Gary S Was

List of Publications by Year
in descending order

215 papers	8,434 citations	61984 43 h-index	56724 83 g-index
342 all docs	342 docs citations	342 times ranked	4405 citing authors

#	Article	IF	Citations
1	Materials challenges in nuclear energy. Acta Materialia, 2013, 61, 735-758.	7.9	1,711
2	The relationship between hardness and yield stress in irradiated austenitic and ferritic steels. Journal of Nuclear Materials, 2005, 336, 267-278.	2.7	373
3	Challenges to the use of ion irradiation for emulating reactor irradiation. Journal of Materials Research, 2015, 30, 1158-1182.	2.6	176
4	Novel features of radiation-induced segregation and radiation-induced precipitation in austenitic stainless steels. Acta Materialia, 2011, 59, 1220-1238.	7.9	162
5	Fundamentals of Radiation Materials Science. , 2017, , .		162
6	Oxidation of ferriticâ€“martensitic alloys T91, HCM12A and HT-9 in supercritical water. Journal of Nuclear Materials, 2007, 371, 1-17.	2.7	154
7	Microstructural and microchemical mechanisms controlling intergranular stress corrosion cracking in light-water-reactor systems. Journal of Nuclear Materials, 1994, 216, 348-363.	2.7	152
8	The Role of grain boundary misorientation in intergranular cracking of Ni-16Cr-9Fe in 360 Â°C argon and high-Purity water. Metallurgical and Materials Transactions A - Physical Metallurgy and Materials Science, 1992, 23, 1195-1206.	1.4	142
9	Materials for future nuclear energy systems. Journal of Nuclear Materials, 2019, 527, 151837.	2.7	142
10	Zirconium alloys for supercritical water reactor applications: Challenges and possibilities. Journal of Nuclear Materials, 2007, 371, 61-75.	2.7	137
11	Assessment of radiation-induced segregation mechanisms in austenitic and ferriticâ€“martensitic alloys. Journal of Nuclear Materials, 2011, 411, 41-50.	2.7	119
12	Combined effect of special grain boundaries and grain boundary carbides on IGSCC of Niâ€“16Crâ€“9Feâ€“xC alloys. Materials Science & Engineering A: Structural Materials: Properties, Microstructure and Processing, 2001, 300, 94-104.	5.6	107
13	Ion beam modification of metals: Compositional and microstructural changes. Progress in Surface Science, 1989, 32, 211-332.	8.3	106
14	Materials Challenges for Generation IV Nuclear Energy Systems. Nuclear Technology, 2008, 162, 342-357.	1.2	105
15	Impact of localized deformation on IASCC in austenitic stainless steels. Journal of Nuclear Materials, 2011, 408, 246-256.	2.7	102
16	Strain localization at dislocation channelâ€“grain boundary intersections in irradiated stainless steel. International Journal of Plasticity, 2014, 56, 219-231.	8.8	96
17	Thermal-spike treatment of ion-induced grain growth: Theory and experimental comparison. Physical Review B, 1993, 47, 2983-2994.	3.2	90
18	Radiation damage and irradiation-assisted stress corrosion cracking of additively manufactured 316L stainless steels. Journal of Nuclear Materials, 2019, 513, 33-44.	2.7	89

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19	Performance of ironâ€“chromiumâ€“aluminum alloy surface coatings on Zircaloy 2 under high-temperature steam and normal BWR operating conditions. Journal of Nuclear Materials, 2016, 470, 327-338.	2.7	87
20	Towards an integrated materials characterization toolbox. Journal of Materials Research, 2011, 26, 1341-1383.	2.6	84
21	Effect of pre-implanted helium on void swelling evolution in self-ion irradiated HT9. Journal of Nuclear Materials, 2015, 462, 458-469.	2.7	77
22	Void swelling and microstructure evolution at very high damage level in self-ion irradiated ferritic-martensitic steels. Journal of Nuclear Materials, 2016, 480, 159-176.	2.7	77
