ELSEVIER

Journal of Nuclear Materials 258–263 (1998) 1669–1674

journal of nuclear materials

Effect of irradiation temperature on irradiation assisted stress corrosion cracking of model austenitic stainless steels

T. Tsukada *, Y. Miwa, H. Tsuji, H. Nakajima

Department of Materials Science and Engineering, Material performance and Testing Laboratory, Japan Atomic Energy Research Institute, Tokai-mura, Naka-gun, Ibaraki-ken 319-11, Japan

Abstract

Effects of irradiation temperature on the irradiation assisted stress corrosion cracking (IASCC) were studied on a high-purity type 304 model stainless steel (HP) and its alloy doped with carbon (HP + C). Solution-annealed and cold-worked specimens of the alloys were irradiated at 513 and 323 K up to 7-9 $\times$ 10^{24} n/m^2 (E > 1 MeV) and then susceptibilities to IASCC were examined by the slow strain rate testing (SSRT) in high-purity 573 K water. Solution-annealed specimens of both alloys, HP and HP + C, irradiated at 323 K showed a susceptibility to intergranular stress corrosion cracking (IGSCC). The specimen of alloy HP + C failed at a shoulder portion with very high susceptibility to IGSCC. Addition of carbon affected the fracture morphology and increased radiation hardening by irradiation at 513 K, while it seems to increase uniform elongation of specimens irradiated at 323 K. The 50% cold-worked specimens of both alloys showed no susceptibility to IASCC.

1. Introduction

The stress corrosion cracking (SCC) has been regarded as a critical issue for the material performance in light water reactors (LWRs) for more than 30 years. The irradiation assisted stress corrosion cracking (IASCC) is recently concerned as one of the possible degradation phenomena of the structural materials for the water-cooled components of ITER [1]. IASCC has been investigated in the last decade mainly from a view point of the assessment and extension of lives of LWRs [2–4]. However, a temperature range of concern for ITER design is lower than that for LWRs which is below about 523 K and the experimental results on IASCC in these lower temperatures are very limited. Temperature dependence of IASCC susceptibility was studied by the authors [5] on type 316 stainless steels (SSs) irradiated under the spectrally tailored neutron condition where specimens irradiated at 333 K had no susceptibility to SCC in oxygenated 333 K water. However, Kodama et al. [6] reported that type 304 SS irradiated at 323 K in the Japan Materials Testing Reactor (JMTR) was susceptible to SCC in 561 K high-temperature water. It is believed that a primary process of IASCC is a radiation induced segregation (RIS) and a consequent depletion of chromium at grain boundaries [4,7]. According to theoretical predictions [8,9], chromium depletion due to RIS is modified with the irradiation temperature, so that it is worth to study the effects of irradiation temperature on IASCC behavior from both viewpoints of understanding its mechanism and fusion reactor design. The other concerns about IASCC are the effects of cold-working and addition of carbon. Those are usual methods to improve the strength of materials and known as the effective ways to diminish void swelling [10]. We had found prominent effects of carbon addition into model SSs on the IASCC fracture morphology and radiation hardening [11] and on the microstructural evolution by neutron irradiation [12]. Therefore it is also meaningful to examine the effects of cold working and carbon addition. This paper presents results of post-irradiation examinations on the model austenitic stainless steels performed to study these effects on IASCC behavior.

* Corresponding author. Tel.: +81 29 282 6082; fax: +81 29 282 5922; e-mail: ttsukada@popsvr.tokai.jaeri.go.jp.

0022-3115/98/$19.00
PII: S 0 0 2 2 - 3 1 1 5 (9 8) 0 0 3 1 7 - 1

Table 1
Chemical compositions of test materials (wt%)

Alloy	C	Si	Mn	P	S	Cr	Ni	Ti	Al	N	Fe
HP	0.003	0.01	1.36	0.001	0.0014	18.17	12.27	0.01	0.16	0.0014	bal.
HP + C	0.098	0.03	1.39	0.001	0.0020	18.30	12.50	<0.01	0.11	0.0016	bal.

