Dalin Zhang

List of Publications by Year in Descending Order

162 papers	1,276 citations	17 h-index	27 g-index
188 ext. papers	1,688 ext. citations	2.6 avg, IF	4.71 L-index

#	Paper	IF	Citations
162	Thermal-electrical coupling characteristic analysis of the heat pipe cooled reactor with static thermoelectric conversion. *Annals of Nuclear Energy*, **2022**, 168, 108870	1.7	0
161	Development and application of TaSNAM 2.0 for advanced pressurized water reactor. *Annals of Nuclear Energy*, **2022**, 166, 108801	1.7	2
160	Study on startup characteristics of prototype once-through steam generator for China fast reactor. *International Journal of Advanced Nuclear Reactor Design and Technology*, **2022**, 4, 26-35	0.6	0
159	Fluoride-salt-cooled high-temperature reactors: Review of historical milestones, research status, challenges, and outlook. *Renewable and Sustainable Energy Reviews*, **2022**, 161, 112345	16.2	2
158	Molecular dynamics simulation of the evaporation of liquid sodium film in the presence of non-condensable gas. *Annals of Nuclear Energy*, **2022**, 170, 109005	1.7	0
157	Numerical simulation on thermal-hydraulic and thermoelectric characteristics of the TOPAZ-II reactor core. *International Journal of Energy Research*, **2021**, 45, 12159-12172	4.5	1
156	Preliminary thermal-hydraulic influence of the poloidal nonuniform neutron wall loading on the CFETR HCCB blanket. *Fusion Engineering and Design*, **2021**, 165, 112232	1.7	1
155	Numerical study on improved design of passive residual heat removal system for China pressurizer reactor. *Nuclear Engineering and Design*, **2021**, 375, 111087	1.8	0
154	Thermal-hydraulic analysis of a leadBismuth small modular reactor under moving conditions. *Annals of Nuclear Energy*, **2021**, 154, 108116	1.7	1
153	Start-up characteristics for the test facility of a prototype sodium heated OTSG. *Nuclear Engineering and Design*, **2021**, 378, 111154	1.8	1
152	Neurtonics/Thermal-hydraulic analyses of the CFETR HCCB blanket for multiple operation modes under the poloidal nonuniform neutron wall loading condition. *Fusion Engineering and Design*, **2021**, 168, 112612	1.7	
151	Review of the experimental research on the thermal-hydraulic characteristics in the pebble bed nuclear reactor core and fusion breeder blankets. *International Journal of Energy Research*, **2021**, 45, 11352-11383	4.5	2
150	Preliminary design and analyses of the helium cooled ceramic breeder blanket for CFETR phase II. *International Journal of Energy Research*, **2021**, 45, 11598-11615	4.5	0
149	Design and optimization of passive residual heat removal system for lead-bismuth reactor SVBR-100. *International Journal of Energy Research*, **2021**, 45, 12124-12146	4.5	1
148	Numerical analysis on flow instability of parallel channels in steam generator for sodium-cooled fast reactor. *International Journal of Energy Research*, **2021**, 45, 11943-11956	4.5	3
147	Improvement and validation of a sub-channel analysis code for a lead-cooled reactor with wire spacers. *International Journal of Energy Research*, **2021**, 45, 12029-12046	4.5	2
146	A porous medium model for simulating conjugate heat transfer between wire-wrapped fuel bundles under forced and mixed convection. *Annals of Nuclear Energy*, **2021**, 151, 107906	1.7	4

145	Analysis of the natural circulation test of PHENIX reactor by the THACS code. *Annals of Nuclear Energy*, **2021**, 152, 108022	1.7	1
144	Experimental and numerical investigation on flow characteristics of inter-wrapper channel in LMFBR. *Annals of Nuclear Energy*, **2021**, 151, 107918	1.7	0
143	An experiment-based validation of a system code for prediction of passive natural circulation in sodium-cooled fast reactor. *International Journal of Energy Research*, **2021**, 45, 12093-12109	4.5	0
142	Thermal-hydraulic analysis of gas-cooled space nuclear reactor power system with closed Brayton cycle. *International Journal of Energy Research*, **2021**, 45, 11851-11867	4.5	1
