclear power industry. The architect-engineering firms, who are doing most of the power plant shielding design, have increased their capability, partly by absorbing experienced engineers from the declining government programs. Environmental and regulatory pressures have increased the importance of balance-of-plant shielding and radiation protection. Shielding people have become increasingly involved in all aspects of radiation protection.
- (4) There is increased emphasis in the nuclear industry on standards. There is increased interest in benchmark data, both experimental and purely computational, integral data testing and methods, and computer code and method validation.

These trends are illustrated by reference to computer codes and data available from RSIC.

*Research sponsored jointly by the U.S. Department of Energy, the U.S. Nuclear Regulatory Commission, and the Defence Nuclear Agency.

7.2 THE RADIATION SHIELDING INFORMATION CENTER: A SOURCE OF COMPUTER CODES AND DATA FOR FUSION REACTOR MATERIALS STUDIES*

R. W. Roussin D. K. Trubey B. F. Maskewitz

(Abstract of paper to be presented at Fifth Topical Meeting on Fusion Reactor Materials, Miami Beach, Florida, January 14-18, 1973)

Fusion reactor materials studies require computing technology and nuclear data for their analysis. The Radiation Shielding Information Center (RSIC) is sponsored by the Department of Energy, Office of Fusion Energy to provide a range of services which help to meet the requirements for materials studies.

The RSIC staff reviews the literature, indexes relevant reports, and maintains a file which can be searched upon request. Subject categories include radiation sources, measurement techniques, radiation damage, transport methods and results, and materials. A large (~320) collection of computer code packages (CCC) for radiation transport calculations has been acquired, tested, and provided upon request to the user community. A second class (~120) of computer code packages (PSR) is similarly maintained at RSIC. These include computer codes for multigroup cross-section generation, spectrum unfolding, kerma and damage factor calculations, etc. In addition to the two computer code collections, a data library collection (~50) is also available. These include multigroup neutron and gamma-ray cross sections, kerma factors, damage functions, and decay data.

In addition to normal services, OFE has sponsored special projects at RSIC on behalf of the general community. A collaboration with DOE-ORRT funded work resulted in the generation and distribution of a general-purpose multigroup cross-section library. The coupled library contains 171 neutron and 36 gamma-ray groups and includes cross sections for most materials in ENDF/B-IV. Kerma factors, recoil spectra, and damage cross sections are also included in this library, which is called DLC-41/VITAMIN-C. Specifications for VITAMIN-C were developed by polling fusion neutronics analysts for their needs.

RSIC serves as a focal point for the exchange of relevant technology within the community concerned with fusion neutronics and materials studies. It continues to provide an efficient mechanism for assuring that resources developed in the community are available so that the state of the art in fusion neutronics and materials studies can advance in a timely fashion.

*Research jointly sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology, and DOE Office of Fusion Energy, Division of Development and Technology.

7.3 BIBLIOGRAPHY, SUBJECT INDEX, AND AUTHOR INDEX OF THE LITERATURE EXAMINED BY THE RADIATION SHIELDING INFORMATION CENTER*

(Reactor and Weapons Radiation Shielding)

D. K. Trubey R. W. Roussin J. Gurney A. B. Gustin

(Abstract of ORNL/RSDI-1, Vol. V, January, 1974)

An indexed bibliography is presented of literature selected by the Radiation Shielding Information Center since the previous volume was published in 1974 in the area of radiation transport and shielding against radiation from nuclear reactors, X-ray machines, radioisotopes, nuclear weapons (including fallout), and low energy accelerators (e.g., neutron generators). The bibliography was typeset from data processed by computer from magnetic tape files. In addition to lists of literature titles by subject categories (accessions 3501-4950), author and keyword indexes are given.

Most of the literature selected for Vol. V was published in the years 1973-1976.

*Research jointly sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology; Defence Nuclear Agency; and Nuclear Regulatory Commission, Division of Technical Information and Document Control.

7.4 A REVIEW OF INFORMATION PROCESSING IN MEDICAL IMAGING: PROCEEDINGS OF THE FIFTH INTERNATIONAL CONFERENCE HELD AT VANDERBILT UNIVERSITY, NASHVILLE, TENNESSEE, JUNE 27 - JULY 1, 1977*

A. Bertrand Brill⁻ Ronald R. Price⁺
William J. McClain[‡] Mildred W. Landay

(Abstract of ORNL/DOE-140, December 1977)

The Fifth International Conference on Information Processing in Medical Imaging was held at Vanderbilt University in Nashville, Tennessee on June 27-July 1, 1977. Like the previous meetings in the series, its objective was to bring together the leaders of research in computer techniques in medical image analysis to share the advances they had made, to constructively criticize each other's work, and to develop directions for further research. An additional objective was to present to the community at large a view of the state of the art in this field by publication of these proceedings.

The participants included medical physicists, mathematicians, computer scientists, engineers, and physicians. The conference emphasized mathematical and computer-related matters over strictly clinical matters, though the clinical relevance of the techniques discussed was a prime concern. In addition to topics relating to scintigraphy, which were featured in previous meetings, this conference also focused much attention on the processing of radiological and ultrasonic images, the reconstruction, enhancement and characterization of computerized axial tomography (CAT), made with x-rays (transmission CAT) and gamma rays (emissions CAT), and display systems for 2- and 3-D information.

* Research jointly supported by DOE Office of Health and Environmental Safety, Division of Human Health Studies, and the HEW/FDA Bureau of Radiological Health.

⁻ Vanderbilt University.

[‡] Computer Sciences Division.

7.5 COMPUTING TECHNOLOGY RESOURCE SHARING*

William J. McClain⁻ Betty F. Maskewitz Jane Gurney

(Abstract of a Report by William J. McClain, Betty F. Maskewitz, and Jane Gurney, ORNL/DOE-140, December 1977)

The expense of developing and maintaining special software and interfaces has been prohibitive to many clinical installations desiring to perform computer-assisted studies. Also, existing developmental groups have been economically unable to adequately transfer their work to such small installations. In response to this need and through the efforts of the Society of Nuclear Medicine Computer Committee, the U.S. Energy Research and Development Administration (ERDA), recently incorporated in the U.S. Department of Energy (DOE), established the Biomedical Computing Technology Information Center (BCTIC) at the Oak Ridge National Laboratory (ORNL). BCTIC collects, organizes, evaluates, and disseminates information on computing technology pertinent to biomedicine in general and nuclear medicine in particular, providing the needed routes of communication between installations, and serving as a clearinghouse for the exchange of biomedical computing software, data, and interface designs.

* The basic sponsor for this work is the U.S. Department of Energy (DOE). Other sponsors include the Bureau of Radiological Health, Food and Drug Administration; NIH; Veterans' Administration; the Society of Nuclear Medicine; and the Society for Computer Medicine. Computer Sciences Division.