23	A systematic study of radiation-induced segregation in ferriticâ€“martensitic alloys. Journal of Nuclear Materials, 2013, 442, 7-16.	2.7	75
24	The effects of grain boundary carbide density and strain rate on the stress corrosion cracking behavior of cold rolled Alloy 690. Corrosion Science, 2015, 97, 107-114.	6.6	73
25	Relationship between localized strain and irradiation assisted stress corrosion cracking in an austenitic alloy. Materials Science & Engineering A: Structural Materials: Properties, Microstructure and Processing, 2011, 528, 3730-3740.	5.6	72
26	Segregation behavior in proton- and heavy-ion-irradiated ferriticâ€“martensitic alloys. Acta Materialia, 2011, 59, 4467-4481.	7.9	72
27	The mechanism of radiation-induced segregation in ferriticâ€“martensitic alloys. Acta Materialia, 2014, 65, 42-55.	7.9	72
28	The effect of grain boundary character distribution on the high temperature deformation behavior of Niâ€“16Crâ€“9Fe alloys. Acta Materialia, 2003, 51, 3831-3848.	7.9	70
29	Effect of irradiation mode on the microstructure of self-ion irradiated ferritic-martensitic alloys. Journal of Nuclear Materials, 2015, 465, 116-126.	2.7	70
30	Selective Internal Oxidation as a Mechanism for Intergranular Stress Corrosion Cracking of Ni-Cr-Fe Alloys. Metallurgical and Materials Transactions A: Physical Metallurgy and Materials Science, 2007, 38, 1244-1259.	2.2	67
31	Insights into the stress corrosion cracking of solution annealed alloy 690 in simulated pressurized water reactor primary water under dynamic straining. Acta Materialia, 2018, 151, 321-333.	7.9	66
32	Phase stability in proton and heavy ion irradiated ferriticâ€“martensitic alloys. Journal of Nuclear Materials, 2011, 419, 52-62.	2.7	63
33	Mechanism of dislocation channel-induced irradiation assisted stress corrosion crack initiation in austenitic stainless steel. Current Opinion in Solid State and Materials Science, 2015, 19, 305-314.	11.5	62
34	Characterization of ion beam irradiated 304 stainless steel utilizing nanoindentation and Laue microdiffraction. Journal of Nuclear Materials, 2015, 458, 70-76.	2.7	61
35	Quantitative linkage between the stress at dislocation channel â€“ Grain boundary interaction sites and irradiation assisted stress corrosion crack initiation. Acta Materialia, 2019, 170, 166-175.	7.9	59
36	Radiation-induced segregation and phase stability in ferriticâ€“martensitic alloy T 91. Journal of Nuclear Materials, 2011, 417, 140-144.	2.7	58

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37	Stress corrosion crack initiation in Alloy 690 in high temperature water. Current Opinion in Solid State and Materials Science, 2018, 22, 16-25.	11.5	55
38	A model for the normal stress dependence of intergranular cracking of irradiated 316L stainless steel in supercritical water. Journal of Nuclear Materials, 2011, 408, 142-152.	2.7	54
39	Application of molecular dynamics to the study of hydrogen embrittlement in Ni-Cr-Fe alloys. Physical Review B, 1989, 40, 10322-10336.	3.2	51
40	The effect of grain boundary structure on the intergranular degradation behavior of solution annealed alloy 690 in high temperature, hydrogenated water. Acta Materialia, 2020, 182, 120-130.	7.9	50
41	Proton irradiation emulation of PWR neutron damage microstructures in solution annealed 304 and cold-worked 316 stainless steels. Journal of Nuclear Materials, 2003, 323, 18-28.	2.7	47
42	Void swelling in high dose ion-irradiated reduced activation ferriticâ€“martensitic steels. Journal of Nuclear Materials, 2015, 462, 119-125.	2.7	47