141	Transient thermal-hydraulic analysis of heat pipe cooled passive residual heat removal system of molten salt reactor. *International Journal of Energy Research*, **2021**, 45, 1599-1612	4.5	
140	A practical methodology devoted to pool-type phenomena simulation in safety analysis for sodium-cooled fast reactor. *International Journal of Energy Research*, **2021**, 45, 11868-11881	4.5	0
139	Generating Hexahedral Mesh for Wire-wrapped Fuel Assembly With RBF Mesh Deformation Method. *Frontiers in Energy Research*, **2021**, 8,	3.8	2
138	Development a methodology for evaluating inter-assembly heat transfer effect through reactor core in system safety analysis of sodium-cooled fast reactor. *International Journal of Energy Research*, **2021**, 45, 12258-12271	4.5	0
137	Thermoelectric performance study on a heat pipe thermoelectric generator for micro nuclear reactor application. *International Journal of Energy Research*, **2021**, 45, 12301-12316	4.5	2
136	Phenomena identification and ranking table of station blackout accidents for China sodium cooled fast reactor. *Annals of Nuclear Energy*, **2021**, 157, 108240	1.7	1
135	CFD/RELAP5 coupling analysis of the ISP No. 43 boron dilution experiment. *Nuclear Engineering and Technology*, **2021**, 54, 97-97	2.6	0
134	Flow blockage analysis for fuel assembly in a lead-based fast reactor. *Nuclear Engineering and Technology*, **2021**, 53, 3217-3228	2.6	1
133	Numerical investigation for the heat transfer mechanisms between subchannels of bar rod bundles cooled by liquid sodium. *Annals of Nuclear Energy*, **2021**, 161, 108460	1.7	0
132	Preliminary design and assessment of a heat pipe residual heat removal system for the reactor driven subcritical facility. *Nuclear Engineering and Technology*, **2021**, 53, 3879-3891	2.6	1
131	Validation of TRACE capability to simulate unprotected transients in Sodium Fast Reactor using FFTF LOFWST Test #13. *Annals of Nuclear Energy*, **2021**, 164, 108600	1.7	1
130	Impact of axial power distribution on thermal-hydraulic characteristics for thermionic reactor. *Nuclear Engineering and Technology*, **2021**, 53, 3910-3917	2.6	
129	Low temperature overpressurization analysis for CPR1000. *International Journal of Advanced Nuclear Reactor Design and Technology*, **2021**, 3, 145-153	0.6	
128	Preliminary study of parameter uncertainty influence on thermal design and analysis for sodium heated once-through steam generator. *Nuclear Engineering and Design*, **2020**, 369, 110858	1.8	2

127	A new method for improving the tritium breeding and releasing performance of China Fusion Engineering Test Reactor phase II helium-cooled ceramic breeder blanket. *International Journal of Energy Research*, **2020**, 44, 5977-5989	4.5	4
126	Conceptual design and analysis of a megawatt power level heat pipe cooled space reactor power system. *Annals of Nuclear Energy*, **2020**, 144, 107576	1.7	6
125	Flow Boiling Pressure Drop Characteristics in Rectangular Channels under Uniform and Non-Uniform Heating. *International Journal of Heat and Mass Transfer*, **2020**, 157, 119811	4.9	0
124	Experimental and theoretical study on fluid-structure interaction and non-equilibrium of the flow through micro cracks. *Annals of Nuclear Energy*, **2020**, 142, 107441	1.7	1
123	Experimental analysis of flow and convective heat transfer in the water-cooled packed pebble bed nuclear reactor core. *Progress in Nuclear Energy*, **2020**, 122, 103298	2.3	6
122	Numerical simulation of temperature heterogeneity inside the AP1000 upper plenum and hot leg. *Nuclear Engineering and Design*, **2020**, 362, 110525	1.8	6
121	Thermal-hydraulic analysis of pellet bed reactor for space nuclear electric propulsion. *Annals of Nuclear Energy*, **2020**, 143, 107482	1.7	5
