Mingjun Wang

List of Publications by Year
in descending order

108	1,903	304602 22	345118 36
papers	citations	h-index	g-index

109	109	109	580
all docs	docs citations	times ranked	citing authors

#	Article	IF	Citations
1	Recent progress of CFD applications in PWR thermal hydraulics study and future directions. Annals of Nuclear Energy, 2021, 150, 107836.	0.9	137
2	Two-phase bubbly flow simulation using CFD method: A review of models for interfacial forces. Progress in Nuclear Energy, 2020, 125, 103360.	1.3	94
3	Review of conceptual design and fundamental research of molten salt reactors in China. International Journal of Energy Research, 2018, 42, 1834-1848.	2.2	79
4	A review on thermohydraulic and mechanical-physical properties of SiC, FeCrAl and Ti3SiC2 for ATF cladding. Nuclear Engineering and Technology, 2020, 52, 1-13.	1.1	67
5	Research on the designed emergency passive residual heat removal system during the station blackout scenario for CPR1000. Annals of Nuclear Energy, 2012, 45, 86-93.	0.9	60
6	Numerical research on thermal mixing characteristics in a 45-degree T-junction for two-phase stratified flow during the emergency core cooling safety injection. Progress in Nuclear Energy, 2019, 114, 91-104.	1.3	56
7	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.	0.9	53
8	Thermal hydraulic and stress coupling analysis for AP1000 Pressurized Thermal Shock (PTS) study under SBLOCA scenario. Applied Thermal Engineering, 2017, 122, 158-170.	3.0	52
9	Study on the coolant mixing phenomenon in a 45Â° T junction based on the thermal-mechanical coupling method. Applied Thermal Engineering, 2018, 144, 600-613.	3.0	43
10	Numerical study on the single bubble rising behaviors under rolling conditions. Nuclear Engineering and Design, 2019, 349, 183-192.	0.8	43
11	Severe accident analysis for a typical PWR using the MELCOR code. Progress in Nuclear Energy, 2014, 71, 30-38.	1.3	40
12	Development of a new Pellet-Clad Mechanical Interaction (PCMI) model and its application in ATFs. Annals of Nuclear Energy, 2017, 104, 146-156.	0.9	38
13	Three-dimensional study on the hydraulic characteristics under the steam generator (SG) tube plugging operations for AP1000. Progress in Nuclear Energy, 2019, 112, 63-74.	1.3	33
14	An evaluation of designed passive Core Makeup Tank (CMT) for China pressurized reactor (CPR1000). Annals of Nuclear Energy, 2013, 56, 81-86.	0.9	31
15	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.	0.9	30
16	Numerical study on the turbulent mixing in channel with Large Eddy Simulation (LES) using spectral element method. Nuclear Engineering and Design, 2019, 348, 169-176.	0.8	28
17	Numerical investigation of flow and heat transfer characteristics in plate-type fuel channels of IAEA MTR based on OpenFOAM. Progress in Nuclear Energy, 2021, 141, 103963.	1.3	28
18	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.	1.0	26