7.6 NRC REACTOR SAFETY RESEARCH DATA REPOSITORY ESTABLISHED AT ORNL*

Betty F. Maskewitz

(Abstract of a paper presented at the Sixth Water Reactor Safety Research Information Meeting, National Bureau of Standards, Washington, D.C., November 4-8, 1975)

A repository and distribution center for reactor safety research data, sponsored by the Division of Reactor Safety Research of the U.S. Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research, has been established at the Oak Ridge National Laboratory. The Reactor Safety Research Data Repository (RSRDR) serves as the archival repository and response center of the NRC/RSR Data Bank System, which also includes Data Base Managers and a Data Bank Processing System located at the Idaho National Engineering Laboratory (INEL). Several Data Base Managers collect and evaluate data to be included in the Data Bank System. The INEL Data Bank Processing System processes the data into a standard format, sends it to RSRDR and makes it directly available to NRC and INEL researchers via on-line interaction with the INEL computer.

The RC/RSR Data Bank System is designed to manage data pertinent to research investigating the loss-of-coolant accident for both pressurized water reactors and boiling water reactors. The data is in the form of raw measured data that has not been processed with the exception of conversion to engineering units and application of calibration factors. Data sources include experimental programs sponsored by NRC, other government agencies, private industry, and foreign governments. Specific examples include data from:

1. Semiscale
2. Loss of Fluid Test (LOFT)
3. Two Loop Test Apparatus (TLTA)
4. Thermal hydraulic Test Facility (THTF)
5. Full Length Emergency Cooling Heat Transfer Facility (FLECHT)
6. Other Heat Transfer Loops
7. Two Phase Pump Performance Facilities

The RSRD Repository will maintain archival records and provide a distribution service. The stored data is available on demand to NRC contractors, to contributors to the Data Bank, and to other individuals or organizations working in the field of reactor safety. A nominal cost recovery fee is charged. However, charges may be waived on determination that the data is to be used in NRC-sponsored programs or that supplying it is in the best interests of NRC's mission.

Procedures have been developed by RSRDR to make its information quickly available to those writing, telephoning, or visiting the Laboratory. A directory of persons interested in the data and its use is being compiled and a mechanism for sharing feedback from use of the data will be established.

A descriptive brochure, written guidelines for the use of those requesting data, and an inventory of the data currently packaged for dissemination are available in the NRC/RSR Data Bank System Exhibit Area in the hall opposite NBS Conference Room A.

*Research sponsored by Nuclear Regulatory Commission, Division of Reactor Safety Research.

7.7 CREDO: A RELIABILITY DATA ANALYSIS CENTER FOR ADVANCED REACTORS*

G. F. Flanagan N. M. Greene P. M. Haas T. F. Bott

(Abstract of paper presented at the Sixth Water Reactor Safety Research Information Meeting, National Bureau of Standards, Washington, D.C., November 4-8, 1975)

In the Fall of 1975, ERDA brought together several experts in reactor reliability and risk assessment for the purpose of establishing a National Advanced Reactor Reliability Data System (NARRDS). Subsequent to the meeting, Oak Ridge National Laboratory was

requested to formulate a program that would accommodate the recommendations of NARRDS, as well as more recent reliability efforts in both the light-water reactor and advanced reactor fields. A comprehensive review by ORNL resulted in the creation of a Centralized Reliability Data Organization (CREDO) at ORNL.

What follows is a description of CREDO's objectives, functional and operational procedures (Fig. 7.7.1), and expected schedule for implementation.

CREDO will have as its primary purpose the collection, screening, cataloging, storing, retrieving, processing, and dissemination of reliability and maintainability/availability data for advanced nuclear reactors. It will be user oriented and computer based. Both qualitative and quantitative data will be collected and disseminated. The quantitative data will be raw or reduced, depending on the user requirements. In all cases, proprietary information will be protected. Although CREDO will focus on the nuclear island, balance-of-plant data will also be collected, primarily from existing data bases such as the Nuclear Power Reactor Data System (NPROS).

In addition to its data-related functions, CREDO will provide specialized evaluation and analysis services on both a periodic and a user-requested basis. It will also serve as the United States focal point for international data exchange and validation efforts.

Data will be collected from existing and to-be-built reactors (e.g., EBR-II, FFTF), as well as from engineering or test facilities such as those existing at the Liquid Metals Engineering Center (LMC) or at Idaho National Engineering Laboratory (INEL), Argonne National Laboratory (ANL), ORNL, and other national research centers or universities. It is also anticipated that data already in existence, such as those collected by the Clinch River Breeder Reactor Project (CRBRP), the Gas-Cooled Reactor Project, the LMC failure data handbook, IEEE-500 Project, WASH-1400, and existing data centers such as NPROS, would be utilized in CREDO.

There are three key considerations in the operational structure of CREDO as shown in Fig. 7.7.1:

1. The foundation on which the data base is organized is the user. Data collection efforts, the data management system, and all user services will be designed to provide simple, efficient, and flexible user interaction.
2. The heart of CREDO is the Data Management System. It is the focal point for all interactive communication.
3. Viable feedback mechanisms directly from CREDO and users that provide useful services to data source organizations are recognized as essential in CREDO.

CREDO will provide several user services such as "CAUTION" and "PLEA" systems, directed searches, special reports and periodic summary reports, code consultation, and dissemination. These are summarized in Fig. 7.7.2.

The data base management system will consist of a combination of two existing systems: the JOSHUA system developed at Savannah River Laboratory and the ORCHIS data management scheme developed at ORNL. Both have attractive features necessary for the management of a large, diverse system as proposed for CREDO. JOSHUA will act as the system manager with ORCHIS providing the data management capability.

The schedule for implementation calls for the issuance of initial qualitative information such as trends analysis and "mini-fault trees" by September 1978. Actual public inquiry of the data base for quantitative data will be made available in late 1979.

* Research sponsored by DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

Computer Sciences Division.

Fig. 7.7.1. Operational procedures for CREDO.

Fig. 7.7.2. CREDO input and output.

7.8 INITIAL DATA COLLECTION EFFORTS OF CREDO: SODIUM VALVE FAILURES*

T. F. Bott P. M. Haas

(Abstract of paper presented at Fifth Symposium on Reliability Technology, University of Bradford, United Kingdom, September 11-14, 1978; Proceedings published in NCSR-880, 83-116, November 1978)

The Centralized Reliability Data Organization (CREDO) has been established at Oak Ridge National Laboratory (ORNL) to define, develop, and maintain a reliability data analysis center for use in advanced reactor safety and licensing. Its primary functions are collection, reduction, evaluation, storage, retrieval and dissemination of reliability/maintainability data.

Data-collection efforts have been initiated at several test loops, at the Experimental Breeder Reactor-II (EBR-II), and at the Fast Flux Test Facility (FFTF). Top priority is being given to collecting data on safety and safety-related systems, primarily for sodium-cooled reactors. Sufficient operating time (more than 25 million hours) has been accumulated on sodium valves at test facilities to provide quantitative estimates of reliability characteristics with a reasonable degree of confidence.

Sodium-valve failures have been categorized according to seat design, size, seal type (freeze-seal or bellows) and actuator type (manual or pneumatic/motor). Attempts have been made to establish the variation of failure rate with time and duty (operating frequency). Estimates of failure rates for sodium valves have been compared to those for water valves and appear to be on the same order of magnitude.