120	Code development and analysis of heat pipe cooled passive residual heat removal system of Molten salt reactor. *Annals of Nuclear Energy*, **2020**, 144, 107527	1.7	3
119	Three-dimensional validation and analyses of the optimized CFETR HCCB blanket. *Fusion Engineering and Design*, **2020**, 161, 111971	1.7	2
118	PIV measurement and numerical investigation on flow characteristics of simulated fast reactor fuel subassembly. *Nuclear Engineering and Technology*, **2020**, 52, 897-907	2.6	1
117	Assessment of ECCMIX component in RELAP5 based on ECCS experiment. *Nuclear Engineering and Technology*, **2020**, 52, 59-68	2.6	4
116	RELAP5/MOD3.4 calculation and model evaluation based on upper plenum entrainment experiment in AP1000. *Annals of Nuclear Energy*, **2020**, 138, 107143	1.7	4
115	Thermoelectric characteristics analysis of thermionic space nuclear power reactor. *International Journal of Energy Research*, **2020**, 44, 855-868	4.5	6
114	Numerical study on the thermal stratification characteristics in the upper plenum of sodium-cooled fast reactor (SFR). *Annals of Nuclear Energy*, **2020**, 138, 107222	1.7	11
113	Parametric investigation of radiation heat transfer and evaporation characteristics of a liquid droplet radiator. *Aerospace Science and Technology*, **2020**, 106, 106214	4.9	3
112	Study of boron diffusion models and dilution accidents in nuclear reactor: A comprehensive review. *Annals of Nuclear Energy*, **2020**, 148, 107659	1.7	3
111	Numerical simulation of a small high-temperature heat pipe cooled reactor with CFD methodology. *Nuclear Engineering and Design*, **2020**, 370, 110907	1.8	3
110	Experimental research on fragmentation characteristics of molten stainless steel discharged into sodium pool and comparison with molten copper. *Progress in Nuclear Energy*, **2020**, 118, 103069	2.3	3

109	Experimental study on transient performance of heat pipe-cooled passive residual heat removal system of a molten salt reactor. *Progress in Nuclear Energy*, **2020**, 118, 103113	2.3	8
108	Hybrid medium model for conjugate heat transfer modeling in the core of sodium-cooled fast reactor. *Nuclear Engineering and Technology*, **2020**, 52, 708-720	2.6	4
107	Thermal-hydraulic analysis of the thermoelectric space reactor power system with a potassium heat pipe radiator. *Annals of Nuclear Energy*, **2020**, 136, 107018	1.7	5
106	Theoretical investigation of two-phase flow instability between parallel channels of natural circulation in rolling motion. *Nuclear Engineering and Design*, **2019**, 343, 257-268	1.8	7
105	Experimental study on spray characteristics of pressure-swirl nozzle in China advanced PWR containment. *Nuclear Engineering and Design*, **2019**, 350, 158-175	1.8	10
104	Transient analysis of tritium transport characteristics in fluoride-salt-cooled high-temperature reactor. *Progress in Nuclear Energy*, **2019**, 117, 103064	2.3	3
103	Development and application of SANPR with coupling numerical methods for analysis of effect of moving condition on natural circulation of nuclear power system. *Nuclear Engineering and Design*, **2019**, 349, 118-135	1.8	3
102	Numerical analysis of simulant effect on natural convection characteristics in corium pools. *Applied Thermal Engineering*, **2019**, 156, 730-740	5.8	4
101	Development of thermal hydraulic design code for SFR steam generators. *Nuclear Engineering and Design*, **2019**, 348, 46-55	1.8	5
100	Development and validation of boron diffusion model in nuclear reactor core subchannel analysis. *Annals of Nuclear Energy*, **2019**, 130, 208-217	1.7	9
99	Prediction of flow boiling heat transfer coefficient in horizontal channels varying from conventional to small-diameter scales by genetic neural network. *Nuclear Engineering and Technology*, **2019**, 51, 1897-1904	2.6	6
98	Flow and heat transfer characteristics in plate-type fuel channels after formation of blisters on fuel elements. *Annals of Nuclear Energy*, **2019**, 134, 284-298	1.7	13