2. Experimental

Chemical compositions of two materials of type 304 austenitic stainless steels are shown in Table 1, where alloy HP is a high-purity type base alloy and alloy HP + C is doped with carbon into the alloy HP. Solution annealed materials and 50% cold-worked materials of both alloys were prepared and machined into the round bar type tensile specimens with dimensions shown in Fig. 1. Those specimens were irradiated at 513 K in the Japan Research Reactor No. 3 (JRR-3) and at 323 K in JMTR, while the cold-worked specimens were irradiated only at 513 K in JMTR. In JRR-3, the specimens were irradiated in an inert gas environment and exposed in a primary coolant water in JMTR. Fast neutron fluences and irradiation time were 6.7×10^{24} n/m^2 ($E > 1$ MeV), 3925 h in JRR-3 and 9.0×10^{24} n/m^2 ($E > 1$ MeV), 1187 h in JMTR.

After the irradiation, susceptibilities to SCC of specimens were examined by the slow strain rate testing (SSRT) in high-temperature water using the test machine installed in a hot-cell at JAERI's hot laboratory. SSRTs were carried out at 573 K in the water saturated with pure-oxygen at room temperature where a dissolved oxygen (DO) level was 42 ppm. A flow rate of water was 5 l/h and conductivity was below 0.2 μS/cm at inlet of autoclave and in a range from 1.0 to 1.7 μS/cm at outlet. An initial strain rate for SSRT was 1.7×10^{-7} s^{-1} and specimens were strained up to failures in the strain rate. Fracture surfaces of SSRT specimens were examined with a scanning electron microscope (SEM) to estimate the susceptibility to IASCC.

3. Results

In Fig. 2, the engineering stress–strain (S–S) curves during SSRTs of specimens of alloys HP and HP + C irradiated at 513 and 323 K and unirradiated specimens are compared. Specimens irradiated at 513 K showed high susceptibilities to IASCC and the dominant fracture modes were intergranular type (IG) in alloy HP and transgranular type (TG) in alloy HP + C. In addition, the maximum stress of alloy HP + C was higher than that of alloy HP. Hence addition of carbon had affected both the fracture mode and radiation hardening. In case of the alloys irradiated at 323 K, as shown in Fig. 2, total elongation of both alloys HP and HP + C are larger and the yield stresses are lower than those of specimens irradiated at 513 K. The effect of carbon addition on a total elongation by SSRT were different between two irradiation temperature; at 513 K the carbon addition enhanced a reduction of the total elongation, while at 323 K the addition increased the total elongation.

SEM photographs in Fig. 3 show fracture surfaces of SSRT specimens of alloy HP. Although fractions of diametrical reduction and IGSCC area are different for each irradiation temperature, the similar morphologies are observed in both specimens. On the other hand, as shown in Fig. 4, fracture surface morphologies of alloy HP + C are remarkably different between the specimens irradiated at 513 and 323 K; at 513 K a cracking initially occurred as IGSCC and then propagated as TGSCC, and at 323 K the specimen was failed by IGSCC. Here, it must be noticed that the failure of the latter specimen occurred at a portion of specimen shoulder where a di-

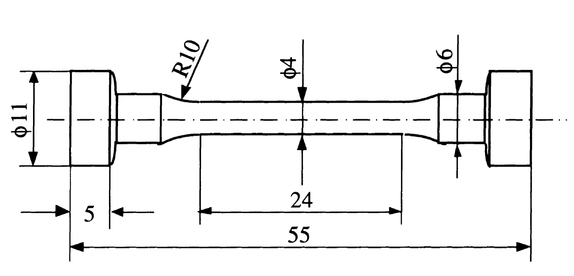

Fig. 1. Dimensions of SSRT specimens (mm).

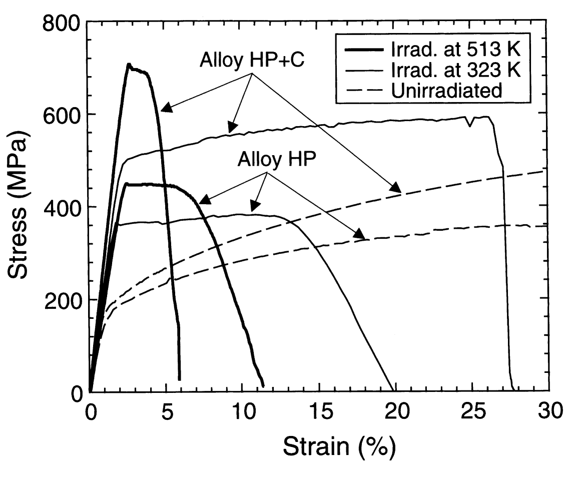

Fig. 2. S–S curves from SSRT on the specimens irradiated at 513 and 323 K.

