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	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
2	Reproducing shadow corrosion on Zircaloy-2 using in-situ proton irradiation. Journal of Nuclear Materials, 2022, 558, 153406.	2.7	4
3	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
4	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
5	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
6	Solute segregation and precipitation across damage rates in dual-ionâ€“irradiated T91 steel. Journal of Nuclear Materials, 2022, 563, 153626.	2.7	2
7	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
8	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
9	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
10	Corrosion behavior of ferritic FeCrAl alloys in simulated BWR normal water chemistry. Journal of Nuclear Materials, 2021, 545, 152744.	2.7	14
11	Predicting structural material degradation in advanced nuclear reactors with ion irradiation. Scientific Reports, 2021, 11, 2949.	3.3	34
12	IASCC of neutron irradiated 316 stainless steel to 125 dpa. Materials Characterization, 2021, 173, 110897.	4.4	15
13	A microscopic and crystallographic study of proton irradiated alloy 718. Journal of Nuclear Materials, 2021, 551, 152954.	2.7	3
14	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
15	Crack initiation of neutron-irradiated 304ÂL stainless steel in PWR primary water. Corrosion Science, 2021, 193, 109902.	6.6	13
16	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
17	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
18	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

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19	Gary Was and Todd Allen Reply. New Labor Forum, 2020, 29, 24-26.	0.4	0
20	Understanding bubble and void nucleation in dual ion irradiated T91 steel using single parameter experiments. Acta Materialia, 2020, 198, 47-60.	7.9	34
21	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
22	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
23	The Importance of Nuclear Power in Our Energy Mix. New Labor Forum, 2020, 29, 14-18.	0.4	1
24	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
25	Toward the Use of Ion Irradiation to Predict Reactor Irradiation Effects. , 2020, , 468-484.		11
26	Mechanisms behind irradiation-assisted stress corrosion cracking. , 2020, , 47-88.		3
27	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
28	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
29	Emulation of fast reactor irradiated T91 using dual ion beam irradiation. Journal of Nuclear Materials, 2019, 527, 151831.	2.7	39
30	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
31	Corrosion Issues in Current and Next-Generation Nuclear Reactors. , 2019, , 211-246.		4
32	Austenitic Stainless Steels. , 2019, , 293-347.		18
33	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
34	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
35	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
36	Oxide growth and dissolution on 316L stainless steel during irradiation in high temperature water. Corrosion Science, 2019, 157, 305-311.	6.6	32

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37	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
38	A historical perspective on understanding IASCC. Journal of Nuclear Materials, 2019, 517, 380-392.	2.7	34
39	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
40	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
41	Microstructural Study on the Stress Corrosion Cracking of Alloy 690 in Simulated Pressurized Water Reactor Primary Environment. Minerals, Metals and Materials Series, 2019, , 535-545.	0.4	1
42	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
43	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
44	Materials for future nuclear energy systems. Journal of Nuclear Materials, 2019, 527, 151837.	2.7	142
45	Solute Clustering in As-irradiated and Post-irradiation-Annealed 304 Stainless Steel. Minerals, Metals and Materials Series, 2019, , 2189-2207.	0.4	0
46	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
47	Irradiation Assisted Stress Corrosion Cracking (IASCC) of Nickel-Base Alloys in Light Water Reactors Environments Part II: Stress Corrosion Cracking. Minerals, Metals and Materials Series, 2019, , 2177-2188.	0.4	0
48	Radiation-Induced Precipitates in a Self-ion Irradiated Cold-Worked 316 Austenitic Stainless Steel Used for PWR Baffle-Bolts. Minerals, Metals and Materials Series, 2019, , 565-580.	0.4	0
49	Corrosion of Multilayer Ceramic-Coated ZIRLO Exposed to High Temperature Water. Minerals, Metals and Materials Series, 2019, , 1497-1508.	0.4	0
50	Effect of Grain Orientation on Irradiation Assisted Corrosion of 316L Stainless Steel in Simulated PWR Primary Water. Minerals, Metals and Materials Series, 2019, , 2303-2312.	0.4	0
51	IASCC Susceptibility of 304L Stainless Steel Irradiated in a BWR and Subjected to Post Irradiation Annealing. Minerals, Metals and Materials Series, 2019, , 2231-2242.	0.4	0
52	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
53	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
54	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

