Wen-Xi Tian

List of Publications by Year
in descending order

306 papers	4,598 citations	145106 33 h-index	242451 47 g-index
309 all docs	309 docs citations	309 times ranked	1647 citing authors

#	Article	IF	Citations
1	CFD/RELAP5 coupling analysis of the ISP No. 43 boron dilution experiment. Nuclear Engineering and Technology, 2022, 54, 97-109.	1.1	4
2	Assessment of the severe accident code MIDAC based on FROMA, QUENCH-06&16 experiments. Nuclear Engineering and Technology, 2022, 54, 579-588.	1.1	12
3	Experiment study on thermal behavior of a horizontal high-temperature heat pipe under motion conditions. Annals of Nuclear Energy, 2022, 165, 108760.	0.9	15
4	Development and application of TaSNAM 2.0 for advanced pressurized water reactor. Annals of Nuclear Energy, 2022, 166, 108801.	0.9	7
5	CFD investigation of the cold wall effect on CHF in a 5ÂÃ—Â5 rod bundle for PWRs. Nuclear Engineering and Design, 2022, 387, 111589.	0.8	20
6	Development of oxidation model for zirconium alloy cladding and application in the analysis of cladding behavior under loss of coolant accident. Journal of Nuclear Materials, 2022, 561, 153564.	1.3	5
7	Molecular dynamics study of liquid sodium film evaporation and condensation by Lennard-Jones potential. Nuclear Engineering and Technology, 2022, 54, 3117-3129.	1.1	4
8	Molecular dynamics simulation of the evaporation of liquid sodium film in the presence of non-condensable gas. Annals of Nuclear Energy, 2022, 170, 109005.	0.9	6
9	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
10	Migrationâ€“deposition coupling characteristics and influence of corrosion products on heat transfer in steam generators. Applied Thermal Engineering, 2022, 211, 118507.	3.0	19
11	SEINA: A two-dimensional steam explosion integrated analysis code. Nuclear Engineering and Technology, 2022, , .	1.1	2
12	Preliminary development of a multi-physics coupled fuel performance code for annular fuel analysis under normal conditions. Nuclear Engineering and Design, 2022, 393, 111810.	0.8	2
13	Transient behavior and maximum heat flux ratios of two-layer corium pool. International Journal of Thermal Sciences, 2022, 179, 107684.	2.6	3
14	Water film covering characteristic on horizontal fuel rod under impinging cooling condition. Nuclear Engineering and Technology, 2022, , .	1.1	0
15	Development of subcooled wall boiling model considering bubble sliding in narrow rectangular channel. International Journal of Thermal Sciences, 2022, 181, 107787.	2.6	19
16	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
17	Thermalâ€hydraulic analysis of an openâ€grid megawatt gasâ€cooled space nuclear reactor core. International Journal of Energy Research, 2021, 45, 11616-11628.	2.2	21
18	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