97	Numerical investigation on thermal-hydraulic characteristics of NaK in a helical wire wrapped annulus. *International Journal of Heat and Mass Transfer*, **2019**, 145, 118689	4.9	12
96	Thermal-hydraulic analysis code development for sodium heated once-through steam generator. *Annals of Nuclear Energy*, **2019**, 127, 385-394	1.7	9
95	Effects of nitrogen and carbon monoxide on the detonation of hydrogen-air gaseous mixtures. *Nuclear Engineering and Design*, **2019**, 343, 1-10	1.8	4
94	Heat transfer characteristics in super-low finned-tube bundles of moisture separator reheaters. *Nuclear Engineering and Design*, **2019**, 341, 368-376	1.8	1
93	Numerical approach to study the thermal-hydraulic characteristics of Reactor Vessel Cooling system in sodium-cooled fast reactors. *Progress in Nuclear Energy*, **2019**, 110, 213-223	2.3	14
92	Numerical research on the coupling optimization design rule of the CFETR HCSB blanket using the NTCOC code. *Fusion Engineering and Design*, **2018**, 127, 234-248	1.7	7

91	Numerical study on the purge gas flow and heat transfer characteristics in helium cooled solid breeder blanket of CFETR. *Progress in Nuclear Energy*, **2018**, 105, 114-123	2.3	6
90	CFD investigation on thermal-hydraulic behaviors of a wire-wrapped fuel subassembly for sodium-cooled fast reactor. *Annals of Nuclear Energy*, **2018**, 113, 256-269	1.7	32
89	Review of conceptual design and fundamental research of molten salt reactors in China. *International Journal of Energy Research*, **2018**, 42, 1834-1848	4.5	41
88	Development of a subchannel analysis code for SFR wire-wrapped fuel assemblies. *Progress in Nuclear Energy*, **2018**, 104, 327-341	2.3	14
87	Experimental study on the heat transfer characteristics of fluoride salt in the new conceptual passive heat removal system of molten salt reactor. *International Journal of Energy Research*, **2018**, 42, 1635-1648	4.5	6
86	Thermal-hydraulic design and analysis of a small modular molten salt reactor (MSR) with solid fuel. *International Journal of Energy Research*, **2018**, 42, 1098-1114	4.5	4
85	Analysis of flow blockage accidents in rectangular fuel assembly based on CFD methodology. *Annals of Nuclear Energy*, **2018**, 112, 71-83	1.7	14
84	Improving the optimization algorithm of NTCOC for application in the HCSB blanket for CFETR Phase II. *Fusion Engineering and Design*, **2018**, 135, 216-227	1.7	8
83	Experimental research on the characteristics of steam-water counter-current flow in the Pressurizer Surge Line assembly. *Experimental Thermal and Fluid Science*, **2018**, 96, 180-191	3	5
82	Upgrade of FROBA code and its application in thermal-mechanical analysis of space reactor fuel. *Nuclear Engineering and Design*, **2018**, 332, 297-306	1.8	14
81	Experimental investigation of entrainment effect on the countercurrent flow in the Hot Leg and Pressurizer Surge Line assembly of third-generation passive nuclear reactors. *Nuclear Engineering and Design*, **2018**, 335, 326-338	1.8	2
80	Analysis code development for the direct reactor auxiliary cooling system of the pool-type sodium-cooled fast reactor. *Kerntechnik*, **2018**, 83, 232-236	0.4	2
79	Modification and application of Relap5 Mod3 code to several types of nonwater-cooled advanced nuclear reactors. *International Journal of Energy Research*, **2018**, 42, 221-235	4.5	2
78	Study of tritium transport characteristics in a transportable fluoride-salt-cooled high-temperature reactor. *International Journal of Energy Research*, **2018**, 42, 1536-1550	4.5	8
77	Conceptual design and comprehensive optimization analysis of a fusion-fission hybrid reactor water-cooled pressure tube blanket. *Progress in Nuclear Energy*, **2018**, 103, 8-19	2.3	3
76	CFD Investigation of Thermal-Hydraulic Behaviors in Full Reactor Core for Sodium-Cooled Fast Reactor **2018**,		2