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19	Experimental study of liquid sodium flow and heat transfer characteristics along a hexagonal 7-rod bundle. Applied Thermal Engineering, 2019, 149, 578-587.	3.0	25
20	Preliminary design of the I2S-LWR containment system. Annals of Nuclear Energy, 2020, 145, 106065.	0.9	25
21	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.	0.8	24
22	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.	1.3	24
23	Numerical research on water hammer phenomenon of parallel pump-valve system by coupling FLUENT with RELAP5. Annals of Nuclear Energy, 2017, 109, 318-326.	0.9	23
24	The comparison of designed water-cooled and air-cooled passive residual heat removal system for 300MW nuclear power plant during the feed-water line break scenario. Annals of Nuclear Energy, 2013, 57, 164-172.	0.9	22
25	Research on the leak-rate characteristics of leak-before-break (LBB) in pressurized water reactor (PWR). Applied Thermal Engineering, 2014, 62, 133-140.	3.0	22
26	Development and validation of boron diffusion model in nuclear reactor core subchannel analysis. Annals of Nuclear Energy, 2019, 130, 208-217.	0.9	22
27	Passive decay heat removal system design for the integral inherent safety light water reactor (I2S-LWR). Annals of Nuclear Energy, 2020, 145, 106987.	0.9	22
28	The development of high fidelity Steam Generator three dimensional thermal hydraulic coupling code: STAF-CT. Nuclear Engineering and Technology, 2021, 53, 763-775.	1.1	22
29	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.	1.1	20
30	Numerical study on the thermal stratification characteristics in the upper plenum of sodium-cooled fast reactor (SFR). Annals of Nuclear Energy, 2020, 138, 107222.	0.9	20
31	CFD simulation on the flow characteristics in the PWR lower plenum with different internal structures. Nuclear Engineering and Design, 2020, 364, 110705.	0.8	20
32	LES and URANS study on turbulent flow through 3ÂÃ—Â3 rod bundle with spacer grid and mixing vanes using spectral element method. Annals of Nuclear Energy, 2021, 161, 108474.	0.9	19
33	Development of an OpenFOAM solver for numerical simulations of shell-and-tube heat exchangers based on porous media model. Applied Thermal Engineering, 2022, 210, 118389.	3.0	19
34	Migrationâ€“deposition coupling characteristics and influence of corrosion products on heat transfer in steam generators. Applied Thermal Engineering, 2022, 211, 118507.	3.0	19
35	Development of subcooled wall boiling model considering bubble sliding in narrow rectangular channel. International Journal of Thermal Sciences, 2022, 181, 107787.	2.6	19
36	Experimental research on the incipient boiling wall superheat ofÂsodium. Progress in Nuclear Energy, 2013, 68, 121-129.	1.3	18

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37	Preliminary study of parameter uncertainty influence on Pressurized Water Reactor core design. Progress in Nuclear Energy, 2013, 68, 200-209.	1.3	17
38	Depressurization study of supercritical fluid blowdown from simple vessel. Annals of Nuclear Energy, 2014, 66, 94-103.	0.9	17
39	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.	1.0	17
40	Study on the hydrogen risk in venturi scrubber filter of filtered containment venting system under PWR severe accident. Nuclear Engineering and Design, 2018, 327, 61-69.	0.8	17
41	Development of a multi-compartment containment code for advanced PWR plant. Nuclear Engineering and Design, 2018, 334, 75-89.	0.8	17
42	Thermal Hydraulic and Neutronics Coupling Analysis for Plate Type Fuel in Nuclear Reactor Core. Science and Technology of Nuclear Installations, 2020, 2020, 1-12.	0.3	16
43	Validation of a methodology for thermal stratification analysis in sodium-cooled fast reactors. International Journal of Energy Research, 2018, 42, 3803-3822.	2.2	15
44	Numerical study on the thermal stratification characteristics of AP1000 pressurizer surge line. Annals of Nuclear Energy, 2019, 130, 8-19.	0.9	15
45	Full-scale numerical study on the flow characteristics and mal-distribution phenomena in SG steam-water separation system of an advanced PWR. Progress in Nuclear Energy, 2020, 118, 103075.	1.3	15
46	Quantification of the effect of Cr-coated-Zircaloy cladding during a short term station black out. Nuclear Engineering and Design, 2020, 363, 110678.	0.8	15
47	Review on heat transfer and ï¬‚ow characteristics of liquid sodium (1): Single-phase. Progress in Nuclear Energy, 2018, 104, 306-316.	1.3	14
48	Local effect model development for the steam generator three dimensional thermal hydraulics analysis code. Annals of Nuclear Energy, 2020, 136, 107020.	0.9	13
49	CFD modeling of liquid entrainment through vertical T-junction of fourth stage automatic depressurization system (ADS-4). Annals of Nuclear Energy, 2021, 159, 108317.	0.9	13
50	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.	0.9	12
51	Review on heat transfer and flow characteristics of liquid sodium (2): Two-phase. Progress in Nuclear Energy, 2018, 103, 151-164.	1.3	12
52	Improving the optimization algorithm of NTCOC for application in the HCSB blanket for CFETR Phase II. Fusion Engineering and Design, 2018, 135, 216-227.	1.0	12
53	Comparative analysis of auxiliary feedwater system and passive safety system under typical accident scenarios for integrated pressurized water reactor (IPWR). Progress in Nuclear Energy, 2019, 115, 42-51.	1.3	12
54	Full-scale numerical study on the thermal-hydraulic characteristics of steam-water separation system in an advanced PWR UTSG. Part two: Droplets separation process. Progress in Nuclear Energy, 2020, 118, 103139.	1.3	12