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19	Numerical analysis on flow instability of parallel channels in steam generator for sodiumâ€cooled fast reactor. International Journal of Energy Research, 2021, 45, 11943-11956.	2.2	7
20	Recent progress of CFD applications in PWR thermal hydraulics study and future directions. Annals of Nuclear Energy, 2021, 150, 107836.	0.9	137
21	Improvement and validation of a subâ€channel analysis code for a leadâ€cooled reactor with wire spacers. International Journal of Energy Research, 2021, 45, 12029-12046.	2.2	8
22	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
23	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
24	Thermalâ€hydraulic analysis of gasâ€cooled space nuclear reactor power system with closed Brayton cycle. International Journal of Energy Research, 2021, 45, 11851-11867.	2.2	18
25	Transient thermalâ€hydraulic analysis of heat pipe cooled passive residual heat removal system of molten salt reactor. International Journal of Energy Research, 2021, 45, 1599-1612.	2.2	3
26	Numerical investigation on heat transfer characteristics of <scp>heliumâ€xenon</scp> gas mixture. International Journal of Energy Research, 2021, 45, 11745-11758.	2.2	9
27	A practical methodology devoted to poolâ€type phenomena simulation in safety analysis for sodiumâ€cooled fast reactor. International Journal of Energy Research, 2021, 45, 11868-11881.	2.2	2
28	Development and validation of transient <scp>thermalâ€hydraulic</scp> evaluation code for a leadâ€based fast reactor. International Journal of Energy Research, 2021, 45, 12215-12233.	2.2	2
29	Development a methodology for evaluating interâ€assembly heat transfer effect through reactor core in system safety analysis of sodiumâ€cooled fast reactor. International Journal of Energy Research, 2021, 45, 12258-12271.	2.2	5
30	Investigations of near-wall bubble behavior in wire heaters pool boiling. Thermal Science, 2021, 25, 3957-3967.	0.5	1
31	Development of thermal hydraulic analysis code of annular fuel under flow blockage condition. Annals of Nuclear Energy, 2021, 151, 107962.	0.9	5
32	Thermoelectric performance study on a heat pipe thermoelectric generator for micro nuclear reactor application. International Journal of Energy Research, 2021, 45, 12301-12316.	2.2	11
33	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
34	Development of a two-fluid based thermal-hydraulic subchannel analysis code with high-resolution numerical method. Progress in Nuclear Energy, 2021, 134, 103671.	1.3	7
35	Thermal-hydraulic analysis of a leadâ€“bismuth small modular reactor under moving conditions. Annals of Nuclear Energy, 2021, 154, 108116.	0.9	4
36	Investigation of single bubble behavior under rolling motions using multiphase MPS method on GPU. Nuclear Engineering and Technology, 2021, 53, 1810-1820.	1.1	2

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37	Analysis for Fuel Rod Performance under LOCA Based on the FROBA-ROD 2.0 Code. International Journal of Advanced Nuclear Reactor Design and Technology, 2021, , .	0.5	3
38	Numerical analysis of melt migration and solidification behavior in LBR severe accident with MPS method. Nuclear Engineering and Technology, 2021, 54, 162-162.	1.1	7
39	Dynamic simulation of a space gas-cooled reactor power system with a closed Brayton cycle. Frontiers in Energy, 2021, 15, 916-929.	1.2	5
40	Study on flow instability in natural circulation loop with parallel channels under moving condition. Nuclear Engineering and Design, 2021, 379, 111246.	0.8	4
41	CFD study on onset of liquid entrainment through ADS-4 branch line in AP1000. Nuclear Engineering and Design, 2021, 380, 111299.	0.8	11
42	Numerical investigation of oxidation and dissolution behavior in the fuel cladding using MPS-CV method. Nuclear Engineering and Design, 2021, 379, 111252.	0.8	2
43	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
44	Numerical simulation of micro-crack leakage on steam generator heat transfer tube. Nuclear Engineering and Design, 2021, 382, 111385.	0.8	7
45	Performance analysis of PRHRS in primary and secondary circuit for offshore floating nuclear plant. Annals of Nuclear Energy, 2021, 164, 108580.	0.9	5
46	Preliminary analysis on the thermal-mechanical behavior of dispersed plate-type fuel under reactivity insertion accident. Annals of Nuclear Energy, 2021, 163, 108509.	0.9	10
47	Thermal-hydraulic performance evaluation of annular fuel based on modified FROBA-ANNULAR. International Journal of Advanced Nuclear Reactor Design and Technology, 2021, 3, 108-118.	0.5	4
48	Low temperature overpressurization analysis for CPR1000. International Journal of Advanced Nuclear Reactor Design and Technology, 2021, 3, 145-153.	0.5	0
49	<scp>Numerical</scp>simulation on thermalâ€hydraulic and thermoelectric characteristics of the<scp>TOPAZâ€I</scp>reactor core. International Journal of Energy Research, 2021, 45, 12159-12172.	2.2	4
50	Preliminary design of the I2S-LWR containment system. Annals of Nuclear Energy, 2020, 145, 106065.	0.9	25
51	Numerical simulation of zircaloy-water reaction based on the moving particle semi-implicit method and combined analysis with the MIDAC code for the nuclear-reactor core melting process. Progress in Nuclear Energy, 2020, 118, 103083.	1.3	8
52	Current achievements on bubble dynamics analysis using MPS method. Progress in Nuclear Energy, 2020, 118, 103057.	1.3	21
53	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.	1.3	11
54	Local effect model development for the steam generator three dimensional thermal hydraulics analysis code. Annals of Nuclear Energy, 2020, 136, 107020.	0.9	13