75	Experimental investigation of flow and convective heat transfer on a high-Prandtl-number fluid through the nuclear reactor pebble bed core. *Applied Thermal Engineering*, **2018**, 145, 48-57	5.8	13
74	Experimental study on heat transfer performance between fluoride salt and heat pipes in the new conceptual passive residual heat removal system of molten salt reactor. *Nuclear Engineering and Design*, **2018**, 339, 215-224	1.8	6

73	Development of a multi-compartment containment code for advanced PWR plant. *Nuclear Engineering and Design*, **2018**, 334, 75-89	1.8	6
72	The development and validation of the inter-wrapper flow model in sodium-cooled fast reactors. *Progress in Nuclear Energy*, **2018**, 108, 54-65	2.3	14
71	Numerical study of tritium transport characteristics in Thorium Molten Salt Reactor with Solid Fuel (TMSR-SF). *Nuclear Engineering and Design*, **2018**, 335, 391-399	1.8	6
70	Validation of a methodology for thermal stratification analysis in sodium-cooled fast reactors. *International Journal of Energy Research*, **2018**, 42, 3803-3822	4.5	11
69	RELAP5 MOD3.2 modification and application to the transient analysis of a fluoride-salt-cooled high-temperature reactor. *Annals of Nuclear Energy*, **2017**, 101, 504-515	1.7	7
68	Performance analysis of heat pipe radiator unit for space nuclear power reactor. *Annals of Nuclear Energy*, **2017**, 103, 74-84	1.7	34
67	Development of a new Pellet-Clad Mechanical Interaction (PCMI) model and its application in ATFs. *Annals of Nuclear Energy*, **2017**, 104, 146-156	1.7	29
66	Conceptual design and analysis of heat pipe cooled silo cooling system for the transportable fluoride-salt-cooled high-temperature reactor. *Annals of Nuclear Energy*, **2017**, 109, 458-468	1.7	10
65	Numerical research on water hammer phenomenon of parallel pump-valve system by coupling FLUENT with RELAP5. *Annals of Nuclear Energy*, **2017**, 109, 318-326	1.7	16
64	Preliminary accident analysis of Loss of Off-Site Power and In-Box LOCA for the CFETR helium cooled solid breeder blanket. *Fusion Engineering and Design*, **2017**, 118, 142-150	1.7	15
63	Transient Safety Analysis of a Transportable Fluoride-Salt-Cooled High-Temperature Reactor Using RELAP5-3D. *Nuclear Technology*, **2017**, 198, 1-16	1.4	3
62	Numerical research on the neutronic/thermal-hydraulic/mechanical coupling characteristics of the optimized helium cooled solid breeder blanket for CFETR. *Fusion Engineering and Design*, **2017**, 114, 141-156	1.7	24
61	Uncertainty analysis of Transportable Fluoride-salt-cooled High-temperature Reactor (TFHR) using coupled DAKOTA with RELAP-3D method. *Nuclear Engineering and Design*, **2017**, 324, 269-279	1.8	7
60	Neutronics/Thermal-hydraulics Coupling Analysis for the Liquid-Fuel MOSART Concept. *Energy Procedia*, **2017**, 127, 343-351	2.3	4
59	Development of a neutronics/thermal-hydraulic coupling optimization code and its application on the CFETR HCSB blanket. *Fusion Engineering and Design*, **2017**, 122, 140-153	1.7	11
58	Evaluation and optimization of tritium breeding, shielding and nuclear heating performances of the helium cooled solid breeder blanket for CFETR. *International Journal of Hydrogen Energy*, **2017**, 42, 24263-24277	6.7	23
57	Development and application of a neutronics/thermal-hydraulics coupling optimization code for the CFETR helium cooled solid breeder blanket with mixed pebble beds. *Fusion Engineering and Design*, **2017**, 125, 24-37	1.7	9
56	Thermal-mechanical coupling behavior analysis on metal-matrix dispersed plate-type fuel. *Progress in Nuclear Energy*, **2017**, 95, 8-22	2.3	14

55	Experimental Investigation of Air-Water CCFL in the Pressurizer Surge Line of AP1000. *Nuclear Technology*, **2016**, 196, 614-640	1.4	6
54	Performance of radial fuel shuffling sodium cooled Breed and Burn reactor core. *Annals of Nuclear Energy*, **2016**, 96, 363-376	1.7	2
