Toshikazu Takeda

List of Publications by Year
in descending order

142 papers	871 citations	687363 13 h-index	642732 23 g-index
142 all docs	142 docs citations	142 times ranked	222 citing authors

#	Article	IF	Citations
1	Revisitation of the studies on covariance data and adjustment analysis: A tribute to M. Salvatores for his great works and remaining future tasks. Annals of Nuclear Energy, 2021, 151, 107895.	1.8	2
2	Estimation of reactivity feedback and determination of safety criteria of inherent-safety fast reactors in unprotected transients based on the asymptotic approximation. Annals of Nuclear Energy, 2021, 164, 108597.	1.8	2
3	An estimation of cross-section covariance data suitable for predicting neutronics parameters uncertainty. Annals of Nuclear Energy, 2020, 145, 107534.	1.8	5
4	Development of an improved quasi-static transient analysis code based on three-dimensional Sn nodal transport theory for fast reactor. Annals of Nuclear Energy, 2020, 143, 107499.	1.8	2
5	Cross-section-induced uncertainty evaluation of MA sample irradiation test calculations with consideration of dosimeter data. Annals of Nuclear Energy, 2019, 130, 118-123.	1.8	1
6	Effect of void propagation to sodium void reactivity in transient analyses of fast reactors with sodium-plenum. Annals of Nuclear Energy, 2018, 119, 175-179.	1.8	5
7	A new cross section adjustment method of removing systematic errors in fast reactors. Annals of Nuclear Energy, 2017, 109, 698-704.	1.8	3
8	Uncertainty analysis of minor actinides transmutation in fast reactor cores. Annals of Nuclear Energy, 2017, 101, 591-599.	1.8	3
9	Minor actinides transmutation performance in a fast reactor. Annals of Nuclear Energy, 2016, 95, 48-53.	1.8	23
10	Burn up management strategy to design Ultra Long Life fast reactors. Progress in Nuclear Energy, 2013, 62, 46-49.	2.9	1
11	Burnup management simulator for small and medium sized reactors. Annals of Nuclear Energy, 2011, 38, 2840-2842.	1.8	1
12	Simple and Efficient Parallelization Method for MOC Calculation. Journal of Nuclear Science and Technology, 2010, 47, 90-102.	1.3	5
13	Simple and Efficient Parallelization Method for MOC Calculation. Journal of Nuclear Science and Technology, 2010, 47, 90-102.	1.3	1
14	Applicability of Constant Flux Approximation in Method of Characteristics with Filtering to Tiny Regions. Journal of Nuclear Science and Technology, 2009, 46, 953-964.	1.3	0
15	A New Uncertainty Reduction Method for Fuel Fabrication Process with Erbia-Bearing Fuel. Journal of Nuclear Science and Technology, 2009, 46, 226-231.	1.3	0
16	Evaluation of Prediction Error Reduction for Breeding Ratio by Bias Factor Method with Use of Experimental Result on Basic Reaction Rate Ratio. Journal of Nuclear Science and Technology, 2008, 45, 211-216.	1.3	0
17	Prediction Accuracy Improvement of Neutronic Characteristics of a Breeding Light Water Reactor Core by Extended Bias Factor Methods with Use of FCA-XXII-1 Critical Experiments. Journal of Nuclear Science and Technology, 2008, 45, 288-303.	1.3	5
18	Theoretical Study on New Bias Factor Methods to Effectively Use Critical Experiments for Improvement of Prediction Accuracy of Neutronic Characteristics. Journal of Nuclear Science and Technology, 2007, 44, 1509-1517.	1.3	17

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19	Subgroup Parameters based on Orthogonal Factorization. Journal of Nuclear Science and Technology, 2007, 44, 36-42.	1.3	3
20	A Fast and Simplified Method to Calculate Exponential of Burnup Matrix. Journal of Nuclear Science and Technology, 2007, 44, 1379-1384.	1.3	0
21	Theoretical Study on New Bias Factor Methods to Effectively Use Critical Experiments for Improvement of Prediction Accuracy of Neutronic Characteristics. Journal of Nuclear Science and Technology, 2007, 44, 1509-1517.	1.3	3