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55	Development and applicability analyses of ADS-4 entrainment model in large advanced PWR. Nuclear Engineering and Design, 2020, 356, 110379.	0.8	7
56	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
57	Experimental investigations on startâ€up performance of static nuclear reactor thermal prototype. International Journal of Energy Research, 2020, 44, 3033-3048.	2.2	14
58	RELAP5/MOD3.4 calculation and model evaluation based on upper plenum entrainment experiment in AP1000. Annals of Nuclear Energy, 2020, 138, 107143.	0.9	5
59	Reactivity feedback evaluation during the start-up of the heat pipe cooled nuclear reactors. Progress in Nuclear Energy, 2020, 120, 103217.	1.3	10
60	Performance analysis of automatic depressurization system in advanced PWR during a typical SBLOCA transient using MIDAC. Nuclear Engineering and Technology, 2020, 52, 937-946.	1.1	3
61	Simulation on mass transfer at immiscible liquid interface entrained by single bubble using particle method. Nuclear Engineering and Technology, 2020, 52, 1172-1179.	1.1	9
62	Thermoelectric characteristics analysis of thermionic space nuclear power reactor. International Journal of Energy Research, 2020, 44, 855-868.	2.2	13
63	Experimental research on heat transfer behavior of large scale two-layer salt melt pool based on COPRA facility. Annals of Nuclear Energy, 2020, 138, 107166.	0.9	7
64	Theoretical Study on the Characteristics of Critical Heat Flux in Rectangular Channel of Natural Circulation under Motion Conditions. Science and Technology of Nuclear Installations, 2020, 2020, 1-18.	0.3	2
65	Parametric investigation of radiation heat transfer and evaporation characteristics of a liquid droplet radiator. Aerospace Science and Technology, 2020, 106, 106214.	2.5	6
66	Numerical simulation of a small high-temperature heat pipe cooled reactor with CFD methodology. Nuclear Engineering and Design, 2020, 370, 110907.	0.8	21
67	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
68	Best estimate plus uncertainty analysis of the China advanced large-scale PWR during LBLOCA scenarios. International Journal of Advanced Nuclear Reactor Design and Technology, 2020, 2, 34-42.	0.5	16
69	Performance analysis of the SPRHRS for the 1000 MWe class PWR in coping with station black out. International Journal of Advanced Nuclear Reactor Design and Technology, 2020, 2, 93-102.	0.5	0
70	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
71	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
72	Flow Boiling Pressure Drop Characteristics in Rectangular Channels under Uniform and Non-Uniform Heating. International Journal of Heat and Mass Transfer, 2020, 157, 119811.	2.5	5

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73	Numerical investigation of natural convection characteristics of a heat pipe-cooled passive residual heat removal system for molten salt reactors. Nuclear Science and Techniques/Hewuli, 2020, 31, 1.	1.3	6
74	Theoretical study of steady-state characteristics of natural circulation system. Annals of Nuclear Energy, 2020, 147, 107663.	0.9	2
75	Boiling incipience of liquid sodium flow in an annular channel based on wavelet analysis. Progress in Nuclear Energy, 2020, 121, 103258.	1.3	2
76	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
77	Experimental investigation of a novel heat pipe thermoelectric generator for waste heat recovery and electricity generation. International Journal of Energy Research, 2020, 44, 7450-7463.	2.2	33
78	Research of twoâ€phase density wave instability in reactor core channels with rolling motion. International Journal of Energy Research, 2020, 44, 7323-7341.	2.2	4
79	Experimental study on heat pipe thermoelectric generator for industrial high temperature waste heat recovery. Applied Thermal Engineering, 2020, 175, 115299.	3.0	70
80	Code development and analysis of heat pipe cooled passive residual heat removal system of Molten salt reactor. Annals of Nuclear Energy, 2020, 144, 107527.	0.9	6
81	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
82	Development of a thermo-mechanical coupling code based on an annular fuel sub-channel code. Nuclear Engineering and Design, 2019, 355, 110284.	0.8	3
83	Theoretical investigation of two-phase flow instability between parallel channels of natural circulation in rolling motion. Nuclear Engineering and Design, 2019, 343, 257-268.	0.8	13
84	Numerical study of bubble rising in quiescent liquid under rolling motion conditions using MPS â€“ MAFL method. Annals of Nuclear Energy, 2019, 133, 469-482.	0.9	5
85	Experimental study on spray characteristics of pressure-swirl nozzle in China advanced PWR containment. Nuclear Engineering and Design, 2019, 350, 158-175.	0.8	14
86	Transient analysis of tritium transport characteristics in fluoride-salt-cooled high-temperature reactor. Progress in Nuclear Energy, 2019, 117, 103064.	1.3	5
87	Simulation on Pelletâ€"Cladding Mechanical Interaction of Accident Tolerant Fuel With Coated Cladding. Journal of Nuclear Engineering and Radiation Science, 2019, 5, .	0.2	0
88	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.	0.8	4
89	Numerical study on the single bubble rising behaviors under rolling conditions. Nuclear Engineering and Design, 2019, 349, 183-192.	0.8	43
90	Thermalâ€hydraulic design features of a micronuclear reactor power source applied for multipurpose. International Journal of Energy Research, 2019, 43, 4170-4183.	2.2	24

