Michael Corradini

List of Publications by Year
in descending order

37 papers	1,020 citations	516710 16 h-index	434195 31 g-index
38 all docs	38 docs citations	38 times ranked	776 citing authors

#	Article	IF	Citations
1	An analysis of air-water flow phenomena due to a pipe break under sub-atmospheric pressures using TRACE. Nuclear Engineering and Design, 2021, 374, 111064.	1.7	2
2	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
3	Pool boiling critical heat flux studies of accident tolerant fuel cladding materials. Nuclear Engineering and Design, 2020, 370, 110919.	1.7	9
4	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
5	A fresh look at nuclear energy. Science, 2019, 363, 105-105.	12.6	49
6	Evaluation of critical heat flux of ATF candidate coating materials in pool boiling. Nuclear Engineering and Design, 2019, 354, 110166.	1.7	21
7	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
8	Nuclear Energy in a Carbon-Constrained World: Big Challenges and Big Opportunities. IEEE Power and Energy Magazine, 2019, 17, 69-77.	1.6	32
9	Stratified steam explosion energetics. Nuclear Engineering and Technology, 2019, 51, 95-103.	2.3	7
10	Transient pool boiling heat transfer of oxidized and roughened Zircaloy-4 surfaces during water quenching. International Journal of Heat and Mass Transfer, 2018, 120, 435-446.	4.8	49
11	Flow pattern transition instabilities in a natural circulation cooling facility. Nuclear Engineering and Design, 2018, 332, 267-278.	1.7	8
12	Evaluation of steam corrosion and water quenching behavior of zirconium-silicide coated LWR fuel claddings. Journal of Nuclear Materials, 2018, 499, 256-267.	2.7	54
13	Wire-mesh sensors: A review of methods and uncertainty in multiphase flows relative to other measurement techniques. Nuclear Engineering and Design, 2018, 337, 205-220.	1.7	64
14	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
15	Thermal Conductivity Measurement of Granular UO₂(NO₃)₂ Â· 6H₂O. Nuclear Technology, 2017, 197, 191-200.	1.2	0
16	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
17	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
18	Mechanistic CHF modeling for natural circulation applications in SMR. Nuclear Engineering and Design, 2016, 310, 604-611.	1.7	10

#	Article	IF	Citations
19	A CFD study of wave influence on film steam condensation in the presence of non-condensable gas. Nuclear Engineering and Design, 2016, 305, 303-313.	1.7	27
20	Study of Critical Heat Flux in Natural Convectionâ€“Cooled TRIGA Reactors with Single Annulus and Rod Bundle Geometries. Nuclear Science and Engineering, 2015, 180, 141-153.	1.1	2
21	Bubble Dynamics in Pool Boiling on Nanoparticle-Coated Surfaces. Heat Transfer Engineering, 2015, 36, 1013-1027.	1.9	33
22	A small reactor design for 99Mo production with novel fuel. Journal of Radioanalytical and Nuclear Chemistry, 2015, 305, 23-30.	1.5	2
23	Boiling performance and material robustness of modified surfaces with multi scale structures for fuel cladding development. Nuclear Engineering and Design, 2015, 291, 204-211.	1.7	8
24	Comparison of CORA & MELCOR core degradation simulation and the MELCOR oxidation model. Nuclear Engineering and Design, 2014, 276, 191-201.	1.7	16
25	Monitoring dry-cask storage using thermoelectric powered wireless sensors. , 2013, , .		2
26	Critical Heat Flux in TRIGA-Fueled Reactors Cooled by Natural Convection. Nuclear Science and Engineering, 2012, 172, 249-258.	1.1	7
27	Revisiting Insights from Three Mile Island Unit 2 Postaccident Examinations and Evaluations in View of the Fukushima Daiichi Accident. Nuclear Science and Engineering, 2012, 172, 223-248.	1.1	11
28	Modeling Molten Fuel-Moderator Interactions for the CANDU Flow Blockage Accident. Nuclear Technology, 2010, 169, 97-113.	1.2	1
29	Long-Term Validation of the Molten Fuelâ€“Moderator Interactions Model. Nuclear Technology, 2010, 169, 114-125.	1.2	6
30	Transient Two-Dimensional Hydrodynamic Experiments. Nuclear Science and Engineering, 2010, 165, 180-199.	1.1	0
31	Measurement of supercritical CO2 critical flow: Effects of L/D and surface roughness. Nuclear Engineering and Design, 2009, 239, 949-955.	1.7	31
32	Heat transfer to water at supercritical pressures in a circular and square annular flow geometry. International Journal of Heat and Fluid Flow, 2008, 29, 156-166.	2.4	109
33	CRITICAL FLOW EXPERIMENT AND ANALYSIS FOR SUPERCRITICAL FLUID. Nuclear Engineering and Technology, 2008, 40, 133-138.	2.3	18
34	NEW REACTOR TECHNOLOGY: SAFETY IMPROVEMENTS IN NUCLEAR POWER SYSTEMS. Health Physics, 2007, 93, 547-559.	0.5	2
35	Solid particle effects on heat transfer in a multi-layered molten pool with gas injection. Nuclear Engineering and Design, 2006, 236, 2245-2263.	1.7	1
36	A thermodynamically consistent and fully conservative treatment of contact discontinuities for compressible multicomponent flows. Journal of Computational Physics, 2004, 195, 528-559.	3.8	34

#	Article	IF	Citations
37	Modeling high-speed viscous liquid sheet atomization. International Journal of Multiphase Flow, 1999, 25, 1073-1097.	3.4	313