As part of the routine operation of CREDO, it is planned to have all data and data analysis reviewed by an "evaluation committee" composed of experts in the various disciplines associated with reliability engineering. This review process has not yet been established, and the results of this work should therefore be recognized as preliminary.

* Research jointly sponsored by DOE Office of Nuclear Energy Programs, Division of Nuclear Power Development, and DOE Office of Nuclear Energy Programs, Division of Reactor Research and Technology.

APPENDIX

SCIENTIFIC AND PROFESSIONAL ACTIVITIES

May 1, 1977 - November 1, 1978

- R. G. Alsmiller, Jr.** Member, DOE Advisory Panel on Accelerator Radiation Safety
Chairman, DOE Advisory Panel on Accelerator Radiation Safety
for Review of 400-GeV Proton-Proton Colliding Beam
Facility (ISABELLE) at BNL
Chairman, Admissions Committee, American Nuclear Society
Seminar Speaker, Argonne National Laboratory, July 26, 1977.
"Medium-Energy Nucleon-Meson Transport Calculations and
Electronuclear Fuel Production"
Invited Paper, Second ASIM-EURATOM Symposium on Reactor
Dosimetry: Dosimetry Methods for Fuels, Cladding, and
Structural Materials, Oct. 3-7, 1977, Palo Alto, CA.
"Neutronics and Radiation Damage Calculations for Fusion
Reactors"
Invited Paper, Annual Meeting of the American Nuclear Society,
June 18-23, 1978, San Diego, CA, "Medium-Energy Nucleon-Meson
Transport Calculations and Electronuclear Fuel Production"
- D. E. Bartine** Chairman, Program Committee, Radiation Protection and Shielding
Division, American Nuclear Society
Member, National Program Committee, American Nuclear Society
Member, ANS-1, Cross Sections for Shielding (Standards
Committee)
Member, Program Committee, Joint Reactor Physics and Shielding
Topical Meeting, Sun Valley, Idaho, 1980
Invited Paper, ASME-IEEE Joint Power Generation Conference,
Long Beach, CA, "Use of Thorium in Power Reactors," ASME
Publication No. 77-JPGC-W-24 (Sept. 1977)
Invited Paper, SALES Conference, Knoxville, TN (Feb. 1978),
"Alternate Fuel Cycles for Breeder Reactors"
Invited Paper, American Nuclear Society Topical Conference,
"Advances in Reactor Physics," Gatlinburg, TN (April 1978)
"Alternate Fuel Cycles for Breeder Reactors"
Informal Seminars, Georgia Institute of Technology, Oct. 1977
and May 1978
Seminar presented at EIR, Würenlingen, Switzerland, Nov. 1977.
"Nonproliferation Aspects of Alternate Fuel Cycles."
Outstanding Service Award, presented by the American Nuclear
Society, Radiation Protection and Shielding Division, at
the San Diego Meeting of the ANS, June 1978
- T. J. Burns** Invited Paper, "Alternate Fuel Cycles for Breeder Reactors,"
with D. E. Bartine, J. R. White, J. M. Fallfetz, and
D. Rowland, ANS Topical Conference, "Advances in Reactor
Physics," Gatlinburg, TN, April 1978.
Nominated for Best Paper Award, American Nuclear Society
Winter Meeting, Nuclear Fuel Cycle Division, Nov. 1977
Oral Presentation, "Performance of Alternate GCFR Fuel
Cycles," GCFR Information Meeting, May 1978
- D. G. Cacuci** Member, Cross Section Evaluation Working Group (CSEWG)
- G. de Saussure** Member, National Program Committee, American Nuclear Society
Member, Steering Subcommittee of National Program Committee,
American Nuclear Society
Member, International Meetings Subcommittee of National Program
Committee, American Nuclear Society
Invited for three months consulting at Bruyères-le-Châtel
Laboratory, May-July, 1978, May-July 1978

- Invited for one month consulting, Los Alamos Scientific Laboratory, July, 1977
 Invited Seminar, "U-238 Cross Section Measurements at ORELA," Geel, Belgium, June 1978
 Invited Consultant, Harwell Physics Division, UK, June 21-23, 1978
- J. K. Dickens Member, American Nuclear Society Standard Working Group 5.1: Decay Heat Standard
 Referee, *Nuclear Science and Engineering*
 Designated by IAEA Working Group on Fission Product Nuclear Data as the correspondent on worldwide experimental measurements of fission product decay heat (September 1977)
 Staff Member, ORNL Senior Planning Group (1978)
 Technical Staff, ORNL Master Plan (1978)
- W. W. Engle, Jr. Invited Lecturer at the "Ettore Majorana Centre for Scientific Culture, The Use of Computers in Health Physics: Computational Techniques in Shielding and Dosimetry Oct. 1978, Lecture 1: "The Method of Discrete Ordinates in Low-Energy Neutron-Photon Transport (MORSE);"
 Lecture 2: "Discrete Ordinates Applications in Low-Energy Neutron-Photon Transport (MORSE);"
 Lecture 3: "The AMP Cross-Section Processing System"
 Seminars sponsored by the State Power Board, Sweden, presented at Stockholm and ASEA-ATOM, Västerås, Sweden, October 1978, "Current Techniques in Discrete Ordinates Calculations"
- G. F. Flanagan Assistant General Chairman, "Conference on Advances in Reactor Physics," Gatlinburg, Tennessee, April 1978
 Vice Chairman, Reactor Safety Division Program Committee, American Nuclear Society
 Member, Technical Program Committee, Fast Reactor Safety Conference, Seattle, Washington
 Member, Standards Committee, American Nuclear Society
 Chairman, Program and Nominating Committees for Local ANS Section
 Guest Lecturer, University of Florida, "Fast Reactor Safety"
 Assistant Professor, University of Tennessee, "Fast Reactor Safety"
 Member, Ph.D. Recruiting Program, ORNL
 Participant, ORNL Speakers' Bureau, "Nuclear Power and the Environment, Biology Society at Davidson College, N.C., October 1977
 Member, WATec Sponsors Committee
 Chairman, Action Steering Committee, American Nuclear Society
 Member, Editorial Staff, ANS Newsletter, Local ANS Section
 Registration Chairman, "Fuel Cycles of the 80's, Topical Meeting, Gatlinburg, Tennessee, 1980"
- C. Y. Fu Member, Cross Section Evaluation Working Group Nuclear Model Codes Subcommittee
 Invited Paper, "Multi-step Hauser-Feshbach Codes with Precompound Effects: A Brief Review of Current and Required Developments and Applications," Symposium on Nuclear Cross Sections from 10 to 40 MeV, OHL-NCS-80611 (1977).
 Organizer, Symposium on "Development and Applications of Nuclear Model Codes," at Oak Ridge for CSWG Nuclear Model Codes Subcommittee, October, 1978
- T. A. Gabriel Invited Lecturer at the "Ettore Majorana Centre for Scientific Culture, The Use of Computers in Health Physics: Computational Techniques in Shielding and Dosimetry, Oct. 1978), Lecture 1: "The Method of Monte Carlo in Low Energy Neutron-Photon Transport (MORSE);"
 Lecture 2: "Monte Carlo Applications in Low-energy Neutron-Photon Transport (MORSE);"
 Lecture 3: "The Radiation Shielding Information Center (RSIC)"
 - Advisor to SLAC for PEP Shielding
 Advisor to SLAC for Calorimeter Design