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55	Self-ion emulation of high dose neutron irradiated microstructure in stainless steels. Journal of Nuclear Materials, 2018, 501, 312-318.	2.7	22
56	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
57	Microstructure evolution of T91 irradiated in the BOR60 fast reactor. Journal of Nuclear Materials, 2018, 504, 122-134.	2.7	47
58	Stress Corrosion Cracking Behavior of Alloy 718 Subjected to Various Thermal Mechanical Treatments in Primary Water. Minerals, Metals and Materials Series, 2018, , 293-305.	0.4	0
59	Microstructural Study on the Stress Corrosion Cracking of Alloy 690 in Simulated Pressurized Water Reactor Primary Environment. Minerals, Metals and Materials Series, 2018, , 535-545.	0.4	1
60	Radiation-Induced Precipitates in a Self-ion Irradiated Cold-Worked 316 Austenitic Stainless Steel Used for PWR Baffle-Bolts. Minerals, Metals and Materials Series, 2018, , 565-580.	0.4	0
61	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
62	Corrosion behavior of ceramic-coated ZIRLOâ„¢ exposed to supercritical water. Journal of Nuclear Materials, 2018, 498, 495-504.	2.7	19
63	The influence of carbon on cavity evolution in ion-irradiated ferritic-martensitic steels. Journal of Nuclear Materials, 2018, 509, 722-735.	2.7	20
64	The influence of helium on cavity evolution in ion-irradiated T91. Journal of Nuclear Materials, 2018, 509, 707-721.	2.7	27
65	Probing long-range ordering in nickel-base alloys with proton irradiation. Acta Materialia, 2018, 156, 446-462.	7.9	33
66	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
67	In-Situ Proton Irradiation-Corrosion Study of ATF Candidate Alloys in Simulated PWR Primary Water. Minerals, Metals and Materials Series, 2018, , 245-258.	0.4	0
68	Corrosion of Multilayer Ceramic-Coated ZIRLO Exposed to High Temperature Water. Minerals, Metals and Materials Series, 2018, , 281-292.	0.4	0
69	Irradiation Assisted Stress Corrosion Cracking (IASCC) of Nickel-Base Alloys in Light Water Reactors Environments Part II: Stress Corrosion Cracking. Minerals, Metals and Materials Series, 2018, , 961-972.	0.4	0
70	Solute Clustering in As-irradiated and Post-irradiation-Annealed 304 Stainless Steel. Minerals, Metals and Materials Series, 2018, , 973-991.	0.4	0
71	IASCC Susceptibility of 304L Stainless Steel Irradiated in a BWR and Subjected to Post Irradiation Annealing. Minerals, Metals and Materials Series, 2018, , 1015-1026.	0.4	0
72	Effect of Grain Orientation on Irradiation Assisted Corrosion of 316L Stainless Steel in Simulated PWR Primary Water. Minerals, Metals and Materials Series, 2018, , 1087-1096.	0.4	0

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73	Fundamentals of Radiation Materials Science. , 2017, , .		162
74	Characterization of M2X formed during 5ÂMeV Fe2+ irradiation. Journal of Nuclear Materials, 2017, 485, 154-158.	2.7	9
75	Top reviewers for the Journal of Nuclear Materials 2016. Journal of Nuclear Materials, 2017, 483, v.	2.7	0
76	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
77	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
78	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
79	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
80	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
81	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
82	Evolution dependence of vanadium nitride nanoprecipitates on directionality of ion irradiation. Journal of Nuclear Materials, 2017, 495, 425-430.	2.7	4
83	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
84	The oxidation of alloy 690 in simulated pressurized water reactor primary water. Corrosion Science, 2017, 126, 227-237.	6.6	44
85	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
86	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
87	Assessment of Corrosion Resistance of Candidate Alloys for Accident Tolerant Fuel Cladding under Reactor Conditions. Microscopy and Microanalysis, 2017, 23, 2226-2227.	0.4	0
88	Emulating Neutron Irradiation Effects with Ions. , 2017, , 631-665.		5
89	Radiation-Induced Segregation. , 2017, , 255-299.		0
90	Effects of Irradiation on Corrosion and Environmentally Assisted Cracking. , 2017, , 951-985.		0