43	Microstructure evolution of T91 irradiated in the BOR60 fast reactor. Journal of Nuclear Materials, 2018, 504, 122-134.	2.7	47
44	Irradiation-assisted stress corrosion cracking. Corrosion Reviews, 2011, 29, .	2.0	46
45	Irradiation-assisted stress corrosion cracking of austenitic alloys in supercritical water. Journal of Nuclear Materials, 2009, 395, 11-22.	2.7	44
46	The oxidation of alloy 690 in simulated pressurized water reactor primary water. Corrosion Science, 2017, 126, 227-237.	6.6	44
47	Irradiation-assisted stress-corrosion cracking in austenitic alloys. Jom, 1992, 44, 8-13.	1.9	43
48	Micromechanics of dislocation channeling in intergranular stress corrosion crack nucleation. Current Opinion in Solid State and Materials Science, 2012, 16, 134-142.	11.5	43
49	Application of the inverse Kirkendall model of radiation-induced segregation to ferriticâ€“martensitic alloys. Journal of Nuclear Materials, 2012, 425, 117-124.	2.7	43
50	Accelerated corrosion and oxide dissolution in 316L stainless steel irradiated in situ in high temperature water. Journal of Nuclear Materials, 2017, 493, 207-218.	2.7	42
51	Accelerated Stress Corrosion Crack Initiation of Alloys 600 and 690 in Hydrogenated Supercritical Water. Metallurgical and Materials Transactions A: Physical Metallurgy and Materials Science, 2017, 48, 1613-1628.	2.2	41
52	A thermodynamic and kinetic basis for understanding metastable phase formation during ion-beam mixing of nickel-aluminum alloys. Journal of Materials Research, 1988, 3, 626-639.	2.6	40
53	Precipitate evolution in ion-irradiated HCM12A. Journal of Nuclear Materials, 2012, 425, 105-111.	2.7	40
54	The effect of cold rolling on grain boundary structure and stress corrosion cracking susceptibility of twins in alloy 690 in simulated PWR primary water environment. Corrosion Science, 2018, 130, 126-137.	6.6	40

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55	Role of Localized Deformation in Irradiation-Assisted Stress Corrosion Cracking Initiation. Metallurgical and Materials Transactions A: Physical Metallurgy and Materials Science, 2012, 43, 136-146.	2.2	39
56	Influence of irradiation damage on slip transfer across grain boundaries. Acta Materialia, 2014, 65, 150-160.	7.9	39
57	Multiple ion beam irradiation for the study of radiation damage in materials. Nuclear Instruments & Methods in Physics Research B, 2017, 412, 1-10.	1.4	39
58	Emulation of fast reactor irradiated T91 using dual ion beam irradiation. Journal of Nuclear Materials, 2019, 527, 151831.	2.7	39
59	Strain incompatibilities and their role in intergranular cracking of irradiated 316L stainless steel. Journal of Nuclear Materials, 2013, 441, 623-632.	2.7	38
60	Crack initiation behavior of neutron irradiated model and commercial stainless steels in high temperature water. Journal of Nuclear Materials, 2014, 444, 331-341.	2.7	38
61	Void swelling in ferritic-martensitic steels under high dose ion irradiation: Exploring possible contributions to swelling resistance. Scripta Materialia, 2016, 112, 9-14.	5.2	38
62	Comparison of the microstructure, deformation and crack initiation behavior of austenitic stainless steel irradiated in-reactor or with protons. Journal of Nuclear Materials, 2015, 456, 85-98.	2.7	37
63	Characterization of microstructure and property evolution in advanced cladding and duct: Materials exposed to high dose and elevated temperature. Journal of Materials Research, 2015, 30, 1246-1274.	2.6	36
64	Radiolysis driven changes to oxide stability during irradiation-corrosion of 316L stainless steel in high temperature water. Journal of Nuclear Materials, 2017, 493, 40-52.	2.7	36
