Kun Mo

List of Publications by Year
in descending order

58 papers	1,090 citations	331259 21 h-index	433756 31 g-index
59 all docs	59 docs citations	59 times ranked	920 citing authors

#	Article	IF	Citations
1	A dynamic neural network aggregation model for transient diagnosis in nuclear power plants. Progress in Nuclear Energy, 2007, 49, 262-272.	1.3	69
2	Experimental and modeling results of creepâ€“fatigue life of Inconel 617 and Haynes 230 at 850Â°C. Journal of Nuclear Materials, 2013, 432, 94-101.	1.3	52
3	Mechanism of plastic deformation of a Ni-based superalloy for VHTR applications. Journal of Nuclear Materials, 2013, 441, 695-703.	1.3	48
4	On the microstructure and strengthening mechanism in oxide dispersion-strengthened 316 steel: A coordinated electron microscopy, atom probe tomography and in situ synchrotron tensile investigation. Materials Science & Engineering A: Structural Materials: Properties, Microstructure and Processing, 2015, 639, 585-596.	2.6	48
5	Atom probe study of irradiation-enhanced Î±â€² precipitation in neutron-irradiated Feâ€“Cr model alloys. Journal of Nuclear Materials, 2015, 462, 242-249.	1.3	46
6	Synchrotron study on load partitioning between ferrite/martensite and nanoparticles of a 9Cr ODS steel. Journal of Nuclear Materials, 2014, 455, 376-381.	1.3	44
7	The microstructure and mechanical properties of Al-containing 9Cr ODS ferritic alloy. Journal of Alloys and Compounds, 2015, 648, 223-228.	2.8	44
8	The interfacial orientation relationship of oxide nanoparticles in a hafnium-containing oxide dispersion-strengthened austenitic stainless steel. Materials Characterization, 2015, 101, 136-143.	1.9	43
9	Effect of creep and oxidation on reduced fatigue life of Ni-based alloy 617 at 850Â°C. Journal of Nuclear Materials, 2014, 444, 393-403.	1.3	41
10	In situ synchrotron tensile investigations on the phase responses within an oxide dispersion-strengthened (ODS) 304 steel. Materials Science & Engineering A: Structural Materials: Properties, Microstructure and Processing, 2015, 625, 146-152.	2.6	33
11	Bubble morphology in U3Si2 implanted by high-energy Xe ions at 300ÂÂ°C. Journal of Nuclear Materials, 2017, 495, 146-153.	1.3	33
12	High Temperature Aging and Corrosion Study on Alloy 617 and Alloy 230. Journal of Engineering for Gas Turbines and Power, 2011, 133, .	0.5	32
13	Short Communication on â€œIn-situ TEM ion irradiation investigations on U3Si2 at LWR temperaturesâ€ Journal of Nuclear Materials, 2017, 484, 168-173.	1.3	31
14	The comparison of microstructures and mechanical properties between 14Cr-Al and 14Cr-Ti ferritic ODS alloys. Materials and Design, 2016, 98, 61-67.	3.3	29
15	The evolution of internal stress and dislocation during tensile deformation in a 9Cr ferritic/martensitic (F/M) ODS steel investigated by high-energy X-rays. Journal of Nuclear Materials, 2015, 467, 50-57.	1.3	28
16	Biaxial thermal creep of Inconel 617 and Haynes 230 at 850 and 950Â°C. Journal of Nuclear Materials, 2014, 447, 28-37.	1.3	26
17	Grain growth and pore coarsening in dense nanoâ€crystalline UO_{2+<i>x</i>} fuel pellets. Journal of the American Ceramic Society, 2017, 100, 2651-2658.	1.9	26
18	Nano-crystallization induced by high-energy heavy ion irradiation in UO2. Scripta Materialia, 2018, 155, 169-174.	2.6	25

#	Article	IF	Citations
19	Microstructure investigations of U3Si2 implanted by high-energy Xe ions at 600â€¯Â°C. Journal of Nuclear Materials, 2018, 503, 314-322.	1.3	23
20	Lattice strain and damage evolution of 9â€“12%Cr ferritic/martensitic steel during in situ tensile test by X-ray diffraction and small angle scattering. Journal of Nuclear Materials, 2010, 407, 10-15.	1.3	22
21	Low cycle fatigue and creep-fatigue behavior of Ni-based alloy 230 at 850 Â°C. Materials Science & Engineering A: Structural Materials: Properties, Microstructure and Processing, 2013, 563, 152-162.	2.6	22
22	The evolution mechanism of the dislocation loops in irradiated lanthanum doped cerium oxide. Journal of Nuclear Materials, 2014, 445, 209-217.	1.3	21
23	A neural network based operation guidance system for procedure presentation and operation validation in nuclear power plants. Annals of Nuclear Energy, 2007, 34, 813-823.	0.9	20
24	Load partitioning between ferrite/martensite and dispersed nanoparticles of a 9Cr ferritic/martensitic (F/M) ODS steel at high temperatures. Materials Science & Engineering A: Structural Materials: Properties, Microstructure and Processing, 2015, 637, 75-81.	2.6	20
25	In situ synchrotron tensile investigations on 14YWT, MA957, and 9-Cr ODS alloys. Journal of Nuclear Materials, 2016, 471, 289-298.	1.3	19
26	High-energy synchrotron study of in-pile-irradiated Uâ€“Mo fuels. Scripta Materialia, 2016, 114, 146-150.	2.6	18
27	Effect of orientation on plastic deformations of Alloy 617 for VHTR applications. Journal of Nuclear Materials, 2013, 443, 366-377.	1.3	17
28	In situ synchrotron investigation of grain growth behavior of nano-grained UO2. Scripta Materialia, 2017, 131, 29-32.	2.6	16
29	Temperature and particle size effects on flow localization of 9â€“12%Cr ferritic/martensitic steel by in situ X-ray diffraction and small angle scattering. Journal of Nuclear Materials, 2010, 398, 220-226.	1.3	15
30	Load-partitioning in an oxide dispersion-strengthened 310 steel at elevated temperatures. Materials and Design, 2016, 111, 622-630.	3.3	14
31	Synchrotron Radiation Study on Alloy 617 and Alloy 230 for VHTR Application. Journal of Pressure Vessel Technology, Transactions of the ASME, 2013, 135, .	0.4	13
32	MeV per nucleon ion irradiation of nuclear materials with high energy synchrotron X-ray characterization. Journal of Nuclear Materials, 2016, 471, 266-271.	1.3	12
33	Phase decomposition and bubble evolution in Xe implanted U3Si2 at 450â˜C. Journal of Nuclear Materials, 2019, 518, 108-116.	1.3	11
34	Correlation between crystallographic orientation and surface faceting in UO2. Journal of Nuclear Materials, 2016, 478, 176-184.	1.3	10
35	The effect of thermal-aging on the microstructure and mechanical properties of 9Cr ferritic/martensitic ODS alloy. Journal of Nuclear Materials, 2019, 522, 212-219.	1.3	10
36	Investigation of High-Energy Ion-Irradiated MA957 Using Synchrotron Radiation under In-Situ Tension. Materials, 2016, 9, 15.	1.3	9