22	ICONE15-10371 Application of Bias Factor Method with Use of Virtual Experimental Value to Prediction Uncertainty Reduction in Void Reactivity Worth of Breeding Light Water Reactor. The Proceedings of the International Conference on Nuclear Engineering (ICONE), 2007, 2007.15, _ICONE1510-_ICONE1510.	0.0	0
23	Extension of Effective Cross Section Calculation Method for Neutron Transport Calculations in Particle-dispersed Media. Journal of Nuclear Science and Technology, 2006, 43, 77-87.	1.3	14
24	Generalized Bias Factor Method for Accurate Prediction of Neutronics Characteristics. Journal of Nuclear Science and Technology, 2006, 43, 1465-1470.	1.3	16
25	Sensitivity Analysis based on Transport Theory. Journal of Nuclear Science and Technology, 2006, 43, 743-749.	1.3	13
26	A Complement Proposal for Optimization of Subgroup Parameters. Journal of Nuclear Science and Technology, 2006, 43, 765-773.	1.3	1
27	Generalized Bias Factor Method for Accurate Prediction of Neutronics Characteristics. Journal of Nuclear Science and Technology, 2006, 43, 1465-1470.	1.3	5
28	A Complement Proposal for Optimization of Subgroup Parameters. Journal of Nuclear Science and Technology, 2006, 43, 765-773.	1.3	3
29	Extension of Effective Cross Section Calculation Method for Neutron Transport Calculations in Particle-dispersed Media. Journal of Nuclear Science and Technology, 2006, 43, 77-87.	1.3	2
30	Effect of Anisotropic Scattering in UO2 and MOX Fueled LWR Cells and Cores. AIP Conference Proceedings, 2005, , .	0.4	0
31	Effect of Nonuniform 3â€D Void Distribution Within BWR Fuel Assemblies on Neutronic Characteristics. Transport Theory and Statistical Physics, 2004, 33, 299-310.	0.4	0
32	Sensitivity Analysis for Multiplication Factor Change of LWR Cell Caused by the Differences between JENDL-3.2 and JENDL-3.3. Journal of Nuclear Science and Technology, 2004, 41, 163-170.	1.3	1
33	Sensitivity Analysis for Multiplication Factor Change of LWR Cell Caused by the Differences between JENDL-3.2 and JENDL-3.3. Journal of Nuclear Science and Technology, 2004, 41, 163-170.	1.3	1
34	Neutron Anisotropic Scattering Effect in Heterogeneous Cell Calculations of Light Water Reactors. Journal of Nuclear Science and Technology, 2003, 40, 464-480.	1.3	16
35	The Characteristics and Subgroup Methods in Square Light Water Reactor Cell Calculations. Nuclear Science and Engineering, 2003, 143, 61-80.	1.1	6
36	Neutron Anisotropic Scattering Effect in Heterogeneous Cell Calculations of Light Water Reactors. Journal of Nuclear Science and Technology, 2003, 40, 464-480.	1.3	12

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37	Evaluation of Eigenvalue Separation by the Monte Carlo Method. Journal of Nuclear Science and Technology, 2002, 39, 129-137.	1.3	3
38	Effect of Moderator Density Distribution of Annular Flow on Fuel Assembly Neutronic Characteristics in Boiling Water Reactor Cores. Journal of Nuclear Science and Technology, 2002, 39, 487-498.	1.3	8
39	Effect of Radial Void Distribution within Fuel Assembly on Assembly Neutronic Characteristics. Journal of Nuclear Science and Technology, 2002, 39, 90-100.	1.3	12
40	Direction and Region Dependent Cross Sections for Use to MOX Fuel Analysis. Journal of Nuclear Science and Technology, 2002, 39, 1057-1060.	1.3	3
41	Evaluation of Eigenvalue Separation by the Monte Carlo Method.. Journal of Nuclear Science and Technology, 2002, 39, 129-137.	1.3	2
42	Effect of Moderator Density Distribution of Annular Flow on Fuel Assembly Neutronic Characteristics in Boiling Water Reactor Cores.. Journal of Nuclear Science and Technology, 2002, 39, 487-498.	1.3	3
43	Effect of Radial Void Distribution within Fuel Assembly on Assembly Neutronic Characteristics.. Journal of Nuclear Science and Technology, 2002, 39, 90-100.	1.3	4
44	Effective Convergence of Fission Source Distribution in Monte Carlo Simulation. Journal of Nuclear Science and Technology, 2001, 38, 324-329.	1.3	26