65	Improved Creep Behavior of Ferritic-Martensitic Alloy T91 by Subgrain Boundary Density Enhancement. Metallurgical and Materials Transactions A: Physical Metallurgy and Materials Science, 2008, 39, 150-164.	2.2	35
66	The co-evolution of microstructure features in self-ion irradiated HT9 at very high damage levels. Journal of Nuclear Materials, 2017, 484, 193-208.	2.7	35
67	Oxidation of Zircaloy-4 during in situ proton irradiation and corrosion in PWR primary water. Journal of Materials Research, 2015, 30, 1335-1348.	2.6	34
68	A historical perspective on understanding IASCC. Journal of Nuclear Materials, 2019, 517, 380-392.	2.7	34
69	Understanding bubble and void nucleation in dual ion irradiated T91 steel using single parameter experiments. Acta Materialia, 2020, 198, 47-60.	7.9	34
70	Predicting structural material degradation in advanced nuclear reactors with ion irradiation. Scientific Reports, 2021, 11, 2949.	3.3	34
71	Tensile and stress corrosion cracking behavior of ferriticâ€“martensitic steels in supercritical water. Journal of Nuclear Materials, 2009, 395, 30-36.	2.7	33
72	Probing long-range ordering in nickel-base alloys with proton irradiation. Acta Materialia, 2018, 156, 446-462.	7.9	33

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73	Effect of post-irradiation annealing on the irradiated microstructure of neutron-irradiated 304L stainless steel. Journal of Nuclear Materials, 2018, 500, 220-234.	2.7	32
74	Oxide growth and dissolution on 316L stainless steel during irradiation in high temperature water. Corrosion Science, 2019, 157, 305-311.	6.6	32
75	A high-resolution characterization of the initiation of stress corrosion crack in Alloy 690 in simulated pressurized water reactor primary water. Corrosion Science, 2020, 163, 108243.	6.6	32
76	Effect of ion bombardment on in-plane texture, surface morphology, and microstructure of vapor deposited Nb thin films. Journal of Applied Physics, 1997, 81, 6754-6761.	2.5	30
77	The role of dislocation channeling in IASCC initiation of neutron irradiated stainless steel. Journal of Nuclear Materials, 2016, 481, 214-225.	2.7	30
78	Characterization of alloy 718 subjected to different thermomechanical treatments. Materials Science & Engineering A: Structural Materials: Properties, Microstructure and Processing, 2017, 691, 195-202.	5.6	30
79	A facility for studying irradiation accelerated corrosion in high temperature water. Journal of Nuclear Materials, 2014, 451, 40-47.	2.7	29
80	The heatâ€ofâ€mixing effect on ionâ€induced grain growth. Journal of Applied Physics, 1991, 70, 1252-1260.	2.5	28
81	Precipitate behavior in self-ion irradiated stainless steels at high doses. Journal of Nuclear Materials, 2014, 449, 200-206.	2.7	28
82	Aspects of ion irradiations to study localized deformation in austenitic stainless steels. Journal of Nuclear Materials, 2014, 452, 328-334.	2.7	27
83	The influence of helium on cavity evolution in ion-irradiated T91. Journal of Nuclear Materials, 2018, 509, 707-721.	2.7	27
84	Radiation tolerance of commercial and advanced alloys for core internals: a comprehensive microstructural characterization. Journal of Nuclear Materials, 2018, 510, 396-413.	2.7	27
85	Irradiation assisted stress corrosion cracking of commercial and advanced alloys for light water reactor core internals. Journal of Nuclear Materials, 2019, 515, 52-70.	2.7	26
86	The effects of proton irradiation on the microstructural and mechanical property evolution of inconel X-750 with high concentrations of helium. Journal of Nuclear Materials, 2017, 492, 213-226.	2.7	25
