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	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
2	Engineering-level system code for severe accident analysis: MELCOR. , 2021, , 417-435.		0
3	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
4	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
5	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
6	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
7	Recent progresses on thermalâ€“hydraulics evaluations of accident tolerant fuel cladding materials. Annals of Nuclear Energy, 2021, 161, 108391.	1.8	7
8	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
9	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
10	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
11	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
12	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
13	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
14	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
15	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
16	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
17	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
18	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

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19	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
20	Review on Core Degradation and Material Migration Research in Light-Water Reactors. Frontiers in Energy Research, 2018, 6, .	2.3	1
21	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
22	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
23	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
24	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
25	Accident Tolerance Cladding Thermal Hydraulic Simulation and Oxidation Kinetic Sensitivity Analysis. , 2017, , .		1
26	Research on the effect of Reynolds correlation in natural convection film condensation. Nuclear Science and Techniques/Hewuli, 2017, 28, 1.	3.4	1
27	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
28	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
29	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
30	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
31	PHEBUS FPT-1 simulation by using MELCOR and primary blockage model exploration. Nuclear Engineering and Design, 2016, 307, 119-129.	1.7	9
32	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
33	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
34	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
35	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
36	Comparison of CORA & MELCOR core degradation simulation and the MELCOR oxidation model. Nuclear Engineering and Design, 2014, 276, 191-201.	1.7	16

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37	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
38	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
39	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
40	A sub-channel analysis code for advanced lead bismuth fast reactor. Progress in Nuclear Energy, 2013, 63, 34-48.	2.9	36
41	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
42	The Research on Core Melting Process: Oxidation. , 2013, , .		1