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91	Methodology for determining void swelling at very high damage under ion irradiation. Journal of Nuclear Materials, 2016, 477, 273-279.	2.7	19
92	The role of dislocation channeling in IASCC initiation of neutron irradiated stainless steel. Journal of Nuclear Materials, 2016, 481, 214-225.	2.7	30
93	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
94	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
95	The diffusion of cesium, strontium, and europium in silicon carbide. Journal of Nuclear Materials, 2016, 476, 155-167.	2.7	19
96	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
97	Radiation enhanced diffusion of cesium, strontium, and europium in silicon carbide. Journal of Nuclear Materials, 2016, 474, 76-87.	2.7	15
98	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
99	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
100	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
101	Materials hurdles for advanced nuclear reactors. MRS Bulletin, 2015, 40, 554-555.	3.5	1
102	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
103	Challenges to the use of ion irradiation for emulating reactor irradiation. Journal of Materials Research, 2015, 30, 1158-1182.	2.6	176
104	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
105	Void swelling in high dose ion-irradiated reduced activation ferriticâ€"martensitic steels. Journal of Nuclear Materials, 2015, 462, 119-125.	2.7	47
106	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
107	Proton irradiation creep of FM steel T91. Journal of Nuclear Materials, 2015, 459, 183-193.	2.7	9
108	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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109	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
110	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
111	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
112	Stress corrosion cracking of ferritic-maretensitic steels in simulated boiling water reactor environment. Corrosion, 2015, , .	1.1	0
113	Characterization of ion beam irradiated 304 stainless steel utilizing nanoindentation and Laue microdiffraction. Journal of Nuclear Materials, 2015, 458, 70-76.	2.7	61
114	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
115	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
116	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
117	Anisotropic dislocation loop distribution in alloy T91 during irradiation creep. Journal of Nuclear Materials, 2014, 454, 255-264.	2.7	8
118	Micromechanistic origin of irradiation-assisted stress corrosion cracking. Philosophical Magazine, 2014, 94, 4197-4218.	1.6	15
119	Strain localization at dislocation channelâ€"grain boundary intersections in irradiated stainless steel. International Journal of Plasticity, 2014, 56, 219-231.	8.8	96
120	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
121	A facility for studying irradiation accelerated corrosion in high temperature water. Journal of Nuclear Materials, 2014, 451, 40-47.	2.7	29
122	The mechanism of radiation-induced segregation in ferriticâ€"martensitic alloys. Acta Materialia, 2014, 65, 42-55.	7.9	72
123	Influence of irradiation damage on slip transfer across grain boundaries. Acta Materialia, 2014, 65, 150-160.	7.9	39
124	Proton irradiation-induced creep of ultra-fine grain graphite. Carbon, 2014, 77, 993-1010.	10.3	19
125	Aspects of ion irradiations to study localized deformation in austenitic stainless steels. Journal of Nuclear Materials, 2014, 452, 328-334.	2.7	27
126	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