87	The role of grain boundary microchemistry in irradiation-assisted stress corrosion cracking of a Fe-13Cr-15Ni alloy. Acta Materialia, 2017, 138, 61-71.	7.9	25
88	Effect of radiation damage and water radiolysis on corrosion of FeCrAl alloys in hydrogenated water. Journal of Nuclear Materials, 2020, 533, 152108.	2.7	24
89	High-Temperature Oxidation of Alloy 617 in Helium Containing Part-Per-Million Levels of CO and CO2 as Impurities. Metallurgical and Materials Transactions A: Physical Metallurgy and Materials Science, 2011, 42, 1245-1265.	2.2	22
90	The effects of strain rate and carbon concentration on the dynamic strain aging of cold rolled Ni-based alloy in high temperature water. Scripta Materialia, 2015, 107, 107-110.	5.2	22

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91	Self-ion emulation of high dose neutron irradiated microstructure in stainless steels. Journal of Nuclear Materials, 2018, 501, 312-318.	2.7	22
92	The effect of post-irradiation annealing on the stress corrosion crack growth rate of neutron-irradiated 304L stainless steel in boiling water reactor environment. Corrosion Science, 2019, 161, 108183.	6.6	22
93	Oxidation of Alloy 600 and Alloy 690: Experimentally Accelerated Study in Hydrogenated Supercritical Water. Metallurgical and Materials Transactions A: Physical Metallurgy and Materials Science, 2017, 48, 1596-1612.	2.2	21
94	Stress Localization Resulting from Grain Boundary Dislocation Interactions in Relaxed and Defective Grain Boundaries. Metallurgical and Materials Transactions A: Physical Metallurgy and Materials Science, 2020, 51, 667-683.	2.2	21
95	Creep and intergranular cracking of Ni-Cr-Fe-C in 360 Â°C argon. Metallurgical and Materials Transactions A: Physical Metallurgy and Materials Science, 1994, 25, 1169-1183.	2.2	20
96	The influence of carbon on cavity evolution in ion-irradiated ferritic-martensitic steels. Journal of Nuclear Materials, 2018, 509, 722-735.	2.7	20
97	In situ proton irradiation creep of ferriticâ€"martensitic steel T91. Journal of Nuclear Materials, 2013, 441, 681-687.	2.7	19
98	Proton irradiation-induced creep of ultra-fine grain graphite. Carbon, 2014, 77, 993-1010.	10.3	19
99	Methodology for determining void swelling at very high damage under ion irradiation. Journal of Nuclear Materials, 2016, 477, 273-279.	2.7	19
100	The diffusion of cesium, strontium, and europium in silicon carbide. Journal of Nuclear Materials, 2016, 476, 155-167.	2.7	19
101	Corrosion behavior of ceramic-coated ZIRLOâ„¢ exposed to supercritical water. Journal of Nuclear Materials, 2018, 498, 495-504.	2.7	19
102	Dual ion irradiation of commercial and advanced alloys: Evaluating microstructural resistance for high dose core internals. Journal of Nuclear Materials, 2019, 516, 125-134.	2.7	19
103	Grain boundary character distributions in Ni-16Cr-9Fe using selected area channeling patterns: Methodology and results. Journal of Electron Microscopy Technique, 1991, 19, 345-360.	1.1	18
104	Aluminum metallization for flat-panel displays using ion-beam-assisted physical vapor deposition. Journal of Materials Research, 1999, 14, 4051-4061.	2.6	18
105	Austenitic Stainless Steels. , 2019, , 293-347.		18
106	A prioridetermination of the sampling size for grain-boundary character distribution and grain-boundary degradation analysis. Philosophical Magazine A: Physics of Condensed Matter, Structure, Defects and Mechanical Properties, 2001, 81, 1951-1965.	0.6	17
107	Atomistic simulation of the obstacle strengths of radiation-induced defects in an Feâ€"Niâ€"Cr austenitic stainless steel. Modelling and Simulation in Materials Science and Engineering, 2019, 27, 085004.	2.0	17
