Tetsuhiro Ozaki

List of Publications by Year
in descending order

19 papers	290 citations	1163117 8 h-index	888059 17 g-index
19 all docs	19 docs citations	19 times ranked	73 citing authors

#	Article	IF	Citations
1	Drift-flux correlation for rod bundle geometries. International Journal of Heat and Fluid Flow, 2014, 48, 1-14.	2.4	61
2	Drift-flux model for rod bundle geometry. Progress in Nuclear Energy, 2015, 83, 229-247.	2.9	52
3	Development of drift-flux model based on 8 Ã— 8 BWR rod bundle geometry experiments under prototypic temperature and pressure conditions. Journal of Nuclear Science and Technology, 2013, 50, 563-580.	1.3	51
4	Modeling of distribution parameter, void fraction covariance and relative velocity covariance for upward steam-water boiling flow in vertical rod bundle. Journal of Nuclear Science and Technology, 2018, 55, 386-399.	1.3	19
5	Development of void fraction-quality correlation for two-phase flow in horizontal and vertical tube bundles. Progress in Nuclear Energy, 2017, 97, 38-52.	2.9	17
6	Constitutive equations for vertical upward two-phase flow in rod bundle. International Journal of Heat and Mass Transfer, 2018, 127, 1252-1266.	4.8	16
7	Modeling of void fraction covariance and relative velocity covariance for upward boiling flow in vertical pipe. International Journal of Heat and Mass Transfer, 2017, 112, 620-629.	4.8	15
8	Channel size effect on drift-flux parameters for adiabatic and boiling two-phase flows. International Journal of Heat and Mass Transfer, 2022, 185, 122410.	4.8	14
9	Development of one-dimensional two-fluid model with consideration of void fraction covariance effect. Journal of Nuclear Science and Technology, 2018, 55, 720-732.	1.3	8
10	Code performance with improved two-group interfacial area concentration correlation for one-dimensional forced convective two-phase flow simulation. Journal of Nuclear Science and Technology, 2018, 55, 911-930.	1.3	7
11	Simplified two-group two-fluid model for three-dimensional two-phase flow Computational Fluid Dynamics for vertical upward flow. Progress in Nuclear Energy, 2018, 108, 503-516.	2.9	7
12	Effect of interfacial drag force model on code prediction for upward adiabatic two-phase bubbly flow in vertical channels. Experimental and Computational Multiphase Flow, 2020, 2, 212-224.	3.9	6
13	Effect of void fraction covariance on two-fluid model based code calculation in pipe flow. Progress in Nuclear Energy, 2018, 108, 319-333.	2.9	5
14	Improvement of Two-Fluid Model Based 1D Code with the State-of-the-Art Constitutive Equations for Two-Phase Flow in Rod Bundle. Japanese Journal of Multiphase Flow, 2020, 34, 342-351.	0.3	3
15	Thermal-hydraulic constitutive equations implemented in the system analysis code AMAGI for nuclear power reactor. Progress in Nuclear Energy, 2021, 141, 103962.	2.9	3
16	Effect of compensation error in drift-flux parameters on predictions of thermal-hydraulic parameters in nuclear safety system analysis codes. Progress in Nuclear Energy, 2016, 88, 398-411.	2.9	2
17	Modeling of Sub-Channel Scale Distribution Parameter and Its Validation Based on Database of Rod Bundle Void Fraction Measurement Test Under Prototypic BWR Conditions. Japanese Journal of Multiphase Flow, 2021, 35, 543-550.	0.3	2
18	Sensitivity analysis using improved two-fluid model-based 1D code with the state-of-the-art constitutive equations for two-phase flow in rod bundle. Experimental and Computational Multiphase Flow, 0, , 1.	3.9	2

#	Article	IF	Citations
19	Effect of interfacial area concentration on one-dimensional code simulation of adiabatic two-phase flows in vertical large size channels. Nuclear Engineering and Design, 2021, 385, 111502.	1.7	0