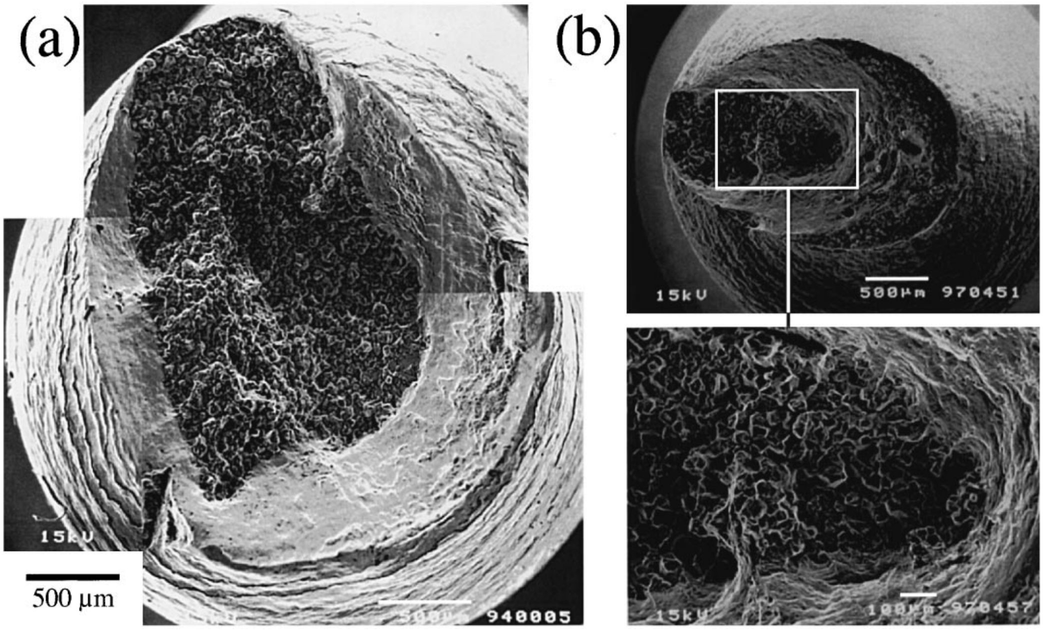

Fig. 3. Fracture surfaces of SSRT specimens of alloy HP irradiated at (a) 513 K and (b) 323 K.

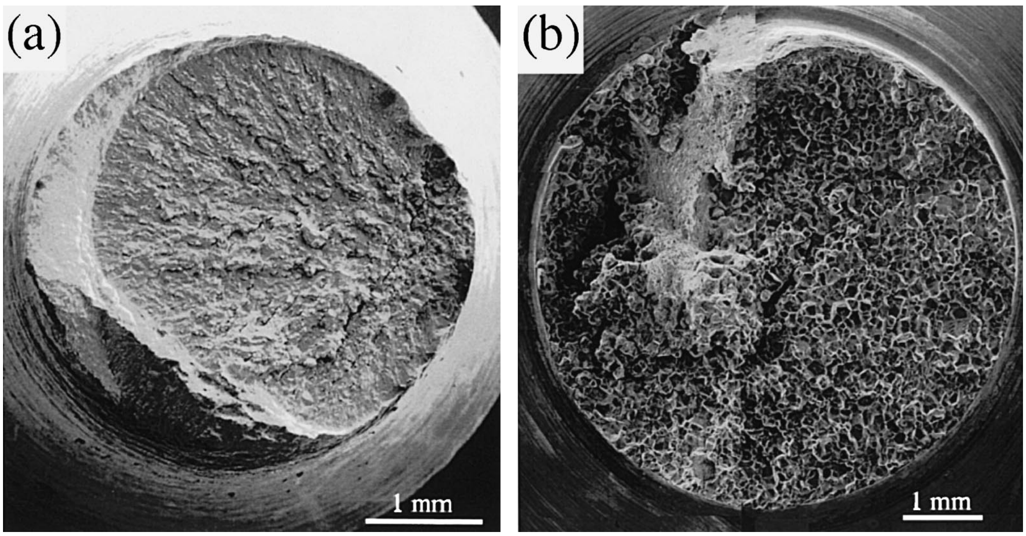

Fig. 4. Fracture surfaces of SSRT specimens of alloy HP + C irradiated at (a) 513 K and (b) 323 K.