53	Thermal-hydraulic analysis of a fluoride-salt-cooled pebble-bed reactor with CFD methodology. *Progress in Nuclear Energy*, **2016**, 91, 83-96	2.3	8
52	Preliminary neutronic and thermal-hydraulic analysis of a 2MW Thorium-based Molten Salt Reactor with Solid Fuel. *Progress in Nuclear Energy*, **2016**, 86, 1-10	2.3	20
51	Mechanism study and theoretical simulation on heat split phenomenon in dual-cooled annular fuel element. *Annals of Nuclear Energy*, **2016**, 94, 44-54	1.7	17
50	Transient thermalhydraulic analysis of a space thermionic reactor. *Annals of Nuclear Energy*, **2016**, 89, 38-49	1.7	15
49	Thermo-Hydraulic Analysis of the Optimized Helium Cooled Solid Breeder Blanket for CFETR **2016**,		1
48	Development of a thermalmechanical behavior coupling analysis code for a dual-cooled annular fuel element in PWRs. *Nuclear Engineering and Design*, **2016**, 301, 353-365	1.8	22
47	Thermal-hydraulic analysis of the improved TOPAZ-II power system using a heat pipe radiator. *Nuclear Engineering and Design*, **2016**, 307, 218-233	1.8	17
46	Coupled neutronics/thermal-hydraulics and safety characteristics of liquid-fueled Molten Salt Reactors. *Kerntechnik*, **2016**, 81, 149-159	0.4	3
45	Minor actinide transmutation in a board type sodium cooled breed and burn reactor core. *Annals of Nuclear Energy*, **2015**, 81, 41-49	1.7	10
44	Thermal hydraulic investigations and optimization on the EVC system of a PWR by CFD simulation. *Nuclear Engineering and Design*, **2015**, 289, 19-34	1.8	1
43	Steady and transient solutions of neutronics problems based on finite volume method (FVM) with a CFD code. *Progress in Nuclear Energy*, **2015**, 85, 366-374	2.3	19
42	Experimental research on the thermal hydraulic characteristics of sodium boiling in an annulus. *Experimental Thermal and Fluid Science*, **2015**, 60, 263-274	3	9
41	Transient analyses for a molten salt fast reactor with optimized core geometry. *Nuclear Engineering and Design*, **2015**, 292, 164-176	1.8	6
40	Three dimensional neutronic/thermal-hydraulic coupled simulation of MSR in transient state condition. *Nuclear Engineering and Design*, **2015**, 282, 93-105	1.8	7
39	Development of a kinetic model for safety studies of liquid-fuel reactors. *Progress in Nuclear Energy* , **2015**, 81, 104-112	2.3	10
38	Preliminary design study of a board type radial fuel shuffling sodium cooled breed and burn reactor core. *Nuclear Engineering and Design*, **2014**, 278, 679-685	1.8	4

37	Numerical studies of stepwise radial fuel shuffling in a traveling wave reactor. *Science China Technological Sciences*, **2014**, 57, 1229-1237	3.5	2
36	Accident analyses for china pressurizer reactor with an innovative conceptual design of passive residual heat removal system. *Nuclear Engineering and Design*, **2014**, 272, 45-52	1.8	17
35	Three dimensional neutronic/thermal-hydraulic coupled simulation of MSR in steady state condition. *Nuclear Engineering and Design*, **2014**, 267, 88-99	1.8	3
34	Depressurization study of supercritical fluid blowdown from simple vessel. *Annals of Nuclear Energy* , **2014**, 66, 94-103	1.7	14
33	Computational Fluid Dynamics Analysis of a Fluoride Salt‐Cooled Pebble-Bed Test Reactor. *Nuclear Science and Engineering*, **2014**, 178, 86-102	1.2	11
32	COUPLE, A Time-Dependent Coupled Neutronics and Thermal-Hydraulics Code, and its Application to MSFR **2014**,		3
31	Development of a Thermal-Hydraulic Analysis Code and Transient Analysis for a FHTR **2014**,		3
30	Preliminary study of coupling CFD code FLUENT and system code RELAP5. *Annals of Nuclear Energy*, **2014**, 73, 96-107	1.7	29
29	Development of a MCNP‐ORIGEN burn-up calculation code system and its accuracy assessment. *Annals of Nuclear Energy*, **2014**, 63, 491-498	1.7	16
28	Study of traveling wave reactor (TWR) and CANDLE strategy: A review work. *Progress in Nuclear Energy*, **2014**, 71, 195-205	2.3	8