45	Nonlinear Behavior under Regional Neutron Flux Oscillations in BWR Cores. Journal of Nuclear Science and Technology, 2001, 38, 312-323.	1.3	7
46	Development and Verification of an Efficient Spatial Neutron Kinetics Method for Reactivity-Initiated Event Analyses. Journal of Nuclear Science and Technology, 2001, 38, 492-502.	1.3	12
47	Nonlinear Behavior under Regional Neutron Flux Oscillations in BWR Cores.. Journal of Nuclear Science and Technology, 2001, 38, 312-323.	1.3	5
48	Effective Convergence of Fission Source Distribution in Monte Carlo Simulation.. Journal of Nuclear Science and Technology, 2001, 38, 324-329.	1.3	3
49	Development and Verification of an Efficient Spatial Neutron Kinetics Method for Reactivity-Initiated Event Analyses.. Journal of Nuclear Science and Technology, 2001, 38, 492-502.	1.3	3
50	Analysis of Differences in Void Coefficient Predictions for Mixed-Oxideâ€"Fueled Tight-Pitch Light Water Reactor Cells. Nuclear Science and Engineering, 2000, 135, 1-22.	1.1	10
51	Rapid Estimation of Core-Power Ratio in Coupled-Core System by Rod Drop Method. Journal of Nuclear Science and Technology, 2000, 37, 565-571.	1.3	4
52	Reaction Rate Calculation in Fast Reactor Blanket Using Multiband SnTheory. Journal of Nuclear Science and Technology, 2000, 37, 428-435.	1.3	8
53	Minor Actinides Incineration by Loading Moderated Targets in Fast Reactor. Journal of Nuclear Science and Technology, 2000, 37, 380-386.	1.3	0
54	Reaction Rate Calculation in Fast Reactor Blanket Using Multiband Sn Theory.. Journal of Nuclear Science and Technology, 2000, 37, 428-435.	1.3	5

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55	Rapid Estimation of Core-Power Ratio in Coupled-Core System by Rod Drop Method.. Journal of Nuclear Science and Technology, 2000, 37, 565-571.	1.3	1
56	Estimation of Error Propagation in Monte-Carlo Burnup Calculations. Journal of Nuclear Science and Technology, 1999, 36, 738-745.	1.3	21
57	Spatial-harmonic Neutron Spectrum Effect on Frequency-domain Modal Analysis of Regional Stability in BWR. Journal of Nuclear Science and Technology, 1999, 36, 81-94.	1.3	1
58	A Multiband Method with Resonance Interference Effect. Nuclear Science and Engineering, 1999, 131, 401-410.	1.1	6
59	Analysis of First-Harmonic Eigenvalue Separation Experiments on KUCA Coupled-Core. Journal of Nuclear Science and Technology, 1998, 35, 216-225.	1.3	9
60	Transport Calculations of MOX and UO₂ Pin Cells by the Method of Characteristics. Journal of Nuclear Science and Technology, 1998, 35, 874-885.	1.3	15
61	Transport Calculations of MOX and UO2 Pin Cells by the Method of Characteristics.. Journal of Nuclear Science and Technology, 1998, 35, 874-885.	1.3	7
62	Resonance Calculations Using the Multiband Method and Interface Currents. Nuclear Science and Engineering, 1997, 125, 178-187.	1.1	6
63	Parametric Study on Fast Reactors with Low Sodium Void Reactivity by the Use of Zirconium Hydride Layer in Internal Blanket. Journal of Nuclear Science and Technology, 1997, 34, 193-201.	1.3	4
64	Effective Cross Section of238Samples for Analyzing Doppler Effect Measurements in Fast Critical Assembly. Journal of Nuclear Science and Technology, 1997, 34, 1022-1026.	1.3	0
65	Improvement of Fitting Method of Multiband Parameters for Cell Calculations. Journal of Nuclear Science and Technology, 1997, 34, 638-643.	1.3	2
66	Parametric Study on Fast Reactors with Low Sodium Void Reactivity by the Use of Zirconium Hydride Layer in Internal Blanket.. Journal of Nuclear Science and Technology, 1997, 34, 193-201.	1.3	3
67	Doppler Reactivity Calculation for Thermal Reactor Cells by Space-Dependent Multiband Method. Journal of Nuclear Science and Technology, 1996, 33, 604-606.	1.3	6