ameter is 6 mm comparing with 4 mm at gauge section. This feature of fracture is seen in Fig. 5, where no other cracking is observed except for the cracking at shoulder portion. Loaded stress concentrated at this portion during SSRT can be estimated as about 260 MPa from a ratio of the cross sections and maximum stress during SSRT. This stress is nearly half of the yield stress of irradiated HP + C alloy as seen in Fig. 2, therefore a strain rate at this failed portion is negligibly small.

In Fig. 6, S–S curves of the cold-worked specimens tested at 513 K are shown which includes results of both the irradiated and unirradiated specimens. Although the maximum stresses were increased by the radiation hardening, basically S–S behavior of the specimens was not changed by the irradiation. Fig. 7 shows SEM photographs of fracture surfaces of the cold-worked and irradiated specimens. Both specimens of alloys HP and HP + C showed no appearance of SCC.

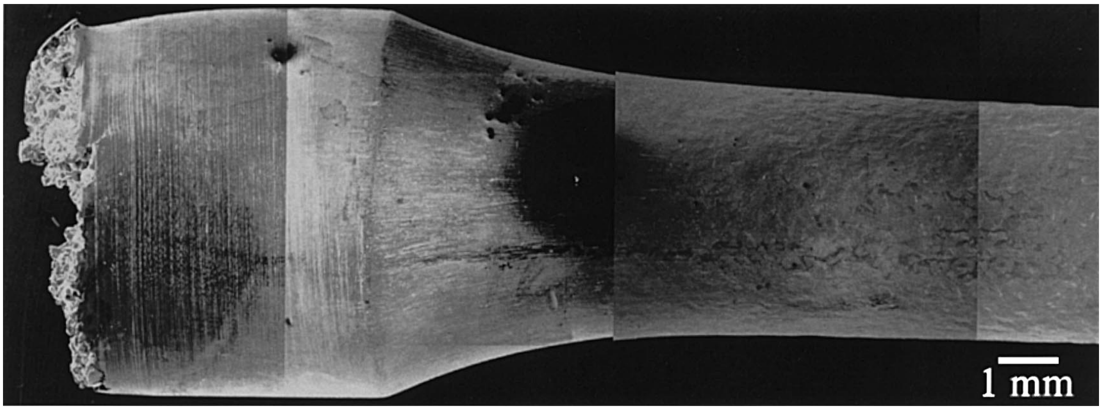

Fig. 5. Fractured SSRT specimen of alloy HP + C irradiated at 323 K.

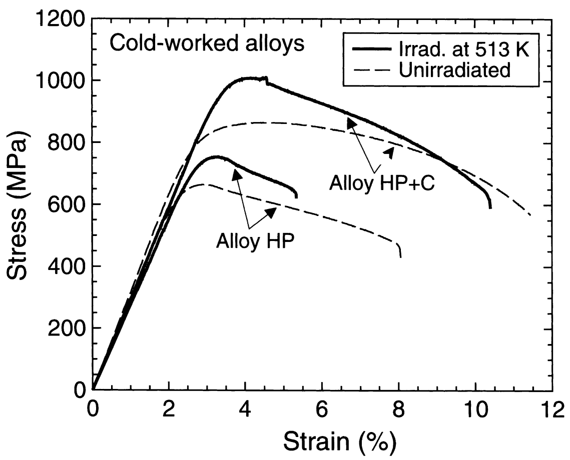

Fig. 6. S–S curves from SSRT on the cold-worked specimens irradiated at 513 K and unirradiated specimens.