- R. Gwin Member ORNL Criticality Committee
- P. M. Haas Member, Critical Reviews Committee, American Nuclear Society
Member, IEEE NSRS Subcommittee, Working Group on Qualitative Data Base Requirements
Member, Publicity Committee and Editorial Staff, ANS Newsletter, ANS Local Section
Section Editor, *Nuclear Safety Journal*
- D. T. Ingersoll Member, Program Committee, Radiation Protection and Shielding Division, American Nuclear Society
Member, Shielding Subcommittee, Cross Section Evaluation Working Group (CSEWG)
- H. E. Knee Member, Editorial Staff, ANS Newsletter, Local ANS Section
- D. C. Larson Member, Shielding Subcommittee, Cross Section Evaluation Working Group (CSEWG)
Member, Codes and Formats Subcommittee, Cross Section Evaluation Working Group (CSEWG)
Invited Paper, "Multistep Hauser-Feshbach Calculations for ^{23}Na ," CSEWG Nuclear Model Codes Subcommittee, ORNL, Oct. 23, 1978
- R. A. Lillie Invited Lecture, "The Finite-Element Method in Nuclear Reactor Calculations," University of Tennessee, January 1978
Invited Lecture, "Discrete Ordinates Method and Applications," Tennessee Industries Week Short Course, U. of Tennessee, Sept. 1978
Chairman, Post Session Committee, International Conference on Reactor Shielding, Knoxville, April 1977
Cochairman, Engineering Physics Division Seminars
- R. E. Maerker Chairman, Shielding Data Testing Sub-Subcommittee, Cross Section Evaluation Working Group (CSEWG)
Member, Shielding Subcommittee, CSEWG
Invited Paper, "Calculations of Several CSEWG Shielding Benchmarks with Preliminary ENDF/B-V Data," Transactions, American Nuclear Society, 28, 737 (1978)
Invited Paper, "General Status of Accuracy of ENDF/B-IV Cross Sections for LMFBR Shielding Calculations," Transactions, American Nuclear Society, 26, 485 (1977)
- F. C. Maienschein Member, DOE Advisory Committee on Reactor Physics (ACRP)
Member, Nuclear Energy Agency Committee on Reactor Physics (NEACRP)
Member, University of Illinois, Department of Nuclear Engineering Advisory Committee
General Chairman, American Nuclear Society Conference on "Advances in Reactor Physics," Gatlinburg, Tennessee, April 1978
Organized Special Session, New York Meeting of the American Nuclear Society on "Definitions of Breeding Ratio (BR) and Doubling Time (DT)," June 1977
Chairman, ORNL Delegation Study Group
Member, ORNL Awards Advisory Committee
- J. H. Marable Member, Data Testing Subcommittee, Cross Section Evaluation Working Group (CSEWG)
Member, ORNL Criticality Committee
Invited Paper, "Sensitivity Analysis of Fast Reactor Systems Including Thorium and Uranium," American Nuclear Society 1978 Summer Meeting
Invited Lecturer, RSIC Seminar-Workshop, "Theory and Applications of Sensitivity and Uncertainty Analysis," Aug. 1978, Oak Ridge

- B. F. Maskewitz** Chairman, Nominations, Radiation Protection and Shielding Division, American Nuclear Society
 Member, N.E.E.D. Committee, American Nuclear Society
 Member, ORNL Honors and Awards Advisory Committee
 Member, ANS-10 Subcommittee of the American Nuclear Society Standards Committee
 Member, Society of Nuclear Medicine Council on Computers
 Member, Organizing Committee for the Annual Scientific Symposium of the Society of Nuclear Medicine Council on Computers
 Referee, Computer Code Abstracts, *Nuclear Science and Engineering*
 Referee, Information Science Papers, 1979 National Computer Conference
 Archivist, MUMPS Development Committee
 Member, Standards Committee, Society for Computer Medicine
 Member, Advisory Committee, National Science Foundation sponsored Ultrasound Tissue Signature Project (Carnegie-Mellon Institute of Research, Pittsburgh)
 Member, Committee for Review of DOE-NEA Information Exchange Agreement
 Member, ORNL IAC Forum
 Invited Lectures, Several lectures on RSIC and ECTIC during 1978 visits to Italy, Hungary, Poland, France, and England
 EPD Technology Utilization Officer Invited Televised Lecture, "Biomedical Computing, sponsored by DHEW/FDA Bureau of Radiological Health, April 1978
- B. L. McGill** Chairman, Guest Program, 5th International Conference on Reactor Shielding, Knoxville, April 1977
 Member, Nominating Committee, IAC Forum, ORNL, 1978
 Chairman, Program Committee, IAC Forum, ORNL, 1977
 Treasurer, Special Interest Group - Information Analysis Centers, American Society for Information Science, 1978-79
 Associate Editor, Newsletter of Special Interest Group - Information Analysis Centers, American Society for Information Science
- D. K. Olsen** Member, Program Committee, Reactor Physics Division, American Nuclear Society
 Cochairman, Engineering Physics Division Seminars
- R. W. Peelle** Member, Cross Section Evaluation Work Group (CSEWG) Normalization and Standards Committee
 Member, DOE Nuclear Data Committee
 Chairman, ORNL Proposal Review Committee (1977 - March 1978)
 Referee, *Applied Physics*
 ANS Fellow, named at Annual Meeting of the American Nuclear Society, June 1978
- F. G. Perey** Member, Nuclear Energy Agency Nuclear Data Committee (NEANDC)
 Chairman, Neutron Standards Subcommittee, NEANDS
 Member, DOE, Nuclear Data Committee
 Member, INDC Standards Subcommittee
 Associate Editor, *Nuclear Data and Reaction Cross Sections*
 Invited Lecture, "Errors and Correlations in Nuclear Data: Their Evaluation and Applications," C.N.E.N., Cassacia, Italy, Oct. 1978
 Invited Paper, "Recent ORELA Results on Fe," NEANDC/NEACRP International Specialist Meeting on Neutron Data of Structural Materials for Fast Reactors, Geel, Belgium, December 1977
 Invited Paper, "Uncertainty Analysis of Dosimetry Spectrum Unfolding," Second ASTM-EURATOM Symposium on Reactor Dosimetry: Dosimetry Methods for Fuels, Cladding, and Structural Materials, Palo Alto, California, October 1977
 Invited Paper, "Spectrum Unfolding by the Least-Squares Method," IAEA Technical Committee Meeting on Current Status of Neutron Spectrum Unfolding, Oak Ridge, Tennessee, October 1977