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91	Experimental study on natural convection heat transfer in a two-layer corium pool based on COPRA facility. Nuclear Engineering and Design, 2019, 349, 136-143.	0.8	11
92	Development and validation of boron diffusion model in nuclear reactor core subchannel analysis. Annals of Nuclear Energy, 2019, 130, 208-217.	0.9	22
93	Experimental study on the natural circulation capability and heat transfer characteristic of liquid lead bismuth eutectic. Progress in Nuclear Energy, 2019, 115, 99-106.	1.3	5
94	Neutronic/thermalâ€hydraulic design features of an improved leadâ€bismuth cooled small modular fast reactor. International Journal of Energy Research, 2019, 43, 3794-3805.	2.2	11
95	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
96	Numerical Study of Bubble Rising and Coalescence Characteristics under Flow Pulsation Based on Particle Method. Science and Technology of Nuclear Installations, 2019, 2019, 1-13.	0.3	0
97	Development of a subchannel analysis code and its application to annular fuel assemblies. Annals of Nuclear Energy, 2019, 129, 428-436.	0.9	9
98	Experimental investigation on saturated pool boiling CHF for downward facing heating surface with different sizes and aspect ratio. International Journal of Thermal Sciences, 2019, 138, 459-466.	2.6	20
99	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
100	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
101	Numerical prediction of CHF based on CFD methodology under atmospheric pressure and low flow rate. Applied Thermal Engineering, 2019, 149, 881-888.	3.0	19
102	Neutronic and Thermo-Hydraulic Analyses of Water-Cooled Blanket Based on Pressurized/Supercritical Water Conditions for CFETR. Journal of Nuclear Engineering and Radiation Science, 2019, 5, .	0.2	1
103	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
104	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
105	Preliminary evaluation of U 3 Si 2 -FeCrAl fuel performance in light water reactors through a multi-physics coupled way. Nuclear Engineering and Design, 2018, 328, 27-35.	0.8	49
106	A comparative study on preliminary performance evaluation of ATFs under normal and accident conditions with FRAP-ATF code. Progress in Nuclear Energy, 2018, 105, 51-60.	1.3	26
107	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
108	Development of safety analysis code for SCWR and its LOCA analysis of CSR1000. Nuclear Engineering and Design, 2018, 327, 100-111.	0.8	3