108	Roadmap for the application of ion beam technologies to the challenges of nuclear energy technologies. Nuclear Instruments & Methods in Physics Research B, 2019, 441, 41-45.	1.4	17

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109	Anisotropy analysis of ultra-fine grain graphite and pyrolytic carbon. Carbon, 2013, 60, 410-420.	10.3	15
110	Micromechanistic origin of irradiation-assisted stress corrosion cracking. Philosophical Magazine, 2014, 94, 4197-4218.	1.6	15
111	Development of a multi-layer diffusion couple to study fission product transport in Î²-SiC. Journal of Nuclear Materials, 2014, 444, 170-174.	2.7	15
112	Radiation enhanced diffusion of cesium, strontium, and europium in silicon carbide. Journal of Nuclear Materials, 2016, 474, 76-87.	2.7	15
113	IASCC of neutron irradiated 316 stainless steel to 125 dpa. Materials Characterization, 2021, 173, 110897.	4.4	15
114	Dynamic Strain Aging of Nickel-Base Alloys 800H and 690. Metallurgical and Materials Transactions A: Physical Metallurgy and Materials Science, 2012, 43, 3428-3432.	2.2	14
115	TEM/STEM study of Zircaloy-2 with protective FeAl(Cr) layers under simulated BWR environment and high-temperature steam exposure. Journal of Nuclear Materials, 2018, 502, 95-105.	2.7	14
116	Corrosion behavior of ferritic FeCrAl alloys in simulated BWR normal water chemistry. Journal of Nuclear Materials, 2021, 545, 152744.	2.7	14
117	A high-resolution characterization of irradiation-assisted stress corrosion cracking of proton-irradiated 316L stainless steel in simulated pressurized water reactor primary water. Corrosion Science, 2022, 199, 110187.	6.6	14
118	Phase formation in ionâ€irradiated and annealed Niâ€rich Niâ€Al thin films. Journal of Applied Physics, 1991, 69, 2021-2028.	2.5	13
119	RADIATION DAMAGE FROM DIFFERENT PARTICLE TYPES. , 2007, , 65-98.		13
120	Crack initiation of neutron-irradiated 304ÂL stainless steel in PWR primary water. Corrosion Science, 2021, 193, 109902.	6.6	13
121	Elastic strain energy control of the precipitate free zone around primary carbides in nickel base alloy 725. Acta Materialia, 2016, 120, 138-149.	7.9	12
122	Surface mechanical properties of aluminum implanted nickel and co-evaporated Niâ€"Al on nickel. Journal of Materials Research, 1990, 5, 1668-1683.	2.6	11
123	Chromia-Assisted Decarburization of W-Rich Ni-Based Alloys in Impure Helium at 1273ÂK (1000ÂÂ°C). Metallurgical and Materials Transactions A: Physical Metallurgy and Materials Science, 2011, 42, 1229-1244.	2.2	11
124	Mechanism of Internal Oxidation of Alloy 617 in He-CO-CO2 Environments at 1123ÂK (850ÂÂ°C). Metallurgical and Materials Transactions A: Physical Metallurgy and Materials Science, 2015, 46, 525-535.	2.2	11
125	Microstructural characterization of cold-worked 316 stainless steel flux thimble tubes irradiated up to 100 dpa in a commercial Pressurized Water Reactor. Journal of Nuclear Materials, 2020, 541, 152400.	2.7	11
126	Toward the Use of Ion Irradiation to Predict Reactor Irradiation Effects. , 2020, , 468-484.		11

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127	Formation of buried TiN in glass by ion implantation to reduce solar load. Journal of Applied Physics, 1996, 80, 2768-2773.	2.5	10
128	Engineered Coatings for Ni Alloys in High Temperature Reactors. Metallurgical and Materials Transactions A: Physical Metallurgy and Materials Science, 2013, 44, 835-847.	2.2	10
129	Determination of dose rate effects on Zircaloy oxidation using proton irradiation and oxygen transport modeling. Journal of Nuclear Materials, 2019, 523, 56-65.	2.7	10