- Invited Paper, "Expectations for ENDF/B-V Covariance Files: Coverage, Strength and Limitations," Workshop on Theory and Application of Sensitivity and Uncertainties Analysis, Oak Ridge, Tennessee, August 1978
- Invited Paper, "Covariance Matrices of Experimental Data," International Conference on Neutron Physics Data for Reactors and Other Applied Purposes, Harwell, England, September 1978
- R. B. Perez** ANS Fellow, named at Annual Meeting of the American Nuclear Society, June 1978
- R. W. Roussin** Chairman, Shielding Subcommittee, Cross Section Evaluation Working Group (CSEWG)
 Chairman, Proceedings, 5th International Conference on Reactor Shielding
 Member, Codes and Formats Committee, (CSEWG)
 Member, ANS 6.1.2 Cross Section Standards Committee
 Member, Executive Committee, Radiation Protection and Shielding Division, American Nuclear Society
 Invited Paper, "Multigroup Data Commonly Available for Fission and Fusion Reactor Shielding, ANS Meeting, New York, 1977
 Invited Paper, "Summary of the 5th International Conference on Reactor Shielding, Atomkernenergie, 30, No. 4, p. 227 (1977)
 Lecturer, University of Tennessee, Tennessee Industrial Week Course, "An Engineering Approach to Monte Carlo Analysis," 1977 and 1978
 Coordinator, Seminar-Workshop on Multigroup Cross Section Preparation and Handling, Oak Ridge, Tennessee, March 1978
 Coordinator, Seminar-Workshop on Theory and Application of Sensitivity and Uncertainty Analysis, Oak Ridge, Tennessee, August 1978
- R. T. Santoro** Member, Publications Committee, American Nuclear Society
 Invited Paper, Second ASTM-EURATOM Symposium on Reactor Dosimetry: Dosimetry Methods for Fuels, Cladding, and Structural Materials, October 3-7, 1977, Palo Alto, CA, "Neutronics and Radiation Damage Calculations for Fusion Reactors"
 Task Leader, Assessment of Advanced Technology Options, A NASAP Study Session Chairman, Fusion Reactor Neutronics and Shielding, Winter Meeting, American Nuclear Society, Washington, D. C., November 1978
- D. L. Selby** Member, Membership Committee, Local Section, American Nuclear Society
 Member, Editorial Staff, ANS Newsletter, Local Section, ANS
 Awards Coordinator, Engineering Physics Division
 Participant, ORNL Speakers' Bureau, "How Does a Nuclear Power Plant Work and Why is There so Much Opposition to Nuclear Power?" Sullivan Business College and Louisville Optimist Club, Louisville, Kentucky, July 31 and August 1, 1978
- E. G. Silver** Member, National Program Committee, American Nuclear Society
 Chairman, National Forum Subcommittee of ANS Program Committee
 Member, Education Committee of East Tennessee Section of ANS
 Member, Program Committee, Physics Division of ANS
 Chairman, Publications, and Member of Program Committee, ANS Topical Meeting on "Advances in Reactor Physics," Gatlinburg, Tennessee, April 1978
 Lecturer, "Physics of Energy," ORAU-EEO
- R. R. Spencer** Invited Paper, "Status Report on the ORNL Absolute Measurement of σ_p for ^{235}Cf at CSWEG Normalization and Standards Subcommittee Workshop, December 1977
 Reviewer, Data on ^{235}Cf for CSWEG Standards and Normalization Subcommittee
 Reviewer, *Nuclear Data and Engineering*
 Reviewer, *The Physical Review*

D. K. Trubey

Chairman, Honors and Awards Committee, Radiation Protection and Shielding Division, American Nuclear Society
 Chairman, ANS-6 Radiation Protection and Shielding Subcommittee, Standards Committee, American Nuclear Society
 Member, ANS-6.5 Shielding Glossary, American Nuclear Society
 Member, Working Group on Code Interfaces and Formats, Safety Analysis Data Coordinating Group
 Coordinator, Metrication, Engineering Physics Division

D. R. Vondy

Member, NSC Benchmark Committee, American Nuclear Society
 Member, ANS-10 Standards Committee, American Nuclear Society
 Alternate Member, ANSI Standards Committee, X-3, American Nuclear Society
 Alternate Member, X-3 Fortran Committee, X3J3, American Nuclear Society
 Member, DOE, Division of Reactor Development and Technology, Physics Branch, Large Core Methods Evaluation Committee
 Member, DOE, Division of Reactor Development and Technology, Physics Branch, Cooperative Computer Coding Committee
 Lecturer, Tennessee Industries Week, "Computational Methods in Nuclear Reactor Analysis," (4 sessions), University of Tennessee

J. J. Wagschal

Comptroller, Israel Nuclear Society
 Seminar, INR-GFK, Karlsruhe, Germany, August 18, 1978, "Physical and Nonphysical Neutron Energy Spectra."
 Invited Paper, "Sensitivity of Reactivities to Current Cross-Section Libraries: A Case for Adjustment," with U. Salmi, Atara Ya'ari and Y. Yeivin, American Nuclear Society, 1977 Winter Meeting
 Associate Professor, The Hebrew University of Jerusalem, Israel
 Visiting Associate Professor, Ben Gurion University of the Negev, Israel (October 1977 - September 1978)

C. R. Weisbin

Chairman, Program Committee, Reactor Physics Division, American Nuclear Society
 Chairman, Data Testing Subcommittee, Cross Section Evaluation Working Group (CSENG)
 Member, Physics of Reactor Design Standards Committee, ANS-19
 Member, Codes and Formats, Fission Products and Shielding, CSENG
 Invited Paper, "Procedures for Evaluating Uncertainties in Estimated Reactor Performance Parameters," for issue in *Advances in Nuclear Science and Technology*; "Specifications for Adjusted Cross Section and Covariance Libraries Based Upon CSENG Fast Reactor Benchmark Experiments and Analysis," with J. H. Marable and M. Salvatores

M. L. Williams

Invited Paper, "Sensitivity Theory for Depletion Analysis," with J. R. White, J. H. Marable, and E. M. Oblow, RSIC Conference on Developments in Sensitivity Theory, Oak Ridge, Tennessee, 1977

B. A. Worley

Member, Program Committee, Reactor Physics Division, American Nuclear Society

ENGINEERING PHYSICS DIVISION SEMINARS AT ORNL

D. K. Olsen and W. A. Rhoades were Seminar Cochairmen during the period covered by this report. R. A. Lillie has recently succeeded W. A. Rhoades in this activity. The following seminars were held during 1977 and through October 31, 1978.