4. Discussion

Results from SSRT on the materials irradiated below 373 K has been scarce except for the report by Kodama et al. [6] where thin plate type specimens with 0.5 mm in thickness had been used. Their high-purity type specimens irradiated in JMTR to 1.7×10^{25} n/m^2 ($E > 1$ MeV) showed a susceptibility to IGSCC of about 30% and it is consistent with the result of present study used round bar type specimens of alloy HP. A primary process of IASCC has been considered to be chromium depletion at the grain boundaries due to RIS, though the other unknown process strongly affects on initiation and propagation of cracking [4,7]. Breummer and Simonen [13] suggested that IASCC occurred when a chromium concentration is below about 16% in type 304 alloys that was easily reached at around 1 dpa in case of irradiation at 573 K. However, at a low temperature of 323 K an extent of RIS, e.g., chromium depletion, has never been identified. Simonen and Bremmer [8] investigated the RIS by a numerical method and showed a negligible segregation at 323 K. Watanabe et al. [9], however, carried out a similar calculation and supported a possible segregation at the low temperature. Although analyses of microchemistry of alloy HP irradiated at 323 K is necessary, as Kodama et al. [6] pointed out, it is considered that the chromium depletion may occur and probably cause IGSCC.

From observations of the fracture surfaces of alloy HP, it can be inferred that a localized deformation occurred easily after IGSCC propagated and then failed by ductile failure. On the other hand, as seen in Fig. 2, the specimen of alloy HP + C showed a work hardening with a large elongation and then cracked at the shoulder portion in a short time. It may imply that an addition of carbon did not reduce IASCC susceptibility, but it increased a durability to localized deformation and consequently suppressed IASCC at the gauge section. Katsura et al. [14] performed the uniaxial constant load (UCL) tests on neutron irradiated SSs and some of their specimens were failed at pinhole sections outside gauge section. A reason of those irregular fractures of specimens was inferred as radiolysis of water in a crevice due to gamma radiation from the specimen itself. High susceptibility to IGSCC of their specimens failed at pinhole section is similar to the present specimen of alloy HP + C. However, in our experiments no other SSRT specimens irradiated at higher temperatures in the past had shown such irregular cracking, so that it is difficult to attribute the cracking to the radiolysis. On the present specimen the gauge section was deformed under a relatively large strain rate, while the cracking portion was not deformed and under the nearly constant loading condition. These situations appear to cause a cracking at the shoulder portion of specimen.

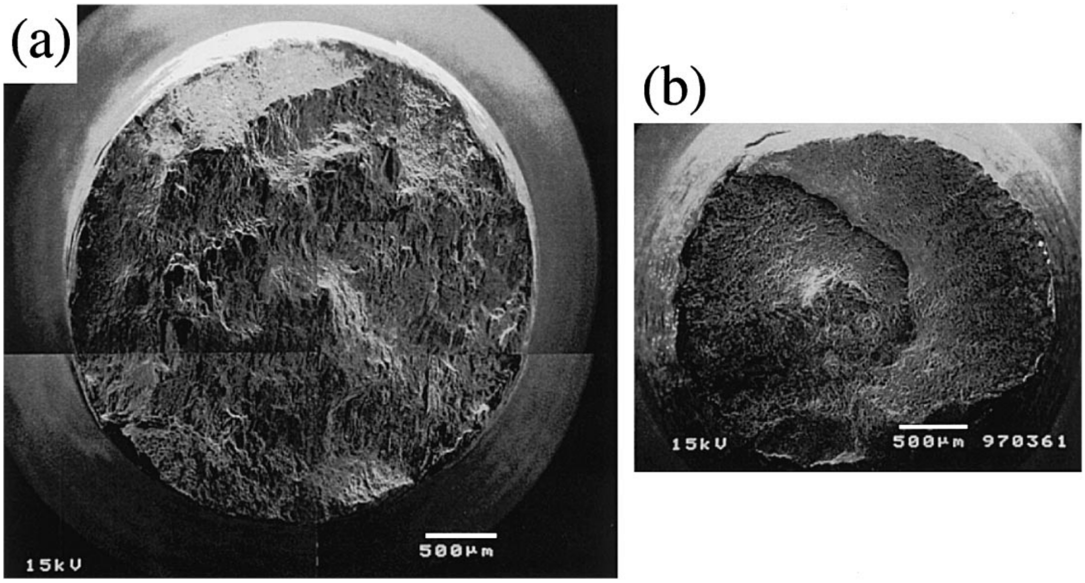

Fig. 7. Fracture surfaces of SSRT specimens of cold-worked alloys (a) HP and (b) HP + C irradiated at 513 K.