130	Synergies between H, He and radiation damage in dual and triple ion irradiation of candidate fusion blanket materials. Journal of Nuclear Materials, 2022, 565, 153722.	2.7	10
131	Proton irradiation creep of beta-silicon carbide. Journal of Nuclear Materials, 2011, 418, 198-206.	2.7	9
132	Proton irradiation creep of FM steel T91. Journal of Nuclear Materials, 2015, 459, 183-193.	2.7	9
133	Surface oxidation of Alloy 617 in low oxygen partial pressure Heâ€“COâ€“CO2 environments at 750â€“850 Â°C. Corrosion Science, 2015, 90, 529-534.	6.6	9
134	Characterization of M2X formed during 5Â MeV Fe2+ irradiation. Journal of Nuclear Materials, 2017, 485, 154-158.	2.7	9
135	Effects of post-irradiation annealing on the IASCC susceptibility of neutron-irradiated 304L stainless steel. Journal of Nuclear Materials, 2019, 526, 151755.	2.7	9
136	The roles of thermal mechanical treatment and Î´ phase in the stress corrosion cracking of alloy 718 in primary water. Corrosion Science, 2019, 160, 108168.	6.6	9
137	Investigation of Rare Earth-Containing Double Phosphates of the Type A₃Ln(PO₄)₂ (Ln = Y, La, Pr, Nd, and Smâ€“Lu) as Potential Nuclear Waste Forms. Chemistry of Materials, 2022, 34, 3819-3830.	6.7	9
138	Anisotropic dislocation loop distribution in alloy T91 during irradiation creep. Journal of Nuclear Materials, 2014, 454, 255-264.	2.7	8
139	Insights into the sources of irradiation hardening in a neutron irradiated 304L stainless steel following post-irradiation annealing. Journal of Nuclear Materials, 2019, 526, 151754.	2.7	8
140	Emulation of neutron damage with proton irradiation and its effects on microstructure and microchemistry of Zircaloy-4. Journal of Nuclear Materials, 2021, 557, 153281.	2.7	8
141	The effect of potassium hydroxide primary water chemistry on the IASCC behavior of 304 stainless steel. Journal of Nuclear Materials, 2022, 558, 153323.	2.7	8
142	Insights into the roles of intergranular carbides in the initiation of intergranular stress corrosion cracking of alloy 690 in simulated PWR primary water. Corrosion Science, 2022, 196, 110048.	6.6	8
143	Crystallization of A₃Ln(BO₃)₂ (A = Na, K; Ln = Lanthanide) from a Boric Acid Containing Hydroxide Melt: Synthesis and Investigation of Lanthanide Borates as Potential Nuclear Waste Forms. Inorganic Chemistry, 2022, 61, 11232-11242.	4.0	7
144	Application of a thermal spike model to experimental ion-induced grain growth data. Surface and Coatings Technology, 1992, 51, 333-337.	4.8	6

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145	Application of adjoint sensitivity analysis to nuclear reactor fuel rod performance. Nuclear Engineering and Design, 1984, 80, 27-38.	1.7	5
146	Hardness of ion-implanted Ni3Al and TiAl. Journal of Materials Research, 1991, 6, 1615-1618.	2.6	5
147	Lattice location and hardness of Ta-implanted Ni3Al. Journal of Materials Research, 1991, 6, 1200-1206.	2.6	5
148	A facility for conducting high-temperature oxidation experiments of alloys in helium environments containing part per million levels of impurities. Measurement Science and Technology, 2009, 20, 095708.	2.6	5
149	Emulating Neutron Irradiation Effects with Ions. , 2017, , 631-665.		5
150	In situ proton irradiation-induced creep at very high temperature. Journal of Nuclear Materials, 2013, 433, 86-94.	2.7	4
151	Evolution dependence of vanadium nitride nanoprecipitates on directionality of ion irradiation. Journal of Nuclear Materials, 2017, 495, 425-430.	2.7	4
152	Corrosion Issues in Current and Next-Generation Nuclear Reactors. , 2019, , 211-246.		4