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37	Size-dependent characteristics of ultra-fine oxygen-enriched nanoparticles in austenitic steels. Journal of Nuclear Materials, 2016, 480, 195-201.	1.3	9
38	Investigation of thermal aging effects on the tensile properties of Alloy 617 by in-situ synchrotron wide-angle X-ray scattering. Materials Science & Engineering A: Structural Materials: Properties, Microstructure and Processing, 2016, 651, 55-62.	2.6	9
39	Development of an Integrated Decision Support System to Aid the Cognitive Activities of Operators in Main Control Rooms of Nuclear Power Plants. , 2007, , .		8
40	Characterization of high energy Xe ion irradiation effects in single crystal molybdenum with depth-resolved synchrotron microbeam diffraction. Journal of Nuclear Materials, 2016, 471, 272-279.	1.3	8
41	Temperature effect of elastic anisotropy and internal strain development in advanced nanostructured alloys: An in-situ synchrotron X-ray investigation. Materials Science & Engineering A: Structural Materials: Properties, Microstructure and Processing, 2017, 692, 53-61.	2.6	7
42	Heat transfer simulations of the UO2 particleâ€"graphite system in TREAT fuel. Nuclear Engineering and Design, 2015, 293, 313-322.	0.8	6
43	Interaction between Al and atomic layer deposited (ALD) ZrN under high-energy heavy ion irradiation. Acta Materialia, 2019, 164, 788-798.	3.8	6
44	FIPD: The SFR metallic fuels irradiation & physics database. Nuclear Engineering and Design, 2021, 380, 111225.	0.8	6
45	In situ TEM and synchrotron characterization of Uâ€"10Mo thin specimen annealed at the fast reactor temperature regime. Materials Characterization, 2015, 110, 208-214.	1.9	5
46	Biaxial Thermal Creep of Alloy 617 and Alloy 230 for VHTR Applications. Journal of Engineering Materials and Technology, Transactions of the ASME, 2016, 138, .	0.8	5
47	A study on texture stability and the biaxial creep behavior of as-hydrided CWSR Zircaloy-4 cladding at the effective stresses from 55ÂMPa to 65ÂMPa and temperatures from 300ÂÂ°C to 400ÂÂ°C. Journal of Nuclear Materials, 2022, 564, 153688.	1.3	5
48	Lattice strain mapping of cracks and indentations in UO2 using synchrotron microdiffraction. Journal of Nuclear Materials, 2020, 529, 151943.	1.3	4
49	In situ synchrotron tensile investigations on ultrasonic additive manufactured (UAM) zirconium. Journal of Nuclear Materials, 2022, 568, 153843.	1.3	4
50	Microstructural Evolution of Alloy 617 and Alloy 230 Following High Temperature Aging. , 2010, , .		3
51	The incorporation and migration of a single xenon atom in ceria. Journal of Nuclear Materials, 2014, 449, 242-247.	1.3	3
52	Effect of reactor radiation on the thermal conductivity of TREAT fuel. Journal of Nuclear Materials, 2017, 487, 453-460.	1.3	3
53	An exploration of measuring lower-length-scale structures in nuclear materials: Thermal conductivity of U-Mo fuel particle. Journal of Nuclear Materials, 2019, 527, 151797.	1.3	3
54	Strain-Rate Sensitivity Analysis for Ni Alloys to Be Used in Very High Temperature Reactors. Nuclear Technology, 2013, 183, 455-463.	0.7	2

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55	Depth profile of oxide volume fractions of Zircaloy-2 in high-temperature steam: An in-situ synchrotron radiation study. Journal of Nuclear Materials, 2014, 454, 192-199.	1.3	2
56	TEM and XAS investigation of fission gas behaviors in U-Mo alloy fuels through ion beam irradiation. Journal of Nuclear Materials, 2017, 494, 165-171.	1.3	1
57	Synchrotron Radiation Study on Alloy 617 and Alloy 230 for VHTR Application. , 2011, , .		0
58	Microstructure investigations of temperature effect on Al-UMo diffusion couples irradiated by swift Xe ions. Journal of Nuclear Materials, 2021, 547, 152757.	1.3	0