Cold working is known as one of the effective methods to suppress a void swelling in a higher neutron fluence region. Dumbill and Hanks [15] suggested that cold working could suppress RIS, e.g. chromium depletion, and consequently susceptibility to IASCC. They analyzed RIS on the ion irradiated specimens and found that chromium depletion was reduced by a cold-working of 5%. Kodama et al. [6] reported that the 20% cold worked type 304 SSs showed no susceptibility to IGSCC at 1.7×10^{25} n/m^2 ($E > 1$ MeV). In the present study, no IGSCC was observed in the cold-worked alloys HP and HP + C and it can be concluded that the 50% cold-working suppressed IASCC. This effect of cold working may be attributed to an increase in a density of dislocation network and therefore a number of sink of radiation induced point defects. As seen in Fig. 6, a fracture elongation of alloy HP specimen decreased after irradiation but it slightly decreased in alloy HP + C specimen, therefore addition of carbon seems to retard a fracture by SSRT of irradiated specimen.

5. Conclusions

Type 304 model SSs, alloys HP and HP + C, were irradiated at 513 K in JRR-3 and at 323 K in JMTR up to $7\text{–}9 \times 10^{24}$ n/m^2 ($E > 1$ MeV). After the irradiation, susceptibilities to SCC of specimens were examined by SSRT in high-purity 573 K water. The following conclusions were drawn:

1. Solution-annealed specimens of both alloys irradiated at 323 K showed susceptibilities to IGSCC. The specimen of alloy HP + C failed at a shoulder portion with very high susceptibility.
2. Addition of carbon enhanced IASCC accompanied by a change in the fracture morphology and increased radiation hardening after irradiation at 513 K, while it seems to increase an uniform elongation of specimens after irradiation at 323 K.
3. The 50% cold-worked specimens of both alloys irradiated at 513 K showed no susceptibility to IASCC.

Acknowledgements

The authors gratefully acknowledge S. Jitsukawa of JAERI for his helpful discussions. We are also grateful to the staffs of JAERI Oarai Hot Laboratory, especially to I. Goto, Y. Kato and H. Mimura for SSRT and SEM examinations.

References

[1] ITER Interim Design Report, San Diego, Garching, Naka (1995).
[2] P.L. Andresen, F.P. Ford, S.M. Murphy, J.M. Perks, Proceedings of the Fourth International Symposium on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors, NACE, 1990, pp. 1–83.
[3] P. Scott, J. Nucl. Mater. 211 (1994) 101.
[4] H.M. Chung, W.E. Ruther, J.E. Sanecki, A. Hins, N.J. Zaluzec, T.F. Kassner, J. Nucl. Mater. 239 (1996) 61.
[5] T. Tsukada, K. Shiba, G.E.C. Bell, H. Nakajima, CORROSION/92, NACE, 1992, Paper No. 104.

[6] M. Kodama, J. Morisawa, S. Nishimura, K. Asano, S. Shima, K. Nakata, J. Nucl. Mater. 212–215 (1994) 1509.
[7] G.S. Wass, S.M. Brummer, J. Nucl. Mater. 216 (1994) 326.
[8] E.P. Simonen, S.M. Bremmer, J. Nucl. Mater. 239 (1996) 185.
[9] S. Watanabe, N. Sakaguchi, N. Hashimoto, H. Takahashi, J. Nucl. Mater. 224 (1995) 158.
[10] F.A. Garner, Proceedings of the Symposium on Phase Stability During Irradiation, AIME, 1981, p. 165.
[11] T. Tsukada, Y. Miwa, H. Nakajima, Proceedings of the Seventh International Symposium on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors, NACE, 1995, p. 1009.
[12] Y. Miwa, T. Tsukada, S. Jitsukawa, S. Kita, S. Hamada, Y. Matsui, M. Shindo, J. Nucl. Mater. 233–237 (1996) 1393.
[13] S.M. Bruemmer, E.P. Simonen, Corrosion 50 (12) (1994) 940.
[14] R. Katsura, J. Morisawa, M. Kodama, S. Nishimura, S. Suzuki, S. Shima, M. Yamamoto, Proceedings of the Sixth International Symposium on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors, TMS, 1994, p. 625.
[15] S. Dumbill, W. Hanks, Proceedings of the Sixth International Symposium on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors, TMS, 1994, p. 521.