

# Dong-Seong Sohn

## List of Publications by Year in descending order

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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	Zirconium metal recovery from irradiated radioactive zirconium alloy via chloride-based electrorefining and thermal decomposition of $ZrCl_4$ . International Journal of Energy Research, 2022, 46, 6164-6176.	2.2	2
2	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
3	Analytical local stress model for UMo/Al dispersion fuel. Journal of Nuclear Materials, 2020, 528, 151881.	1.3	4
4	Short communication on self-crack-healing behavior of oxide formed on a zirconium alloy cladding tube. Journal of Nuclear Materials, 2019, 526, 151749.	1.3	3
5	Study of mechanism of oxidation resistance enhancement induced by preformed oxide on zirconium alloys. Corrosion Science, 2019, 158, 108105.	3.0	9
6	Performance analysis of UO <sub>2</sub> fuel rod for low-power research reactor application. Annals of Nuclear Energy, 2019, 129, 233-239.	0.9	0
7	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
8	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
9	Development of PRIME for irradiation performance analysis of U-Mo/Al dispersion fuel. Journal of Nuclear Materials, 2018, 502, 331-348.	1.3	11
10	Thermophysical properties of heat-treated U-7Mo/Al dispersion fuel. Journal of Nuclear Materials, 2018, 501, 31-44.	1.3	7
11	Microstructural characteristics and different effects of 800~1200°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
12	Thermophysical properties of sintered aluminum-silicon. Journal of Alloys and Compounds, 2018, 749, 1028-1035.	2.8	6
13	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
14	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
15	Microstructural analysis of preformed oxides on a zirconium alloy before and after subsequent oxidation at 1000~1200°C. Corrosion Science, 2018, 139, 410-420.	3.0	10
16	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
17	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
18	Effect of stress evolution on microstructural behavior in U-Mo/Al dispersion fuel. Journal of Nuclear Materials, 2017, 487, 265-279.	1.3	9

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19	Thermal properties of U-7Mo/Al dispersion fuel. Journal of Nuclear Materials, 2017, 496, 274-285.	1.3	7
20	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
21	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
22	Thermal conductivity modeling of U-Mo/Al dispersion fuel. Journal of Nuclear Materials, 2015, 466, 576-582.	1.3	20
23	Mechanical analysis of UMo/Al dispersion fuel. Journal of Nuclear Materials, 2015, 466, 509-521.	1.3	20
24	Gas releases and rod internal pressure of BN-added UO <sub>2</sub> fuel. Journal of Nuclear Science and Technology, 2015, 52, 1208-1216.	0.7	2
25	Thermal conductivity of U <sup>18</sup> 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
26	FABRICATION OF GD CONTAINING DUPLEX STAINLESS STEEL SHEET FOR NEUTRON ABSORBING STRUCTURAL MATERIALS. Nuclear Engineering and Technology, 2013, 45, 689-694.	1.1	31
27	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
28	FUEL PERFORMANCE CODE COSMOS FOR ANALYSIS OF LWR UO <sub>2</sub> AND MOX FUEL. Nuclear Engineering and Technology, 2011, 43, 499-508.	1.1	5
29	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
30	Verification of the Output Signal Change of Control Rod Position Indicator at the End of Its Life Time. , 2010, , .		0
31	A measurement of a control rod drop using an LVDT. Proceedings of SPIE, 2009, , .	0.8	0
32	Development of Control Rod Position Indicator Using Seismic-Resistance Reed Switches. , 2009, , .		0
33	Thermal stability of co-extruded U <sup>18</sup> Zr/Zr <sup>18</sup> Nb alloys. Journal of Nuclear Materials, 2008, 373, 275-279.	1.3	10
34	Zircaloy-4 cladding corrosion model covering a wide range of PWR experiences. Journal of Nuclear Materials, 2008, 378, 127-133.	1.3	7
35	Molecular dynamics simulation of the pressure <sup>18</sup> volume <sup>18</sup> temperature data of xenon for a nuclear fuel. Journal of Nuclear Materials, 2008, 372, 89-93.	1.3	15
36	Characteristics of GdxMyOz (M=Ti, Zr or Al) as a burnable absorber. Journal of Nuclear Materials, 2008, 372, 340-349.	1.3	28

