Jun Wang

List of Publications by Year
in descending order

42 papers	591 citations	623734 14 h-index	642732 23 g-index
42 all docs	42 docs citations	42 times ranked	326 citing authors

#	Article	IF	Citations
1	Microstructure and high-temperature steam oxidation properties of thick Cr coatings prepared by magnetron sputtering for accident tolerant fuel claddings: The role of bias in the deposition process. Corrosion Science, 2020, 165, 108378.	6.6	75
2	A review on thermohydraulic and mechanical-physical properties of SiC, FeCrAl and Ti3SiC2 for ATF cladding. Nuclear Engineering and Technology, 2020, 52, 1-13.	2.3	67
3	A sub-channel analysis code for advanced lead bismuth fast reactor. Progress in Nuclear Energy, 2013, 63, 34-48.	2.9	36
4	The development of Module In-vessel degraded severe accident Analysis Code MIDAC and the relevant research for CPR1000 during the station blackout scenario. Progress in Nuclear Energy, 2014, 76, 44-54.	2.9	36
5	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.	2.9	28
6	Accident tolerant clad material modeling by MELCOR: Benchmark for SURRY Short Term Station Black Out. Nuclear Engineering and Design, 2017, 313, 458-469.	1.7	26
7	Experimental investigation on the characteristics of melt jet breakup in water: The importance of surface tension and Rayleigh-Plateau instability. International Journal of Heat and Mass Transfer, 2019, 132, 388-393.	4.8	25
8	Uncertainty analysis of ATF Cr-coated-Zircaloy on BWR in-vessel accident progression during a station blackout. Reliability Engineering and System Safety, 2021, 213, 107770.	8.9	25
9	Prediction of falling film evaporation on the AP1000 passive containment cooling system using ANSYS FLUENT code. Annals of Nuclear Energy, 2016, 95, 168-175.	1.8	22
10	The development of a zirconium oxidation calculating program module for Module In-vessel Degraded Analysis Code MIDAC. Progress in Nuclear Energy, 2014, 73, 162-171.	2.9	19
11	Potential Recovery Actions from a Severe Accident in a PWR: MELCOR Analysis of a Station Blackout Scenario. Nuclear Technology, 2018, 204, 1-14.	1.2	19
12	Code development and safety analyses for Pbâ€“Bi-cooled direct contact boiling water fast reactor (PBWFR). Progress in Nuclear Energy, 2013, 68, 177-187.	2.9	18
13	Comparison of CORA & MELCOR core degradation simulation and the MELCOR oxidation model. Nuclear Engineering and Design, 2014, 276, 191-201.	1.7	16
14	Quantification of the effect of Cr-coated-Zircaloy cladding during a short term station black out. Nuclear Engineering and Design, 2020, 363, 110678.	1.7	15
15	3D-2D coupling multi-dimension simulation for the heat pipe micro-reactor by MOOSE&SAM. Progress in Nuclear Energy, 2021, 138, 103790.	2.9	14
16	Analysis of KROTOS KS-2 and KS-4 steam explosion experiments with TEXAS-VI. Nuclear Engineering and Design, 2016, 309, 104-112.	1.7	12
17	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.	1.2	11
18	Effect of ATF Cr-coated-Zircaloy on BWR In-vessel Accident Progression during a Station Blackout. Nuclear Engineering and Design, 2021, 372, 110979.	1.7	11

#	Article	IF	Citations
19	Accident tolerant fuels (FeCrAl Cladding & Coating) performance analysis in Boiling Water Reactor (BWR) by the MELCOR 1.8.6 UDGC. Nuclear Engineering and Design, 2021, 371, 110974.	1.7	10
20	PHEBUS FPT-1 simulation by using MELCOR and primary blockage model exploration. Nuclear Engineering and Design, 2016, 307, 119-129.	1.7	9
21	Innovative flow-resistance performance in the single-phase natural circulation loop and relevant experiment verification. International Journal of Heat and Mass Transfer, 2017, 107, 66-73.	4.8	9
22	Transient safety evaluation of the heat pipe microreactor â€“ Potential energy source for hydrogen production. International Journal of Hydrogen Energy, 2021, 46, 38887-38902.	7.1	9
23	Development of CHF models for inner and outer RPV gaps in a meltdown severe accident. Nuclear Engineering and Design, 2013, 265, 1045-1056.	1.7	8
24	DNB type critical heat flux prediction in rod bundles with simplified grid spacer based on Liquid Sublayer Dryout model. Nuclear Engineering and Design, 2019, 351, 94-105.	1.7	8
25	Development of correlations for liquid entrainment through a large-scale inclined branch pipe connected to the main horizontal pipe. Experimental Thermal and Fluid Science, 2018, 96, 128-136.	2.7	7
26	Numerical investigation of azimuthal heat conduction effects on CHF phenomenon in rod bundle channel. Annals of Nuclear Energy, 2018, 121, 203-209.	1.8	7
27	Effectiveness of Cr-Coated Zr-Alloy Clad in Delaying Fuel Degradation for a PWR During a Station Blackout Event. Nuclear Technology, 2020, 206, 467-477.	1.2	7
28	Recent progresses on thermalâ€“hydraulics evaluations of accident tolerant fuel cladding materials. Annals of Nuclear Energy, 2021, 161, 108391.	1.8	7
29	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.	1.8	6
30	Development of cladding oxidation analysis code [COAC] and application for early stage severe accident simulation of AP1000. Progress in Nuclear Energy, 2015, 85, 352-365.	2.9	6
31	Ironâ€“Chromiumâ€“Aluminum (FeCrAl) Cladding Oxidation Kinetics and Auxiliary Feedwater Sensitivity Analysisâ€”Short-Term Station Blackout Simulation of Surry Nuclear Power Plant. Journal of Nuclear Engineering and Radiation Science, 2018, 4, .	0.4	4
32	Spatial temperature distribution of fuel assembly pre-simulation for a new simple core degradation experiment. Progress in Nuclear Energy, 2019, 111, 174-182.	2.9	4
33	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, .	0.4	3
34	Improving the flotation performance of coking coal using the reverse-direct flotation process. Energy Sources, Part A: Recovery, Utilization and Environmental Effects, 2018, 40, 2886-2894.	2.3	3
35	Research on pressure drop characteristics in inverted half U-tube bundle under two-phase cross-flow condition. Annals of Nuclear Energy, 2018, 120, 265-271.	1.8	3
36	An innovation idea for SBLOCA mitigation strategy: Self-coagulation system in comparison with human hemostatic mechanism. Progress in Nuclear Energy, 2021, 138, 103832.	2.9	2

#	Article	IF	Citations
37	Accident Tolerance Cladding Thermal Hydraulic Simulation and Oxidation Kinetic Sensitivity Analysis. , 2017, , .		1
38	Research on the effect of Reynolds correlation in natural convection film condensation. Nuclear Science and Techniques/Hewuli, 2017, 28, 1.	3.4	1
39	Review on Core Degradation and Material Migration Research in Light-Water Reactors. Frontiers in Energy Research, 2018, 6, .	2.3	1
40	The Research on Core Melting Process: Oxidation. , 2013, , .		1
41	Analysis of deficiencies in current prediction method for hydrogen generated from fuel cladding and potential improvement approaches. Nuclear Engineering and Design, 2018, 326, 244-260.	1.7	0
42	Engineering-level system code for severe accident analysis: MELCOR. , 2021, , 417-435.		0