153	An Investigation of Alloy 182 Stress Corrosion Cracking in Simulated PWR Environment. , 0, , 279-296.		4
154	Reproducing shadow corrosion on Zircaloy-2 using in-situ proton irradiation. Journal of Nuclear Materials, 2022, 558, 153406.	2.7	4
155	Adjoint sensitivity analysis in nuclear reactor fuel behavior modeling. Nuclear Engineering and Design, 1981, 66, 125-139.	1.7	3
156	Thermal Spike Model of Ion-Induced Grain Growth. Materials Research Society Symposia Proceedings, 1990, 202, 205.	0.1	3
157	Modification of Fracture Energy of Niobium/sapphire Interface By Impurity Doping. Materials Research Society Symposia Proceedings, 1996, 458, 191.	0.1	3
158	Mechanism of Decarburization of Alloy 617 at 1000Â°C in Helium Containing CO and CO2 as impurities. Materials Research Society Symposia Proceedings, 2008, 1125, 1.	0.1	3
159	Enhanced Oxygen Diffusion Within the Internal Oxidation Zone of Alloy 617 in Controlled Impurity Helium Environments from 1023ÂK to 1123ÂK (750ÂÂ°C to 850ÂÂ°C). Metallurgical and Materials Transactions A: Physical Metallurgy and Materials Science, 2015, 46, 1628-1638.	2.2	3
160	A methodology for customizing implantation profiles of light ions using a single thin foil energy degrader. Nuclear Instruments & Methods in Physics Research B, 2020, 478, 274-283.	1.4	3
161	A microscopic and crystallographic study of proton irradiated alloy 718. Journal of Nuclear Materials, 2021, 551, 152954.	2.7	3
162	Oxidation of a Proton-Irradiated 316 Stainless Steel in Simulated BWR NWC Environment. , 2011, , 1329-1338.		3

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163	Mechanisms behind irradiation-assisted stress corrosion cracking. , 2020, , 47-88.		3
164	In-situ microstructure observation of oxidized SiC layer in surrogate TRISO fuel particles under krypton ion irradiation. Journal of Alloys and Compounds, 2022, 920, 165833.	5.5	3
165	FCODE-BETA/SS: A Fuel Performance Code for Stainless Steel Clad Pressurized Water Reactor Fuel. Nuclear Technology, 1985, 69, 198-209.	1.2	2
166	Degradation modes of austenitic and ferriticâ€“martensitic stainless steels in Heâ€“COâ€“CO2 and liquid sodium environments of equivalent oxygen and carbon chemical potentials. Journal of Nuclear Materials, 2013, 441, 633-643.	2.7	2
167	A Multi-Pinhole Faraday Cup Device for Measurement of Discrete Charge Distribution of Heavy and Light Ions. IEEE Transactions on Nuclear Science, 2016, 63, 854-860.	2.0	2
168	Corrosion Potential Monitoring in Swedish BWRS on Hydrogen Water Chemistry. , 0, , 453-460.		2
169	Evidence for the Reduction of Sulfates Under Representative SG Secondary Side Conditions, and for the Role of Reduced Sulfates on Alloy 600 Tubing Degradation. , 0, , 567-575.		2
170	The Effect of Primary Coolant Zinc Additions on the SCC Behaviour of Alloy 600 and 690. , 0, , 96-103.		2
171	In-Situ Proton Irradiation-Corrosion Study of ATF Candidate Alloys in Simulated PWR Primary Water. Minerals, Metals and Materials Series, 2019, , 1461-1474.	0.4	2
172	Solute segregation and precipitation across damage rates in dual-ionâ€“irradiated T91 steel. Journal of Nuclear Materials, 2022, 563, 153626.	2.7	2
173	Cogenerationâ€"An energy alternative for the U.S.?. Energy, 1979, 4, 1023-1031.	8.8	1
174	Statistical Patterns of Fuel Failure in Stainless Steel Clad Light Water Reactor Fuel Rods. Nuclear Technology, 1985, 71, 445-457.	1.2	1
175	A computational framework for modeling irradiation creep and swelling in single crystal nickel. Journal of Nuclear Materials, 1986, 139, 137-150.	2.7	1
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