68	An Improvement of the Transverse Leakage Treatment for the Nodal <i>S_N </i> Transport Calculation Method in Hexagonal-<i>Z</i> Geometry. Journal of Nuclear Science and Technology, 1996, 33, 620-627.	1.3	13
69	Doppler Reactivity Calculation for Thermal Reactor Cells by Space-Dependent Multiband Method.. Journal of Nuclear Science and Technology, 1996, 33, 604-606.	1.3	2
70	An Improvement of the Transverse Leakage Treatment for the Nodal SN Transport Calculation Method in Hexagonal-Z Geometry.. Journal of Nuclear Science and Technology, 1996, 33, 620-627.	1.3	4
71	Monte-Carlo/Collision Probability Hybrid Method for LWR Fuel Assembly Burnup Calculations. Journal of Nuclear Science and Technology, 1995, 32, 683-690.	1.3	0
72	A New Nodal SNTransport Method for Three-Dimensional Hexagonal Geometry. Journal of Nuclear Science and Technology, 1994, 31, 497-509.	1.3	21

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73	Approximate Calculation Method for Second Order Sensitivity Coefficient. Journal of Nuclear Science and Technology, 1994, 31, 1151-1159.	1.3	1
74	New Control Rod Homogenization Method for Fast Reactors. Journal of Nuclear Science and Technology, 1994, 31, 647-653.	1.3	7
75	Fast Reactor of Negative Na Void Reactivity and Its Transient Behavior. Journal of Nuclear Science and Technology, 1994, 31, 748-750.	1.3	2
76	Three-Dimensional Transient Analysis of Fast Reactors with Improved Coarse Mesh Method. Journal of Nuclear Science and Technology, 1994, 31, 12-23.	1.3	3
77	Approximate Calculation Method for Second Order Sensitivity Coefficient.. Journal of Nuclear Science and Technology, 1994, 31, 1151-1159.	1.3	1
78	Three-Dimensional Transient Analysis of Fast Reactors with Improved Coarse Mesh Method.. Journal of Nuclear Science and Technology, 1994, 31, 12-23.	1.3	1
79	A New Nodal SN Transport Method for Three-Dimensional Hexagonal Geometry.. Journal of Nuclear Science and Technology, 1994, 31, 497-509.	1.3	12
80	Optimization of Internal Blanket Configuration of Large Fast Reactor. Journal of Nuclear Science and Technology, 1993, 30, 481-484.	1.3	6
81	A Combined Cross-Section Adjustment and Bias Factor Method for Accurate Prediction of Fast Breeder Reactor Core Performance Parameters. Nuclear Science and Engineering, 1993, 114, 64-75.	1.1	8
82	Application of Multiband Method to Pin Cell Calculations. Journal of Nuclear Science and Technology, 1993, 30, 1167-1174.	1.3	10
83	Application of Multiband Method to Pin Cell Calculations.. Journal of Nuclear Science and Technology, 1993, 30, 1167-1174.	1.3	2
84	Prediction Uncertainty Analysis of Neutronic Properties of Fast Breeder Reactor by Use of Sensitivity-Based Methodology. Journal of Nuclear Science and Technology, 1992, 29, 1033-1042.	1.3	3
85	Development of Transport Code for Hexagonal Geometry. Journal of Nuclear Science and Technology, 1992, 29, 1143-1151.	1.3	5
86	Application of Multiband Method to KUCA Tight-Pitch Lattice Analysis. Journal of Nuclear Science and Technology, 1991, 28, 863-869.	1.3	23
87	3-D Neutron Transport Benchmarks. Journal of Nuclear Science and Technology, 1991, 28, 656-669.	1.3	90
88	Uncertainty Evaluation of Burnup Properties of Large Fast Reactors Using Data Adjustment Method. Journal of Nuclear Science and Technology, 1991, 28, 275-284.	1.3	4
89	3-D Neutron Transport Benchmarks.. Journal of Nuclear Science and Technology, 1991, 28, 656-669.	1.3	32
90	Application of Multiband Method to KUCA Tight-Pitch Lattice Analysis.. Journal of Nuclear Science and Technology, 1991, 28, 863-869.	1.3	1

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91	Three-Dimensional Multigroup Diffusion Code ANDEX Based on Nodal Method for Cartesian Geometry. Journal of Nuclear Science and Technology, 1990, 27, 350-359.	1.3	7
92	Sensitivity Analysis of Fast Reactor Core Performance Parameters Calculated by JENDL-2 and JENDL-3. Journal of Nuclear Science and Technology, 1990, 27, 581-583.	1.3	4