27	Transient behavior of the sodium‐potassium alloy heat pipe in passive residual heat removal system of molten salt reactor. *Progress in Nuclear Energy*, **2013**, 68, 142-152	2.3	39
26	Theoretical and numerical studies of TWR based on ESFR core design. *Energy Conversion and Management*, **2013**, 72, 12-18	10.6	6
25	Coupled neutronics/thermal-hydraulics for analysis of molten salt reactor. *Nuclear Engineering and Design*, **2013**, 258, 144-156	1.8	20
24	Study on the characteristics of the sodium heat pipe in passive residual heat removal system of molten salt reactor. *Nuclear Engineering and Design*, **2013**, 265, 691-700	1.8	54
23	Simulations of unprotected loss of heat sink and combination of events accidents for a molten salt reactor. *Annals of Nuclear Energy*, **2013**, 53, 309-319	1.7	17
22	The effects of core zoning on optimization of design analysis of molten salt reactor. *Nuclear Engineering and Design*, **2013**, 265, 967-977	1.8	4
21	Experimental research on the incipient boiling wall superheat of sodium. *Progress in Nuclear Energy*, **2013**, 68, 121-129	2.3	9
20	Development of a thermal hydraulic code for an integral reactor. *Progress in Nuclear Energy*, **2013**, 68, 31-42	2.3	4

19	Fundamental solution of nuclear solitary wave. *Energy Conversion and Management*, **2012**, 59, 40-49	10.6	15
18	Numerical analysis for a molten salt reactor in the presence of localized perturbations. *Progress in Nuclear Energy*, **2012**, 60, 61-72	2.3	4
17	Safety-Related Optimization and Analyses of an Innovative Fast Reactor Concept. *Sustainability*, **2012**, 4, 1274-1291	3.6	5
16	Theoretical Modeling of Radial Standing Wave Reactor. *Fusion Science and Technology*, **2012**, 61, 275-280	1.1	3
15	Numerical Studies of Axial Fuel Shuffling. *Fusion Science and Technology*, **2012**, 61, 287-292	1.1	
14	Prevention and mitigation of severe accident developments and recriticalities in advanced fast reactor systems. *Progress in Nuclear Energy*, **2011**, 53, 835-841	2.3	12
13	Numerical studies of nuclear traveling waves in a supercritical water cooled fast reactor. *Progress in Nuclear Energy*, **2011**, 53, 806-813	2.3	9
12	Study on the Coupled Neutronic and Thermal-Hydraulic Characteristics of the New Concept Molten Salt Reactor. *Journal of Engineering for Gas Turbines and Power*, **2010**, 132,	1.7	1
11	Solitary breeding/burning waves in a supercritical water cooled fast reactor. *Energy Conversion and Management*, **2010**, 51, 1792-1798	10.6	10
10	Numerical research on natural convection in molten salt reactor with non-uniformly distributed volumetric heat generation. *Nuclear Engineering and Design*, **2010**, 240, 796-806	1.8	8
9	Development of a safety analysis code for molten salt reactors. *Nuclear Engineering and Design*, **2009**, 239, 2778-2785	1.8	34
8	Analysis on the neutron kinetics for a molten salt reactor. *Progress in Nuclear Energy*, **2009**, 51, 624-636	2.3	34
7	Development of a steady state analysis code for a molten salt reactor. *Annals of Nuclear Energy*, **2009**, 36, 590-603	1.7	32
6	Natural Convection With Non-Uniform Heat Generation in Molten Salt Reactor **2009**,		1
5	Steady thermal hydraulic analysis for a molten salt reactor. *Nuclear Science and Techniques/Hewuli*, **2008**, 19, 187-192	2.1	10
4	Evaluation of Static Thermophysical Properties of the Ternary Molten Salt System Li, Na and Be/F Based on the Modified Peng-Robinson Equation. *Journal of Power and Energy Systems*, **2008**, 2, 826-833		4
3	Numerical Research on Steady Coupling of Neutronics and Thermal-Hydraulics for a Molten Salt Reactor **2008**,		1
2	Estimation of thermodynamic properties of the ternary molten salt system, LiF-NaF-BeF2, by the modified Peng-Robinson equation. *Frontiers of Energy and Power Engineering in China*, **2007**, 1, 174-180		4

1	Dynamic simulation of a space gas-cooled reactor power system with a closed Brayton cycle. *Frontiers in Energy*,1	2.6	0