- J. Boldeman, AEC, Lucas Heights, Australia, "Current Research at Lucas Heights and the Status of ^{252}Cf Nu-bar Standard"
- R. S. Booth, Instrumentation and Controls Division, "Surveillance and Diagnostics of Nuclear Power Plants by Noise Analysis"
- W. R. Burrus, Tennecomp Systems, "Confidence Limits on Fast Neutron Damage"
- G. T. Chapman, Engineering Physics Division, "Use of the ORELA PDP-10"
- J. G. Craven, Computer Sciences Division, "Use of the ORELA PDP-10"
- R. Cusson, Duke University, "Survey of (n, γ) and (n, f) Actinide Cross-Section Calculations"
- J. B. Czirr, Lawrence Livermore Laboratory, "Neutron Detectors of the Future"
- J. W. T. Dabbs, Physics Division, "Detecting Fission in the Presence of High Alpha Background"
- J. K. Dickens, Engineering Physics Division, "Measurement of Decay Heat from 2 to 14000 sec after Thermal Fission of ^{235}U and ^{239}Pu "
- E. Greenspan, Consultant, Nuclear Research Centre - Negev, Israel, "Accelerator-Driven Natural Uranium Fueled Light-Water Moderated Breeding Power Reactors"
- Y. S. Gur, Soreq Nuclear Research Center, Israel, "Nuclear Data in the Unresolved Resonance Region"
- D. R. Harris, Rensselaer Polytechnic Institute, " ^{238}U Capture Self-Indication Measurement and Analysis"
- N. W. Hill, Instrumentation and Controls Division, "Plattard-Version, Blons-Designed Fission Gaseous Scintillator"
- G. D. James, Harwell, AERE, Oxfordshire, England, "Nuclear Energy Standards"
- H. B. Hurt, Health Division, "Alcohol Abuse"
- M. J. Kenney, Brookhaven National Laboratory and Australian AEC, "Recent Neutron Capture Measurements at Brookhaven National Laboratory"
- H. Klein, W. Germany, "The Neutron Facility at the Physikalisches Technische Bundesanstalt (PTB)"
- R. L. Macklin, Physics Division, "Deuterated Pulse-Height-Weighted Neutron Capture Tank"
- H. Mook, Solid State Division, "Can ORELA be Used for Solid State Physics?"
- G. L. Morgan, Engineering Physics Division, "Cross Sections for the $^{14}\text{N}(n, p_0)$, (n, α_0) , and (n, α_1) Reactions from 0.5 to 15 MeV"

F. G. Perey, Engineering Physics Division, "Data Adjustments - When, How, and Why?" and "A New Approach to the Few Channel Problem - Application of Bayes' Theorem to Spectrum Unfolding and Data Adjustment:

Cleide Renner, Instituto de Energia Atomica - IEA, Sao Paulo, Brazil, " ${}^6\text{Li}(n,\alpha)$ Cross-Section Measurement Around 240 keV"

E. M. Robinson, Industrial Safety Department, "Defensive Driving"

H. Ruley, Health Division, "Medical Hazards of High Voltage"

N. M. Steen, Bettis Atomic Power Laboratory, "Fission Neutron Spectra"

R. R. Spencer, Engineering Physics Division, "Absolute Measurement of ${}^{252}\text{Cf}$ Nu-bar"

J. H. Todd, Instrumentation and Controls Division, "Fast ADCs"

D. R. Vondy, Engineering Physics Division, "The Once Through, Continuous Fueling Exposure Model," and "Power Reactor Fuel Cost Calculations (with Consideration of the Economic Model, Interest Charges, Discounting)"

R. R. Winters, Physics Department, Denison University, "Cosmology, Nucleosynthesis and Neutron Capture at ORELA"

PUBLICATIONS*

ALSMILLER, F. S.,** R. G. ALSMILLER, JR., H. W. BERTINI,** AND C. L. BEGOVICH**

"Monte Carlo Simulation of Turbulent Atmospheric Transport and Comparisons with Experimental Data," ORNL/TM-6291 (March 1978). (6.45)

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"Neutron-Photon Multigroup Cross Sections for Neutron Energies ≤ 60 MeV," ORNL/TM-6486 (August 1978). (6.28)

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**Not a Division member.

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STEVENS, P. N.,** C. E. CLIFFORD, F. J. MUCKENTHALER, AND W. YOON,** "The Thermoluminescent Dosimeter Measurement of Gamma-Ray Energy Deposition Distributions in Reactor Shields" (4.4)

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BURNS, T. J., J. R. WHITE,** D. E. BARTINE, J. M. KALLFELZ,** AND D. M. ROWLAND,** "Denatured Fast Reactor Symbiosis," p. 454. (6.12)

CHILDS, R. L.,** J. H. MARABLE, AND C. R. WEISBIN, "Two-Dimensional Reactor Sensitivity Analysis Using Discrete-Ordinates Transport Theory," p. 909. (3.27)

FLANAGAN, G. F., N. M. GREENE,** P. M. HAAS, AND T. F. BOTT, "CREDO: A Reliability Data Analysis Center for Advanced Reactors," p. 638. (7.7)

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LILLIE, R. A., R. G. ALSMILLER, JR., AND J. T. MIHALCZO,** "Design Calculations for a Neutron Collimator for the TFTR," p. 797. (6.27)

MAERKER, R. E., F. J. MUCKENTHALER, AND C. E. CLIFFORD, "Radiation Heating Studies in a Stainless Steel and Sodium Shield," p. 772. (4.5)

OLSEN, D. K., G. de SAUSSURE, R. B. PEREZ, F. C. DIFILIPPO,** R. W. INGLE,** AND H. WEAVER, "Measurement and Resonance Analysis of 150-m Neutron Transmissions Through Uranium-238," p. 870. (1.11)

RHOADES, W. A., AND W. W. ENGLE, JR., "A New Weighted-Difference Formulation for Discrete Ordinates Calculations," p. 776. (5.2)

SELBY, D. L., G. F. FLANAGAN, AND D. L. LOYD,** "Analysis of the ZPPR-5 Source Level Flux Monitor (SLFM) Experiment," p. 905. (6.5)

TOMLINSON, E. T.,** "Transport/Diffusion Comparisons for Small-Core LMFBR Disruptive Accidents," p. 558. (5.21)

WEISBIN, C. R., E. M. OBIOW, J. H. MARABLE, AND M. SALVATORES,** "Data Adjustment: A Cautiously Optimistic View for the Improvement of Design Performance Calculations and Data Assessment," p. 681. (3.10)

WILLIAMS, M. L., AND W. W. ENGLE, JR., "Analysis of Streaming Through the Primary Coolant Pipe Chaseway of the Clinch River Breeder Reactor," p. 779. (6.2)

WORLEY, B. A., AND A. F. HENRY,** "The Computation of Homogenized Group Parameters," p. 392. (5.23)

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CACUCI, D. G., AND M. A. BJERKE,** "Analytical Inequalities Satisfied by the Cross-Section Self-Shielding Factors," p. 227. (3.30)

CULLEN, D. E.,** E. F. PLECHATY,** R. J. DOYAS,** C. R. WEISBIN, AND J. E. WHITE,** "Cross-Section Probability Tables in Multi-Group Transport Calculations," p. 155. (3.2)

MacFARLANE, R. E.,** C. R. WEISBIN, AND N. C. PAIK,** "The Shielding Factor Method for Producing Effective Cross Sections: MINX/SPHINX and the CCCC Interface System," p. 83. (3.4)

ROUSSIN, R. W., C. R. WEISBIN, J. E. WHITE,** R. Q. WRIGHT,** J. M. GREENE,** W. E. FORD, III,** J. B. WRIGHT, AND B. R. DIGGS,** "Experience in Developing and Using the VITAMIN-C 17 $\frac{1}{2}$ -Neutron, 36-Gamma-Ray Group Cross-Section Library," p. 107. (3.3)

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MARABLE, J. H., AND C. R. WEISBIN, "Uncertainties in the Breeding Ratio of a Large LMFBR," p. 231. (3.19)