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127	Precipitate behavior in self-ion irradiated stainless steels at high doses. Journal of Nuclear Materials, 2014, 449, 200-206.	2.7	28
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	In situ proton irradiation-induced creep at very high temperature. Journal of Nuclear Materials, 2013, 433, 86-94.	2.7	4
130	Anisotropy analysis of ultra-fine grain graphite and pyrolytic carbon. Carbon, 2013, 60, 410-420.	10.3	15
131	A systematic study of radiation-induced segregation in ferriticâ€"martensitic alloys. Journal of Nuclear Materials, 2013, 442, 7-16.	2.7	75
132	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
133	Strain incompatibilities and their role in intergranular cracking of irradiated 316L stainless steel. Journal of Nuclear Materials, 2013, 441, 623-632.	2.7	38
134	In situ proton irradiation creep of ferriticâ€"martensitic steel T91. Journal of Nuclear Materials, 2013, 441, 681-687.	2.7	19
135	Materials challenges in nuclear energy. Acta Materialia, 2013, 61, 735-758.	7.9	1,711
136	Capabilities for Conducting Crack Growth Test of Neutron-Irradiated Alloys in Light Water Reactor Environments. Corrosion, 2013, 69, 136-144.	1.1	0
137	Microstructural Evolution of Self-Ion Irradiation HT9. , 2013, , .		0
138	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
139	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
140	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
141	Precipitate evolution in ion-irradiated HCM12A. Journal of Nuclear Materials, 2012, 425, 105-111.	2.7	40
142	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
143	Radiation-induced segregation and phase stability in ferriticâ€"martensitic alloy T 91. Journal of Nuclear Materials, 2011, 417, 140-144.	2.7	58
144	Proton irradiation creep of beta-silicon carbide. Journal of Nuclear Materials, 2011, 418, 198-206.	2.7	9

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145	Phase stability in proton and heavy ion irradiated ferriticâ€"martensitic alloys. Journal of Nuclear Materials, 2011, 419, 52-62.	2.7	63
146	Assessment of radiation-induced segregation mechanisms in austenitic and ferriticâ€"martensitic alloys. Journal of Nuclear Materials, 2011, 411, 41-50.	2.7	119
147	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
148	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
149	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
150	Novel features of radiation-induced segregation and radiation-induced precipitation in austenitic stainless steels. Acta Materialia, 2011, 59, 1220-1238.	7.9	162
151	Segregation behavior in proton- and heavy-ion-irradiated ferriticâ€"martensitic alloys. Acta Materialia, 2011, 59, 4467-4481.	7.9	72
152	Impact of localized deformation on IASCC in austenitic stainless steels. Journal of Nuclear Materials, 2011, 408, 246-256.	2.7	102
153	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
154	Towards an integrated materials characterization toolbox. Journal of Materials Research, 2011, 26, 1341-1383.	2.6	84
155	Irradiation-assisted stress corrosion cracking. Corrosion Reviews, 2011, 29, .	2.0	46
156	Oxidation of a Proton-Irradiated 316 Stainless Steel in Simulated BWR NWC Environment. , 2011, , 1329-1338.		3
157	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
158	Irradiation-assisted stress corrosion cracking of austenitic alloys in supercritical water. Journal of Nuclear Materials, 2009, 395, 11-22.	2.7	44
159	Tensile and stress corrosion cracking behavior of ferriticâ€"martensitic steels in supercritical water. Journal of Nuclear Materials, 2009, 395, 30-36.	2.7	33
160	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
161	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
162	Materials Challenges for Generation IV Nuclear Energy Systems. Nuclear Technology, 2008, 162, 342-357.	1.2	105

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163	Oxidation of ferriticâ€“martensitic alloys T91, HCM12A and HT-9 in supercritical water. Journal of Nuclear Materials, 2007, 371, 1-17.	2.7	154
164	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
165	Zirconium alloys for supercritical water reactor applications: Challenges and possibilities. Journal of Nuclear Materials, 2007, 371, 61-75.	2.7	137
166	RADIATION DAMAGE FROM DIFFERENT PARTICLE TYPES. , 2007, , 65-98.		13
167	The relationship between hardness and yield stress in irradiated austenitic and ferritic steels. Journal of Nuclear Materials, 2005, 336, 267-278.	2.7	373
168	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
169	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
170	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
171	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
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