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109	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.	2.2	8
110	Sub-channel analysis for Pb-Bi-cooled direct contact boiling water fast reactor. International Journal of Energy Research, 2018, 42, 2643-2654.	2.2	6
111	Transient safety analysis for simplified accelerator driven system with gas-lift pump. Progress in Nuclear Energy, 2018, 106, 181-194.	1.3	4
112	Experimental investigation of gas lift pump in a lead-bismuth eutectic loop. Nuclear Engineering and Design, 2018, 330, 516-523.	0.8	19
113	Large eddy simulation on turbulent heat transfer in reactor vessel lower head corium pools. Annals of Nuclear Energy, 2018, 111, 293-302.	0.9	31
114	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.	2.2	8
115	Study of tritium transport characteristics in a transportable fluoride-salt-cooled high-temperature reactor. International Journal of Energy Research, 2018, 42, 1536-1550.	2.2	12
116	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.	1.3	4
117	Hydraulic Characteristics Research on SG Under Tube Plugging Operations Using FLUENT. , 2018, , .		1
118	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.	0.8	10
119	An improved Multiphase Moving Particle Semi-implicit method in bubble rising simulations with large density ratios. Nuclear Engineering and Design, 2018, 340, 370-387.	0.8	23
120	COPRA experiment and numerical research on the behavior of internally-heated melt pool with eutectic salt. Applied Thermal Engineering, 2018, 140, 313-324.	3.0	22
121	Study on safety boundary of flow instability and CHF for parallel channels in motion. Nuclear Engineering and Design, 2018, 335, 219-230.	0.8	32
122	CFD simulation of secondary side fluid flow and heat transfer of the passive residual heat removal heat exchanger. Nuclear Engineering and Design, 2018, 337, 27-37.	0.8	19
123	Conceptual design and analysis of a multipurpose micro nuclear reactor power source. Annals of Nuclear Energy, 2018, 121, 118-127.	0.9	56
124	Reactor Core Design and Analysis for a Micronuclear Power Source. Frontiers in Energy Research, 2018, 6, .	1.2	4
125	Experimental investigation of the frictional pressure drop of steam-water two-phase flow in AP1000 surge line. Experimental Thermal and Fluid Science, 2018, 98, 328-335.	1.5	5
126	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

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127	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
128	Upgrade of FROBA code and its application in thermal-mechanical analysis of space reactor fuel. Nuclear Engineering and Design, 2018, 332, 297-306.	0.8	16
129	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.	0.8	2
130	Performance analysis of heat pipe radiator unit for space nuclear power reactor. Annals of Nuclear Energy, 2017, 103, 74-84.	0.9	55
131	A new unidentified leak detection method based on the EVR ventilation condensate. Progress in Nuclear Energy, 2017, 98, 11-22.	1.3	6
132	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
133	Three dimensional thermal hydraulic characteristic analysis of reactor core based on porous media method. Annals of Nuclear Energy, 2017, 104, 178-190.	0.9	26
134	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
135	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.	0.9	14
136	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
137	Numerical investigation on the dissolution kinetics of ZrO 2 by molten zircaloy using MPS method. Nuclear Engineering and Design, 2017, 319, 117-125.	0.8	11
138	Numerical analysis of the dissolution of uranium dioxide by molten zircaloy using MPS method. Progress in Nuclear Energy, 2017, 100, 1-10.	1.3	15
139	Numerical analysis of the melt behavior in a fuel support piece of the BWR by MPS. Annals of Nuclear Energy, 2017, 102, 422-439.	0.9	25
140	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
141	Transient Safety Analysis of a Transportable Fluoride-Salt-Cooled High-Temperature Reactor Using RELAP5-3D. Nuclear Technology, 2017, 198, 1-16.	0.7	4
142	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
143	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.	0.8	7
144	Experimental Study on Natural Circulation in a Lead Bismuth Eutectic Loop. , 2017, , .		0