93	Sensitivity analysis of fast reactor core performance parameters calculated by JENDL-2 and JENDL-3.. Journal of Nuclear Science and Technology, 1990, 27, 581-583.	1.3	1
94	Diffusion Calculation Model for Streaming Effects in Low Density Channels in LMFBRs. Journal of Nuclear Science and Technology, 1989, 26, 712-720.	1.3	5
95	Prediction Uncertainty Evaluation Methods of Core Performance Parameters in Large Liquid-Metal Fast Breeder Reactors. Nuclear Science and Engineering, 1989, 103, 157-165.	1.1	43
96	Evaluation of Neutron Streaming in Fast Breeder Reactor Fuel Subassembly by Double Heterogeneous Modeling. Nuclear Science and Engineering, 1989, 101, 179-184.	1.1	3
97	Some Nuclear Characteristics of Thorium Fueled Light Water Reactors. Journal of Nuclear Science and Technology, 1988, 25, 943-947.	1.3	1
98	Two-Dimensional Cell Heterogeneity Effect in Analysis of Fast Critical Assemblies. Nuclear Science and Engineering, 1988, 98, 128-137.	1.1	2
99	An Improved Cell Calculation Method for Liquid-Metal Fast Breeder Reactor Blanket Analysis. Nuclear Science and Engineering, 1988, 98, 118-127.	1.1	1
100	Analysis of Large Liquid-Metal Fast Breeder Reactor Critical Experiments by Improved Methods. Nuclear Science and Engineering, 1988, 100, 538-548.	1.1	8
101	Coarse-Mesh Three-Dimensional Transport Calculation Method. Journal of Nuclear Science and Technology, 1987, 24, 960-964.	1.3	0
102	Determination of Cell Averaged Diffusion Constants Based on Transport/Diffusion Perturbation Theory. Journal of Nuclear Science and Technology, 1987, 24, 999-1008.	1.3	0
103	Sensitivity Analysis of Cell Neutronic Parameters in High-Conversion Light-Water Reactors. Journal of Nuclear Science and Technology, 1987, 24, 610-620.	1.3	7
104	New Calculational Method of Sensitivity Coefficients of Cell Parameters and Its Application. Journal of Nuclear Science and Technology, 1986, 23, 681-694.	1.3	11
105	Sensitivity of Sodium Void Worth in Fast Reactors to Cross Sections. Journal of Nuclear Science and Technology, 1986, 23, 181-184.	1.3	1
106	Three-Dimensional Transport Correction in Fast Reactor Core Analysis. Journal of Nuclear Science and Technology, 1986, 23, 849-858.	1.3	10
107	Development of Three-Dimensional Transport and Diffusion Codes Based on Nodal Method. Journal of Nuclear Science and Technology, 1986, 23, 565-568.	1.3	0
108	Application of Depletion Perturbation Theory to Fuel Loading Optimization. Journal of Nuclear Science and Technology, 1986, 23, 1-10.	1.3	3

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109	Sensitivity analysts of jendl library using fast reactor physics parameters. Radiation Effects, 1986, 96, 265-268.	0.4	0
110	Application of depletion perturbation theory to fuel loading optimization.. Journal of Nuclear Science and Technology, 1986, 23, 1-10.	1.3	2
111	Three-dimensional transport correction in fast reactor core analysis.. Journal of Nuclear Science and Technology, 1986, 23, 849-858.	1.3	3
112	Burnup Sensitivity Analysis in a Fast Breeder Reactorâ€"Part II: Prediction Accuracy of Burnup Characteristics. Nuclear Science and Engineering, 1985, 91, 11-33.	1.1	7
113	Three-Dimensional Transport Calculation Method for Eigenvalue Problems Using Diffusion Synthetic Acceleration. Journal of Nuclear Science and Technology, 1985, 22, 841-850.	1.3	26
114	Comparison of Cell Models for Analysis of Heterogeneous Fast Critical Assembly. Journal of Nuclear Science and Technology, 1985, 22, 755-764.	1.3	1
115	Effective Spatial Homogenization with Neutron Leakage Effect for FBR Control Rods. Journal of Nuclear Science and Technology, 1985, 22, 320-327.	1.3	4
116	Three-dimensional transport calculation method for eigenvalue problems using diffusion synthetic acceleration.. Journal of Nuclear Science and Technology, 1985, 22, 841-850.	1.3	7