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37	Irradiation tests for U <sub>3</sub> Si-Al dispersion fuels with aluminum cladding. Journal of Nuclear Materials, 2008, 373, 9-15.	1.3	12
38	Sintering behavior of U <sup>80at.%</sup> Zr powder compacts in a vacuum environment. Journal of Nuclear Materials, 2008, 372, 394-399.	1.3	1
39	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
40	Fabrication method for UO <sub>2</sub> pellets with large grains or a single grain by sintering in air. Journal of Nuclear Materials, 2008, 375, 209-212.	1.3	10
41	Behavior of unirradiated Zr based uranium metal fuel under reactivity initiated accident conditions. Nuclear Engineering and Design, 2008, 238, 1592-1600.	0.8	0
42	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
43	Improvement of Fuel Performance Code COSMOS with Recent In-Pile Data for MOX and UO <sub>2</sub> Fuels. Nuclear Technology, 2007, 157, 53-64.	0.7	12
44	Measurement of the thermal properties of gadolinium and dysprosium titanate. Thermochemica Acta, 2007, 455, 100-104.	1.2	7
45	Measurement of the specific heat of Zr <sup>40wt%</sup> U metallic fuel. Journal of Nuclear Materials, 2007, 360, 315-320.	1.3	10
46	Variation of the lattice parameter and thermal expansion coefficient of (U,Dy)O <sub>2</sub> as a function of DyO <sub>1.5</sub> content. Journal of Alloys and Compounds, 2006, 407, 263-267.	2.8	5
47	Fabrication method and thermal conductivity assessment of molybdenum-precipitated uranium dioxide pellets. Journal of Nuclear Materials, 2006, 352, 151-156.	1.3	13
48	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
49	Thermal expansion of near stoichiometric (U,Er)O <sub>2</sub> solid solutions. Materials Letters, 2006, 60, 1480-1483.	1.3	4
50	The role of oxygen potential in the sintering of UO <sub>2</sub> <sup>5wt%</sup> CeO <sub>2</sub> powder mixed with M <sub>3</sub> O <sub>8</sub> (M=U+Ce). Journal of Nuclear Materials, 2005, 344, 254-258.	1.3	1
51	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
52	Three-dimensional simulation of threshold porosity for fission gas release in the rim region of LWR UO <sub>2</sub> fuel. Journal of Nuclear Materials, 2003, 321, 249-255.	1.3	12
53	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
54	Resintering Behavior in Oxidatively Sintered UO <sub>2</sub> and UO <sub>2</sub> -5wt%CeO <sub>2</sub> Pellet. Journal of Nuclear Science and Technology, 2002, 39, 697-700.	0.7	1

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55	Sintering of U <sub>3</sub> O <sub>8</sub> -CeO <sub>2</sub> Powder Mixture and Its Microstructure Evolution in Different Atmospheres. Journal of Nuclear Science and Technology, 2002, 39, 693-696.	0.7	1
56	Modeling of creep behavior of Zircaloy-4 by considering metallurgical effect. Annals of Nuclear Energy, 2002, 29, 1-12.	0.9	4
57	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
58	Rim Characteristics and Their Effects on the Thermal Conductivity in High Burnup UO <sub>2</sub> Fuel. Journal of Nuclear Science and Technology, 2001, 38, 45-52.	0.7	11
59	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
60	Pore pressure and swelling in the rim region of LWR high burnup UO <sub>2</sub> fuel. Journal of Nuclear Materials, 2001, 295, 213-220.	1.3	45
61	Rim Characteristics and Their Effects on the Thermal Conductivity in High Burnup UO <sub>2</sub> Fuel.. Journal of Nuclear Science and Technology, 2001, 38, 45-52.	0.7	1
62	Analysis of fission gas release and gaseous swelling in UO <sub>2</sub> fuel under the effect of external restraint. Journal of Nuclear Materials, 2000, 280, 86-98.	1.3	20
63	Multi-layer coating of silicon carbide and pyrolytic carbon on UO <sub>2</sub> pellets by a combustion reaction. Journal of Nuclear Materials, 2000, 281, 163-170.	1.3	49
64	Modeling of fission gas release in MOX fuel considering the distribution of Pu-rich particles. AIP Conference Proceedings, 2000, , .	0.3	1
65	Characterization of U-Nb-Zr dispersion fuel prepared by centrifugal atomization process. Journal of Nuclear Materials, 1999, 265, 38-43.	1.3	5
66	Cosmos: A computer code to analyze LWR UO <sub>2</sub> and MOX fuel up to high burnup. Annals of Nuclear Energy, 1999, 26, 47-67.	0.9	24
67	Modeling of the Rim Effect on Thermal Behavior of High-Burnup UO <sub>2</sub> Fuel. Nuclear Technology, 1999, 127, 151-159.	0.7	4
68	FREE VIBRATION OF CLAMPED-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
69	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
70	A comparative analysis of UO <sub>2</sub> and MOX fuel behavior under reactivity initiated accident. Annals of Nuclear Energy, 1997, 24, 859-870.	0.9	7
71	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
72	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

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73	Effects of sintering processes on the duplex grain structure of UO <sub>2</sub> . Journal of Nuclear Materials, 1993, 200, 41-49.	1.3	9
74	Yield Stress Determination by the Massachusetts Institute of Technology Miniaturized Disk Bend Test. Nuclear Technology, 1990, 92, 383-388.	0.7	0
75	Miniaturized disk bend tests of neutron irradiated path a type alloys. Journal of Nuclear Materials, 1984, 122, 146-151.	1.3	4