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145	Simulation on Pellet-Cladding Mechanical Interaction (PCMI) of Accident Tolerant Fuel (ATF) With Coated Cladding. , 2017, , .		0
146	Experimental Study on the Steam-Water Co-Current Flow Friction Multiplier of the AP1000 Surge Line. , 2017, , .		0
147	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.	3.8	31
148	Model application research for liquid entrainment through ADS-4 pipe in AP1000. Annals of Nuclear Energy, 2017, 110, 1043-1051.	0.9	4
149	Preliminary safety assessment on two LOCAs under ITER-like condition for the optimized CFETR helium cooled solid breeder blanket. Fusion Engineering and Design, 2017, 121, 235-244.	1.0	10
150	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
151	MELCOR severe accident analysis for a natural circulation small modular reactor. Progress in Nuclear Energy, 2017, 100, 197-208.	1.3	26
152	Thermal-mechanical coupling behavior analysis on metal-matrix dispersed plate-type fuel. Progress in Nuclear Energy, 2017, 95, 8-22.	1.3	19
153	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
154	The Development of Candling Module Code in Module In-vessel Degraded Analysis Code MIDAC and the Relevant Calculation for CPR1000 During Large-Break LOCA. Journal of Nuclear Engineering and Radiation Science, 2016, 2, .	0.2	3
155	Numerical Study on the Heat Transfer Characteristics of COPRA-L1 Melt Pool Based on the LES Method. , 2016, , .		0
156	Development of a thermalâ€"mechanical behavior coupling analysis code for a dual-cooled annular fuel element in PWRs. Nuclear Engineering and Design, 2016, 301, 353-365.	0.8	34
157	COPRA experiments on natural convection heat transfer in a volumetrically heated slice pool with high Rayleigh numbers. Annals of Nuclear Energy, 2016, 87, 81-88.	0.9	31
158	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
159	Thermal-hydraulic analysis of the improved TOPAZ-II power system using a heat pipe radiator. Nuclear Engineering and Design, 2016, 307, 218-233.	0.8	23
160	Preliminary design and thermal analysis of a liquid metal heat pipe radiator for TOPAZ-II power system. Annals of Nuclear Energy, 2016, 97, 208-220.	0.9	30
161	PHEBUS FPT-1 simulation by using MELCOR and primary blockage model exploration. Nuclear Engineering and Design, 2016, 307, 119-129.	0.8	9
162	Experimental Investigation of Air-Water CCFL in the Pressurizer Surge Line of AP1000. Nuclear Technology, 2016, 196, 614-640.	0.7	7

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163	Thermal-hydraulic analysis of a fluoride-salt-cooled pebble-bed reactor with CFD methodology. Progress in Nuclear Energy, 2016, 91, 83-96.	1.3	17
164	Development and preliminary validation of a steam generator 3D thermohydraulics analysis code STAF. Nuclear Engineering and Design, 2016, 298, 135-148.	0.8	15
165	Simulation of the small modular reactor severe accident scenario response to SBO using MELCOR code. Progress in Nuclear Energy, 2016, 86, 87-96.	1.3	14
166	COPRA: A large scale experiment on natural convection heat transfer in corium pools with internal heating. Progress in Nuclear Energy, 2016, 86, 132-140.	1.3	39
167	Mechanism study and theoretical simulation on heat split phenomenon in dual-cooled annular fuel element. Annals of Nuclear Energy, 2016, 94, 44-54.	0.9	26
168	Transient thermalâ€"hydraulic analysis of a space thermionic reactor. Annals of Nuclear Energy, 2016, 89, 38-49.	0.9	27
169	Comparative study on effect of airâ€"water and steamâ€"water mediums on liquid entrainment through ADS-4 in AP1000. Experimental Thermal and Fluid Science, 2015, 69, 149-157.	1.5	13
170	Comparison of Hydrogen Generation Rate between CORA-13 Test and MELCOR Simulation: Clad Solid-Phase Oxidation Models Using Self-Developed Code MYCOAC. Nuclear Technology, 2015, 192, 25-34.	0.7	11
171	Analysis of Westinghouse MB2 test using the Steam-generator Thermohydraulics Analysis code STAF. Annals of Nuclear Energy, 2015, 85, 127-136.	0.9	4
172	CFD analysis on subcooled boiling phenomena in PWR coolant channel. Progress in Nuclear Energy, 2015, 81, 254-263.	1.3	24
173	Simulation of the PHEBUS FPT-1 experiment using MELCOR and exploration of the primary core degradation mechanism. Annals of Nuclear Energy, 2015, 85, 193-204.	0.9	6
174	MELCOR simulation of core thermal response during a station blackout initiated severe accident in China pressurized reactor (CPR1000). Progress in Nuclear Energy, 2015, 81, 6-15.	1.3	28
175	Three dimensional neutronic/thermal-hydraulic coupled simulation of MSR in transient state condition. Nuclear Engineering and Design, 2015, 282, 93-105.	0.8	10
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