Dong-Seong Sohn

List of Publications by Year
in descending order

75 papers	718 citations	623188 14 h-index	676716 22 g-index
75 all docs	75 docs citations	75 times ranked	444 citing authors

#	Article	IF	Citations
1	Multi-layer coating of silicon carbide and pyrolytic carbon on UO2 pellets by a combustion reaction. Journal of Nuclear Materials, 2000, 281, 163-170.	1.3	49
2	Pore pressure and swelling in the rim region of LWR high burnup UO2 fuel. Journal of Nuclear Materials, 2001, 295, 213-220.	1.3	45
3	FREE VIBRATION OF CLAMPEDâ€“FREE CIRCULAR CYLINDRICAL SHELL WITH A PLATE ATTACHED AT AN ARBITRARY AXIAL POSITION. Journal of Sound and Vibration, 1998, 213, 75-88.	2.1	35
4	Enhanced high-temperature oxidation resistance of a zirconium alloy cladding by high-temperature preformed oxide on the cladding. Corrosion Science, 2018, 131, 116-125.	3.0	32
5	FABRICATION OF GD CONTAINING DUPLEX STAINLESS STEEL SHEET FOR NEUTRON ABSORBING STRUCTURAL MATERIALS. Nuclear Engineering and Technology, 2013, 45, 689-694.	1.1	31
6	Characteristics of GdxMyOz (M=Ti, Zr or Al) as a burnable absorber. Journal of Nuclear Materials, 2008, 372, 340-349.	1.3	28
7	Cosmos: A computer code to analyze LWR UO2 and MOX fuel up to high burnup. Annals of Nuclear Energy, 1999, 26, 47-67.	0.9	24
8	High-temperature steam oxidation and oxide crack effects of Zr-1Nb-1Sn-0.1Fe fuel cladding. Journal of Nuclear Materials, 2017, 496, 343-352.	1.3	24
9	Analysis of fission gas release and gaseous swelling in UO2 fuel under the effect of external restraint. Journal of Nuclear Materials, 2000, 280, 86-98.	1.3	20
10	Thermal conductivity modeling of U-Mo/Al dispersion fuel. Journal of Nuclear Materials, 2015, 466, 576-582.	1.3	20
11	Mechanical analysis of UMo/Al dispersion fuel. Journal of Nuclear Materials, 2015, 466, 509-521.	1.3	20
12	Release of unstable fission products from defective fuel rods to the coolant of a PWR. Journal of Nuclear Materials, 1994, 209, 248-258.	1.3	18
13	Molecular dynamics simulation of the pressureâ€“volumeâ€“temperature data of xenon for a nuclear fuel. Journal of Nuclear Materials, 2008, 372, 89-93.	1.3	15
14	An analysis method for the fuel rod gap inventory of unstable fission products during steady-state operation. Journal of Nuclear Materials, 1994, 209, 62-78.	1.3	14
15	Fabrication method and thermal conductivity assessment of molybdenum-precipitated uranium dioxide pellets. Journal of Nuclear Materials, 2006, 352, 151-156.	1.3	13
16	The irradiation test of inert-matrix fuel in comparison to uranium plutonium mixed oxide fuel at the halden reactor. Progress in Nuclear Energy, 2001, 38, 309-312.	1.3	12
17	Three-dimensional simulation of threshold porosity for fission gas release in the rim region of LWR UO2 fuel. Journal of Nuclear Materials, 2003, 321, 249-255.	1.3	12
18	Improvement of Fuel Performance Code COSMOS with Recent In-Pile Data for MOX and UO₂Fuels. Nuclear Technology, 2007, 157, 53-64.	0.7	12