WEISBIN, C. R., J. H. MARABLE, J. HARDY, JR.,** AND R. D. MCKNIGHT,** "Projections of ENDF/B Version V Performance for Fast and Thermal Reactors Using Sensitivity Coefficients," p. 79. (3.6)

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CRAMER, S. N., "Solid Angle Subtended at a Point by a Cylinder," p. 263. (5.10)

INGERSOLL, D. T., C. O. SLATER, AND M. L. WILLIAMS, "Design of a Prototypic GCFR Grid-Plate Shield Confirmation Experiment," p. 630. (4.12)

LOYD, D. L.,** J. C. ROBINSON,** AND D. L. SELBY,** "Evaluation of a Variational Method for Estimation of Neutron Detection Efficiency," p. 773. (6.4)

MAERKER, R. E., "Calculations of Several CSEWG Shielding Benchmarks with Preliminary ENDF/B-V Data," p. 737. (3.5)

MARABLE, J. H., AND C. R. WEISBIN, "Sensitivity Analyses of Fast Reactor Systems Including Thorium and Uranium," p. 718. (3.24)

MUCKENTHALEP, F. J., J. L. HULL, AND J. J. MANNING, "The GCFR Grid-Plate Shield Design Experiment," p. 631. (4.13)

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SANTORO, R. T., R. A. LILLIE, R. G. ALSMILLER, JR., AND J. M. BARNES,** "Two- and Three-Dimensional Neutronics Analyses of the TFTR Neutral-Beam Injectors," p. 655. (6.26)

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INGERSOLL, D. T., AND B. W. WEHRING,** "Sensitivity Analysis Applied to an Integral Test of Niobium Cross Sections" (3.28)

PEREY, F. G., "Expectations for ENDF/B-V Covariance Files: Coverage, Strengths and Limitations" (2.14)

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BROADHEAD, B. L.,** H. L. DODDS, JR.,** J. L. LUCIUS,** F. G. PEREY, AND C. R. WEISBIN, "Calculation of the Flux Covariance Matrix for the National Bureau of Standards Intermediate Energy Standard Neutron Field (NBS-LSNF)," p. 590. (3.20)

CHILDS, R. L.,** J. L. LUCIUS,** V. C. BAKER,** AND J. D. DRISCHLER, "Sensitivity and Uncertainty Analysis for Mixed-Oxide Thermal Reactor Parameters," p. 687. (3.22)

RHOADES, W. A., R. L. CHILDS,** AND W. W. ENCLE, JR., "Comparison of Rebalance Stabilization Methods for Two-Dimensional Transport Calculations," p. 583. (5.3)

TRUBEY, D. K., D. E. BARTINE, J. V. PACE III,** AND J. R. KNIGHT,** "Development of a Mid-Head Tissue Dose Response Function," p. 582. (6.43)

WILLIAMS, M. L., AND S. RAMAN,** "Calculational and Experimental Analysis of a ${}^{239}\text{Pu}$ Sample Irradiated in EBR-II," p. 706. (6.7)

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ENGLE, W. W., JR., F. R. MYNATT, AND R. K. DISNEY,** "Methods for U.S. Shielding Calculations - Applications of FFTF and CRBR Designs" (5.12)

MAIENSCHEIN, F. C., R. E. MAERKER, AND F. J. MUCKENTHALER, "U. S. Integral and Benchmark Experiments" (4.15)

MAIENSCHEIN, F. C., AND F. J. MUCKENTHALER, "Shielding Experimental Methods" (4.14)

OBLow, E. M., AND R. E. MAERKER, "Investigation of Steel-Sodium-Iron Shields" (4.16)

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RHOADES, W. A., AND E. T. TOMLINSON,** "A Comparison of Codes for Deep Penetration Transport Studies" (5.4)

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AUTHOR INDEX

- Alsmiller, F. S. 1.33, 6.45
 Alsmiller, R. G., Jr. 1.33, 6.22, 6.26,
 6.27, 6.28, 6.31, 6.33, 6.39, 6.40,
 6.45, 6.46
 Amburgey, J. D. 6.32, 6.36
 Baker, V. C. 3.22, 3.29
 Barish, J. 1.33, 6.28, 6.32, 6.33, 6.39
 Barnes, J. M. 6.20, 6.23, 6.24, 6.26,
 6.29, 6.30, 6.31, 6.46
 Bartine, D. E. 6.11, 6.12, 6.13, 6.15,
 6.16, 6.17, 6.39, 6.43
 Begovich, C. L. 6.45
 Bertini, H. W. 6.37, 6.45
 Bishop, B. L. 6.24, 6.25, 6.32, 6.34,
 6.36, 6.47
 Bott, T. F. 7.7, 7.8
 Brill, A. E. 7.4
 Broadhead, B. L. 3.20, 3.21
 Burns, T. J. 6.12, 6.13, 6.14, 6.17,
 6.33
 Cacuci, D. G. 3.30, 5.13
 Chapman, G. T. 1.16
 Childs, R. L. 3.22, 3.27, 5.1, 5.3
 Cleveland, J. C. 6.17
 Clifford, C. E. 4.4, 4.5, 4.7, 4.8
 Cramer, S. N. 4.1, 4.2, 5.10, 5.11
 Culkowski, A. H. 6.37
 Cullen, D. E. 3.2
 Cunningham, G. W. 5.15, 5.16
 Dearing, J. F. 6.44
 de Saussure, G. 1.2, 1.7, 1.8, 1.9,
 1.10, 1.11, 1.12, 1.13, 2.3, 2.6, 2.7
 Dickens, J. K. 1.26, 1.27, 1.28, 1.29
 1.30, 1.31
 Difilippo, F. C. 1.7, 1.8, 1.9, 1.11,
 1.12, 2.3, 2.5, 2.6
 Diggs, B. P. 3.3
 Disney, R. K. 5.12
 Dodd, H. L. 3.20
 Doyas, R. J. 3.2
 Drischler, J. D. 3.10, 3.17, 3.18,
 3.22, 3.23
 Emery, J. F. 1.26, 1.27, 1.28, 1.29
 Emmett, M. B. 4.11, 5.5
 Engel, J. R. 6.44
 Engle, W. W., Jr. 5.1, 5.2, 5.3, 5.12, 6.2
 Erickson, P. J. 5.8
 Flanagan, G. F. 6.5, 6.8, 7.7
 Ford, W. E. III 3.3
 Fowler, T. B. 5.14, 5.15, 5.16, 5.17, 5.18,
 5.19, 5.20, 6.1
 Fu, C. Y. 2.1, 2.2, 2.8
 Gabriel, T. A. 6.22, 6.23, 6.24, 6.25, 6.32,
 6.33, 6.34, 6.35, 6.36, 6.39, 6.47
 Gentry, J. C. 1.25
 Gilai, D. 3.15, 3.16
 Gmur, K. 5.15
 Gove, N. B. 6.37
 Graf, H. 6.15
 Greene, N. M. 3.3, 7.7
 Greenspan, E. 3.15, 3.16, 6.41
 Grimes, W. R. 6.44
 Gurney, J. 7.3, 7.5
 Gustin, A. B. 7.3
 Guthrie, M. P. 6.37
 Gwin, R. 1.6, 2.9
 Haas, P. M. 7.7, 7.8
 Halperin, J. 1.10
 Hardy, J., Jr. 3.6, 3.7
 Harvey, J. A. 1.18
 Hendrickson, H. R. 3.1
 Henry, A. F. 5.23
 Hermann, O. W. 6.37
 Hill, N. W. 1.18
 Hull, J. L. 4.13
 Ingersoll, D. T. 3.28, 4.12, 5.6, 6.15
 Ingle, R. W. 1.6, 1.7, 1.8, 1.9, 1.11, 1.12
 Jenal, J. P. 5.3
 Jermann, M. 6.15
 Kalbfelz, J. M. 6.12, 6.13
 Karni, Y. 3.15
 Kasten, P. R. 6.16
 Kinney, W. E. 1.16
 Knight, J. R. 6.43
 Landay, M. W. 7.4
 Larson, D. C. 1.17, 1.18, 1.19
 Lillie, R. A. 6.26, 6.27, 6.42