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55	Large eddy simulation on the turbulent mixing phenomena in 3Ã—3 bare tight lattice rod bundle using spectral element method. Nuclear Engineering and Technology, 2020, 52, 1945-1954.	1.1	12
56	Development of thermal hydraulic design code for SFR steam generators. Nuclear Engineering and Design, 2019, 348, 46-55.	0.8	11
57	Study of boron diffusion models and dilution accidents in nuclear reactor: A comprehensive review. Annals of Nuclear Energy, 2020, 148, 107659.	0.9	11
58	Numerical simulation of temperature heterogeneity inside the AP1000 upper plenum and hot leg. Nuclear Engineering and Design, 2020, 362, 110525.	0.8	11
59	Experimental study on vertically upward steam-water two-phase flow patterns in narrow rectangular channel. Nuclear Engineering and Technology, 2021, 53, 61-68.	1.1	11
60	CFD study on onset of liquid entrainment through ADS-4 branch line in AP1000. Nuclear Engineering and Design, 2021, 380, 111299.	0.8	11
61	Study on the reactor core barrel instantaneous characteristics in case of Loss of Coolant Accident (LOCA) scenarios for loop-type PWR. Nuclear Engineering and Design, 2017, 324, 93-102.	0.8	10
62	The primary reactor coolant system concept of the integral, inherently-safe light water reactor. Annals of Nuclear Energy, 2017, 100, 53-67.	0.9	10
63	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.	1.3	10
64	Thermal fatigue analysis of structures subjected to liquid metal jets at different temperatures in the Gen-IV nuclear energy system. Energy, 2022, 256, 124681.	4.5	10
65	Ammonia-water mixture property code (AWProC) development, verification and Kalina cycle design for nuclear power plant. Progress in Nuclear Energy, 2016, 91, 26-37.	1.3	9
66	Thermal hydraulic characteristics analysis of direct residual heat removal system for China pressurized reactor. Progress in Nuclear Energy, 2016, 93, 231-237.	1.3	9
67	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.	1.0	9
68	Experimental study on flow characteristics of rectangular narrow channel. International Journal of Advanced Nuclear Reactor Design and Technology, 2020, 2, 60-68.	0.5	9
69	Analysis on the behavior of dispersed plate-type fuel based on fluid-solid coupling method. Progress in Nuclear Energy, 2020, 126, 103398.	1.3	9
70	CFD simulation on the transient process of coolant mixing phenomenon in reactor pressure vessel. Annals of Nuclear Energy, 2021, 153, 108045.	0.9	9
71	A modified system analysis code for thermo-hydraulic calculation of hydrogen in a nuclear thermal propulsion (NTP) system. Annals of Nuclear Energy, 2021, 164, 108632.	0.9	9
72	Experimental study of the helium flow characteristics in pebble-bed under the condition of CFETR's blanket module. Progress in Nuclear Energy, 2017, 100, 283-291.	1.3	8