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19	Irradiation tests for U3Siâ€“Al dispersion fuels with aluminum cladding. Journal of Nuclear Materials, 2008, 373, 9-15.	1.3	12
20	Rim Characteristics and Their Effects on the Thermal Conductivity in High Burnup UO₂Fuel. Journal of Nuclear Science and Technology, 2001, 38, 45-52.	0.7	11
21	Modeling and parametric studies of the effect of inhomogeneity on fission gas release in LWR MOX fuel. Annals of Nuclear Energy, 2002, 29, 271-286.	0.9	11
22	Corrosion and Wear Properties of Cold Rolled 0.087% Gd Lean Duplex Stainless Steels for Neutron Absorbing Material. Nuclear Engineering and Technology, 2016, 48, 164-168.	1.1	11
23	Development of PRIME for irradiation performance analysis of U-Mo/Al dispersion fuel. Journal of Nuclear Materials, 2018, 502, 331-348.	1.3	11
24	Measurement of the specific heat of Zrâ€“40wt%U metallic fuel. Journal of Nuclear Materials, 2007, 360, 315-320.	1.3	10
25	Thermal stability of co-extruded Uâ€“Zr/Zrâ€“Nb alloys. Journal of Nuclear Materials, 2008, 373, 275-279.	1.3	10
26	Fabrication method for UO2 pellets with large grains or a single grain by sintering in air. Journal of Nuclear Materials, 2008, 375, 209-212.	1.3	10
27	Microstructural analysis of preformed oxides on a zirconium alloy before and after subsequent oxidation at 1000â€“1200â€¯Â°C. Corrosion Science, 2018, 139, 410-420.	3.0	10
28	Development of integral type spent fuel pool storage rack with gadolinium and europium-containing structure materials. Annals of Nuclear Energy, 2019, 130, 107-117.	0.9	10
29	Effects of sintering processes on the duplex grain structure of UO2. Journal of Nuclear Materials, 1993, 200, 41-49.	1.3	9
30	Statistical failure analysis of metallic U-10zr/HT9 fast reactor fuel pin by considering the weibull distribution and cumulative damage fraction. Annals of Nuclear Energy, 1998, 25, 1441-1453.	0.9	9
31	Evaluation of a pellet-clad mechanical interaction in mixed oxide fuels during power transients by using axisymmetric finite element modeling. Nuclear Engineering and Design, 2004, 231, 39-50.	0.8	9
32	Effect of stress evolution on microstructural behavior in U-Mo/Al dispersion fuel. Journal of Nuclear Materials, 2017, 487, 265-279.	1.3	9
33	Microstructural characteristics and different effects of 800â€“1200â€¯Â°C preformed oxides on high-temperature steam oxidation of a zirconium alloy cladding. Journal of Alloys and Compounds, 2018, 753, 119-129.	2.8	9
34	Study of mechanism of oxidation resistance enhancement induced by preformed oxide on zirconium alloys. Corrosion Science, 2019, 158, 108105.	3.0	9
35	Thermal conductivity of Uâ€“Mo/Al dispersion fuel: effects of particle shape and size, stereography, and heat generation. Journal of Nuclear Science and Technology, 2015, 52, 1328-1337.	0.7	8
36	Structural stability analysis of waste packages containing low- and intermediate-level radioactive waste in a silo-type repository. Nuclear Engineering and Technology, 2021, 53, 1524-1533.	1.1	8

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37	A comparative analysis of UO2 and MOX fuel behavior under reactivity initiated accident. Annals of Nuclear Energy, 1997, 24, 859-870.	0.9	7
38	Measurement of the thermal properties of gadolinium and dysprosium titanate. Thermochimica Acta, 2007, 455, 100-104.	1.2	7
39	Zircaloy-4 cladding corrosion model covering a wide range of PWR experiences. Journal of Nuclear Materials, 2008, 378, 127-133.	1.3	7
40	Thermal properties of U-7Mo/Al dispersion fuel. Journal of Nuclear Materials, 2017, 496, 274-285.	1.3	7
41	Thermophysical properties of heat-treated U-7Mo/Al dispersion fuel. Journal of Nuclear Materials, 2018, 501, 31-44.	1.3	7
42	Thermophysical properties of sintered aluminum-silicon. Journal of Alloys and Compounds, 2018, 749, 1028-1035.	2.8	6
43	Effect of coating thickness and annealing temperature on ZrN coating failure of U-Mo particles under heat treatment. Journal of Nuclear Materials, 2018, 507, 347-359.	1.3	6
44	Effect of pre-oxidation temperature on the ductility of a zirconium alloy cladding under simulated accident conditions. Journal of Nuclear Materials, 2019, 515, 80-90.	1.3	6
45	Characterization of Uâ€“Nbâ€“Zr dispersion fuel prepared by centrifugal atomization process. Journal of Nuclear Materials, 1999, 265, 38-43.	1.3	5
46	Variation of the lattice parameter and thermal expansion coefficient of (U,Dy)O2 as a function of DyO1.5 content. Journal of Alloys and Compounds, 2006, 407, 263-267.	2.8	5
47	FUEL PERFORMANCE CODE COSMOS FOR ANALYSIS OF LWR UO2AND MOX FUEL. Nuclear Engineering and Technology, 2011, 43, 499-508.	1.1	5
48	Miniaturized disk bend tests of neutron irradiated path a type alloys. Journal of Nuclear Materials, 1984, 122, 146-151.	1.3	4
49	Modeling of the Rim Effect on Thermal Behavior of High-Burnup UO2Fuel. Nuclear Technology, 1999, 127, 151-159.	0.7	4
50	Modeling of creep behavior of Zircaloy-4 by considering metallurgical effect. Annals of Nuclear Energy, 2002, 29, 1-12.	0.9	4
51	Thermal expansion of near stoichiometric (U,Er)O2 solid solutions. Materials Letters, 2006, 60, 1480-1483.	1.3	4
52	An extension of the two-zone method for evaluating a fission gas release under an irradiation-induced resolution flux. Journal of Nuclear Materials, 2008, 373, 280-288.	1.3	4
53	Analytical local stress model for UMo/Al dispersion fuel. Journal of Nuclear Materials, 2020, 528, 151881.	1.3	4
54	A two-zone method with an enhanced accuracy for a numerical solution of the diffusion equation. Journal of Nuclear Materials, 2006, 359, 139-149.	1.3	3