117	Drift and Diffusion Coefficients for Asymmetric Cells of Fast Critical Assembly. Journal of Nuclear Science and Technology, 1984, 21, 21-31.	1.3	2
118	Extension of Fission Product Model for Use in Lattice Calculation of Thorium Fueled BWR. Journal of Nuclear Science and Technology, 1983, 20, 523-525.	1.3	1
119	Effect of Neutron Leakage in Cell Calculations of Fast Criticai Assembly. Journal of Nuclear Science and Technology, 1981, 18, 645-648.	1.3	1
120	Unified Diffusion Coefficient for Analysis of Sodium-Void Worth in Fast Critical Assembly with Control-Rod Channels. Journal of Nuclear Science and Technology, 1981, 18, 93-115.	1.3	7
121	Determination of Effective Diffusion Parameters in Thermal Reactor Assemblies. Journal of Nuclear Science and Technology, 1980, 17, 44-56.	1.3	0
122	Application of Improved Coarse Mesh Method to BWR Core Calculations. Journal of Nuclear Science and Technology, 1980, 17, 150-153.	1.3	3
123	Determination of effective diffusion parameters in thermal reactor assemblies.. Journal of Nuclear Science and Technology, 1980, 17, 44-56.	1.3	1
124	Extension of Askew's Coarse Mesh Method to Few-Group Problems for Calculating Two-Dimensional Power Distribution in Fast Breeder Reactors. Journal of Nuclear Science and Technology, 1978, 15, 523-532.	1.3	21
125	Extension of Askew's Coarse Mesh Method to Few-Group Problems for Calculating Two-Dimensional Power Distribution in Fast Breeder Reactors. Journal of Nuclear Science and Technology, 1978, 15, 523-532.	1.3	13
126	Synthesis Method of Integral Type. Journal of Nuclear Science and Technology, 1973, 10, 207-213.	1.3	0

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127	Anisotropic Diffusion Coefficient in a Cylindrical Cell by Integral Transport Theory. Journal of Nuclear Science and Technology, 1973, 10, 651-662.	1.3	7
128	Effect of Higher Order Anisotropy on Disadvantage Factor in a Cell. Journal of Nuclear Science and Technology, 1972, 9, 249-250.	1.3	3
129	Comparison of the Anisotropic Diffusion Coefficients Based on the Benoist and Deniz Formulas. Journal of Nuclear Science and Technology, 1972, 9, 682-685.	1.3	1
130	Effect of Anisotropic Scattering in a Square Cell with Thin Moderator. Journal of Nuclear Science and Technology, 1972, 9, 53-54.	1.3	3
131	Calculation of the Anisotropic Diffusion Coefficient. Journal of Nuclear Science and Technology, 1972, 9, 697-704.	1.3	8
132	Effect of Higher Order Anisotropy on Disadvantage Factor in a Cell. Journal of Nuclear Science and Technology, 1972, 9, 249-250.	1.3	1
133	Effect of Anisotropic Scattering in a Square Cell with Thin Moderator. Journal of Nuclear Science and Technology, 1972, 9, 53-54.	1.3	1
134	Comparison of the Anisotropic Diffusion Coefficients Based on the Benoist and Deniz Formulas. Journal of Nuclear Science and Technology, 1972, 9, 682-685.	1.3	1
135	Calculation of the Anisotropic Diffusion Coefficient. Journal of Nuclear Science and Technology, 1972, 9, 697-704.	1.3	1
136	Asymptotic Neutron Distribution in a Finite Cylinder. Journal of Nuclear Science and Technology, 1971, 8, 133-140.	1.3	0
137	Improvement to the Calculation of Collision Probabilities in Annular System and Its Application to Cluster Geometry. Journal of Nuclear Science and Technology, 1971, 8, 503-512.	1.3	2
138	Moderator Correction for a Wide Resonance by the Greuling-Goertzel Approximation. Journal of Nuclear Science and Technology, 1971, 8, 649-650.	1.3	0
139	Disadvantage Factor in a Cell with Thin Moderator. Journal of Nuclear Science and Technology, 1971, 8, 539-545.	1.3	3
140	Anisotropic Collision Probabilities in Cell Problems. Journal of Nuclear Science and Technology, 1971, 8, 663-672.	1.3	8
141	Disadvantage Factor in a Cell with Thin Moderator. Journal of Nuclear Science and Technology, 1971, 8, 539-545.	1.3	2
142	Anisotropic Collision Probabilities in Cell Problems. Journal of Nuclear Science and Technology, 1971, 8, 663-672.	1.3	2