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73	Multiscale Thermal Hydraulic Study under the Inadvertent Safety Injection System Operation Scenario of Typical Pressurized Water Reactor. Science and Technology of Nuclear Installations, 2017, 2017, 1-15.	0.3	8
74	Numerical study on thermal deformation behaviors of the single subassembly in sodium-cooled fast reactors based on Euler-Bernoulli beam theory. Nuclear Engineering and Design, 2019, 345, 28-39.	0.8	8
75	Hybrid medium model for conjugate heat transfer modeling in the core of sodium-cooled fast reactor. Nuclear Engineering and Technology, 2020, 52, 708-720.	1.1	8
76	LES study on the turbulent thermal stratification and thermo-mechanical fatigue analysis for NPP surge line. International Journal of Thermal Sciences, 2022, 178, 107608.	2.6	8
77	Numerical simulation on the thermal stratification in the lead pool of lead-cooled fast reactor (LFR). Annals of Nuclear Energy, 2022, 174, 109176.	0.9	8
78	Development and applicability analyses of ADS-4 entrainment model in large advanced PWR. Nuclear Engineering and Design, 2020, 356, 110379.	0.8	7
79	A new method for improving the tritium breeding and releasing performance of China Fusion Engineering Test Reactor phase II heliumâ€cooled ceramic breeder blanket. International Journal of Energy Research, 2020, 44, 5977-5989.	2.2	7
80	Development and application of TaSNAM 2.0 for advanced pressurized water reactor. Annals of Nuclear Energy, 2022, 166, 108801.	0.9	7
81	Numerical Study of Integral Inherently Safe Light Water Reactor in Case of Inadvertent DHR Operation Based on the Multiscale Method. Nuclear Technology, 2018, 203, 194-204.	0.7	6
82	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.	1.5	6
83	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.	0.8	6
84	A simulation of I2S-LWR selected transients. Annals of Nuclear Energy, 2020, 145, 105421.	0.9	5
85	Assessment of ECCMIX component in RELAP5 based on ECCS experiment. Nuclear Engineering and Technology, 2020, 52, 59-68.	1.1	5
86	Spatial temperature distribution of fuel assembly pre-simulation for a new simple core degradation experiment. Progress in Nuclear Energy, 2019, 111, 174-182.	1.3	4
87	Numerical study on the thermal-hydraulic characteristics of the choked flow through micro cracks for various conditions. Nuclear Engineering and Design, 2020, 358, 110381.	0.8	4
88	Preliminary design and analyses of the helium cooled ceramic breeder blanket for CFETR phase II. International Journal of Energy Research, 2021, 45, 11598-11615.	2.2	4
89	CFD Simulation of thermal hydraulic characteristics in a typical upper plenum of RPV. Frontiers in Energy, 0, , 1.	1.2	4
90	Numerical study on improved design of passive residual heat removal system for China pressurizer reactor. Nuclear Engineering and Design, 2021, 375, 111087.	0.8	4

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91	CFD/RELAP5 coupling analysis of the ISP No. 43 boron dilution experiment. Nuclear Engineering and Technology, 2022, 54, 97-109.	1.1	4
92	The fouling and thermal hydraulic coupling study on the typical 5 Ã— 5 rod bundle in PWRs. Progress in Nuclear Energy, 2022, 148, 104221.	1.3	4
93	Numerical study on the enhanced heat transfer characteristics of steam generator with axial economizer. International Journal of Thermal Sciences, 2022, 182, 107794.	2.6	4
94	CFD simulation of thermal hydraulic phenomena in enclosed cavity of nuclear power plants. Annals of Nuclear Energy, 2021, 151, 107953.	0.9	3
95	Analysis of the natural circulation test of PHENIX reactor by the THACS code. Annals of Nuclear Energy, 2021, 152, 108022.	0.9	3
96	CFD Investigation of Thermal-Hydraulic Behaviors in Full Reactor Core for Sodium-Cooled Fast Reactor. , 2018, , .		2
97	Study on the water seal formation process in advanced PWR pressurizer using CFD method. Annals of Nuclear Energy, 2020, 135, 106949.	0.9	2
98	Theoretical study of steady-state characteristics of natural circulation system. Annals of Nuclear Energy, 2020, 147, 107663.	0.9	2
99	Boiling incipience of liquid sodium flow in an annular channel based on wavelet analysis. Progress in Nuclear Energy, 2020, 121, 103258.	1.3	2
100	An experimentâ€based validation of a system code for prediction of passive natural circulation in sodiumâ€cooled fast reactor. International Journal of Energy Research, 2021, 45, 12093-12109.	2.2	2
101	Three-dimensional validation and analyses of the optimized CFETR HCCB blanket. Fusion Engineering and Design, 2020, 161, 111971.	1.0	2
102	Code development and characteristics analysis for Leak Before Break in the pipelines of Sodium-cooled Fast Reactor. Annals of Nuclear Energy, 2019, 133, 777-794.	0.9	1
103	Experimental and theoretical study on fluid-structure interaction and non-equilibrium of the flow through micro cracks. Annals of Nuclear Energy, 2020, 142, 107441.	0.9	1
104	The Severe Accident Analysis of CPR1000 Based on MELCOR Code. , 2013, , .		0
105	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.	1.0	0
106	Development and Basic Verification of Decay Heat Removal Analysis Code of Sodium-Cooled Fast Reactor. , 2018, , .		0
107	Numerical Research on Fuel Rod Progression During Core Degradation Process Using MELCOR. , 2018, , .		0
108	Experimental Investigation and Model Analysis on Upper Plenum Entrainment in AP1000. , 2018, , .		0