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55	Dependence of thickness, morphology, and crystallographic properties of Mo and ZrN coatings on U-Mo substrate size. Journal of Nuclear Materials, 2018, 512, 156-168.	1.3	3
56	Short communication on â€œself-crack-healing behavior of oxide formed on a zirconium alloy cladding tubeâ€. Journal of Nuclear Materials, 2019, 526, 151749.	1.3	3
57	Gas releases and rod internal pressure of BN-added UO₂fuel. Journal of Nuclear Science and Technology, 2015, 52, 1208-1216.	0.7	2
58	Application of a Dynamic-Nanoindentation Method to Analyze the Local Structure of an Fe-18 at.% Gd Cast Alloy. Nuclear Engineering and Technology, 2017, 49, 576-580.	1.1	2
59	Preliminary Estimation of Long-lived Activation Products in the Reactor Structures of SMART. Progress in Nuclear Science and Technology, 2011, 1, 24-27.	0.3	2
60	CORRELATION BETWEEN THE TENSILE STRENGTH AND CORROSION BEHAVIOR OF HEAT TREATED ZR-1.0NB ALLOY. Nuclear Engineering and Technology, 2008, 40, 505-510.	1.1	2
61	Zirconium metal recovery from irradiated radioactive zirconium alloy via c <scp>hlorideâ€based</scp> electrorefining and thermal decomposition of <scp>ZrCl</scp>. International Journal of Energy Research, 2022, 46, 6164-6176.	2.2	2
62	Modeling of fission gas release in MOX fuel considering the distribution of Pu-rich particles. AIP Conference Proceedings, 2000, , .	0.3	1
63	A Unified Thermal Conductivity Model of LWR MOX Fuel Considering Its Microstructural Characteristics. Journal of Nuclear Science and Technology, 2002, 39, 705-708.	0.7	1
64	Resintering Behavior in Oxidativdy Sintered UO2 and UO2-5wt%CeO2 Pellet. Journal of Nuclear Science and Technology, 2002, 39, 697-700.	0.7	1
65	Sintering of U3O8-CeO2 Powder Mixture and Its Microstructure Evolution in Different Atmospheres. Journal of Nuclear Science and Technology, 2002, 39, 693-696.	0.7	1
66	The role of oxygen potential in the sintering of UO2â€“5wt%CeO2 powder mixed with M3O8 (M=U+Ce). Journal of Nuclear Materials, 2005, 344, 254-258.	1.3	1
67	Sintering behavior of Uâ€“80at.%Zr powder compacts in a vacuum environment. Journal of Nuclear Materials, 2008, 372, 394-399.	1.3	1
68	Rim Characteristics and Their Effects on the Thermal Conductivity in High Burnup UO2 Fuel.. Journal of Nuclear Science and Technology, 2001, 38, 45-52.	0.7	1
69	Yield Stress Determination by the Massachusetts Institute of Technology Miniaturized Disk Bend Test. Nuclear Technology, 1990, 92, 383-388.	0.7	0
70	Behavior of unirradiated Zr based uranium metal fuel under reactivity initiated accident conditions. Nuclear Engineering and Design, 2008, 238, 1592-1600.	0.8	0
71	A measurement of a control rod drop using an LVDT. Proceedings of SPIE, 2009, , .	0.8	0
72	Irradiation Test of MOX Fuel Rods Fabricated by Attrition-Milling and Analysis of In-Pile Data with COSMOS Code. Nuclear Technology, 2010, 172, 246-254.	0.7	0

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73	Performance analysis of UO2 fuel rod for low-power research reactor application. Annals of Nuclear Energy, 2019, 129, 233-239.	0.9	0
74	Development of Control Rod Position Indicator Using Seismic-Resistance Reed Switches. , 2009, , .		0
75	Verification of the Output Signal Change of Control Rod Position Indicator at the End of Its Life Time. , 2010, , .		0