BLANK PAGE

- Lin, S. C. 6.3
 Love, T. A. 1.26, 1.27, 1.28, 1.29
 Loyd, D. L. 6.4, 6.5
 Lucius, J. L. 3.20, 3.22, 3.23, 3.26, 6.18, 6.19, 6.20, 6.21
 MacFarlane, R. E. 3.4
 Macklin, R. L. 1.10, 2.7
 Maerker, R. E. 3.5, 4.5, 4.6, 4.7, 4.9, 4.10, 4.15, 4.16
 Maienschein, F. C. 4.14, 4.15
 Manning, J. J. 4.13
 Marable, J. H. 3.6, 3.7, 3.10, 3.12, 3.17, 3.18, 3.19, 3.21, 3.24, 3.25, 3.26, 3.27, 3.29
 Maskewitz, B. F. 7.1, 7.2, 7.5, 7.6
 McAdoo, J. W. 6.8
 McClain, W. J. 7.4, 7.5
 McCombie, C. 6.15
 McConnell, J. W. 1.22, 1.23, 1.26, 1.27, 1.28, 1.29, 1.31
 McKnight, R. D. 3.6, 3.7
 Mihalcz, J. T. 6.27
 Morgan, G. L. 1.14, 1.17, 1.19, 1.20, 1.21, 1.22, 1.23, 4.3
 Muckenthaler, F. J. 4.4, 4.5, 4.7, 4.8, 4.10, 4.11, 4.13, 4.14, 4.15
 Mynatt, F. R. 5.12
 Northcutt, K. J. 1.26, 1.27, 1.28, 1.29
 Oblow, E. M. 1.15, 3.9, 3.10, 3.12, 3.13, 3.14, 3.16, 3.26, 4.1, 4.2, 4.16, 6.18, 6.19, 6.20, 6.21
 Olsen, D. K. 1.7, 1.9, 1.11, 1.12, 1.13, 1.22, 1.23, 1.32, 2.3, 2.6
 Pace, J. V. III 6.43
 Paik, N. C. 3.4
 Peelle, R. W. 1.2, 1.26, 1.27, 1.28, 1.29, 3.26
 Perey, C. M. 1.16
 Perey, F. G. 1.14, 1.15, 1.16, 1.25, 2.1, 2.10, 2.11, 2.12, 2.13, 2.14, 2.15, 3.20
 Perez, R. B. 1.7, 1.8, 1.9, 1.10, 1.11, 1.12, 1.13, 2.3, 2.4, 2.5, 2.6
 Plechaty, E. F. 3.2
 Price, R. R. 7.4
 Ragheb, M. M. H. 6.29
 Raman, S. 6.7
 Rhoades, W. A. 5.1, 5.2, 5.3, 5.4, 5.5, 5.8, 6.44
 Richmond, R. 6.15
 Robinson, J. C. 6.3, 6.4
 Roussin, R. W. 3.3, 7.1, 7.2, 7.3
 Rowland, D. M. 6.12, 6.13
 Sadler, F. B. 5.7, 5.9
 Saltmarsh, M. J. 6.29
 Salvatores, M. 3.10
 Santoro, R. T. 6.18, 6.19, 6.20, 6.21, 6.22, 6.23, 6.24, 6.26, 6.29, 6.30, 6.31, 6.42, 6.46
 Schmock, U. 6.15
 Seki, Y. 6.18, 6.19, 6.20, 6.21
 Selby, D. L. 5.7, 6.3, 6.4, 6.5, 6.6
 Seth, S. 6.15
 Simpson, D. B. 5.1, 5.8
 Slater, C. O. 4.11, 4.12
 Spencer, R. R. 1.6
 Spiwak, I. 6.17
 Stevens, P. N. 4.4
 Tang, J. S. 6.31
 Thomas, W. E. 6.17
 Todd, J. H. 1.3, 1.4, 1.5, 1.6
 Tomlinson, E. T. 3.23, 5.4, 5.21
 Trubey, D. K. 3.1, 6.43, 7.1, 7.2, 7.3
 Vondy, D. R. 5.14, 5.15, 5.16, 5.17, 5.18, 5.19, 5.20, 6.1, 6.9, 6.10
 Weaver, H. 1.6, 1.11, 1.26, 1.27, 1.28, 1.29
 Wehring, B. W. 3.28
 Weigmann, H. 1.18
 Weisbin, C. R. 3.2, 3.3, 3.4, 3.6, 3.7, 3.8, 3.10, 3.11, 3.17, 3.18, 3.19, 3.20, 3.24, 3.25, 3.26, 3.27
 Weston, L. W. 1.1, 1.3, 1.4, 1.5, 1.24
 Wharton, D. M. 1.32
 White, J. E. 3.2, 3.3
 White, J. R. 3.12, 6.12, 6.13, 6.14, 6.17
 Wiffen, F. W. 6.47
 Williams, L. R. 6.11
 Williams, M. L. 3.11, 3.12, 4.12, 5.8, 5.9, 5.22, 6.2, 6.7, 6.8, 6.15
 Worley, B. A. 5.23, 6.10
 Wright, J. B. 3.3
 Wright, R. Q. 3.3
 Yoon, W. 4.4

ENGINEERING PHYSICS DIVISIONAL MANPOWER

	Manyyears	
	EPD ^a	Loanees ^{a,b}
Liquid-Metal Fast Breeder Reactor - Physics	15	10
Liquid-Metal Fast Breeder Reactor - Safety	3	9.0
Clinch River Breeder Reactor Project	2	0.5
Gas-Cooled Breeder Reactors	4	-
Alternate Reactors	5	1.5
High-Temperature Gas Reactor Development	1	-
Nuclear Regulatory Commission (NRC)	3	0.5
Fusion Energy Studies	4	0.5
Basic Energy Sciences and High Energy Physics	11	13
Defense Nuclear Agency (DNA)	4	3.0
Electric Power Research Institute (EPRI)	4	1.5
Other - DOE and Work for Others	2	0.5
Total	63	40

^aManyyears rounded off.

^bManpower from Instrumentation and Controls Division, Computer Sciences Division, Plant and Equipment Division, and Analytical Chemistry Division.