RADIATION HARDENING AND DEFORMATION BEHAVIOR OF IRRADIATED FERRITIC-MARTENSITIC STEELS — J. P. Robertson, R. L. Klueh (Oak Ridge National Laboratory), K. Shiba (Japan Atomic Energy Research Institute), and A. F. Rowcliffe (ORNL)

OBJECTIVES

The objectives of this work are to present recently generated tensile data and to review the existing database for ferritic/martensitic steels.

SUMMARY

Tensile data from several 8-12% Cr alloys irradiated in the High Flux Isotope Reactor (HFIR) to doses up to 34 dpa at temperatures ranging from 90 to 600°C are discussed in this paper. One of the critical questions surrounding the use of ferritic-martensitic steels in a fusion environment concerns the loss of uniform elongation after irradiation at low temperatures. Irradiation and testing at temperatures below 200-300°C results in uniform elongations less than 1% and stress-strain curves in which plastic instability immediately follows yielding, implying dislocation channeling and flow localization. Reductions in area and total elongations, however, remain high.

PROGRESS AND STATUS

Introduction

There are four basic goals to this work: (1)present Sandvik HT-9 (12Cr-1MoVW), F82H (8Cr-2WVTa), and modified 9Cr-1Mo (9Cr-1MoVNb) tensile data generated as part of the Collaboration on Fusion Materials between the United States Department of Energy (U.S. DOE) and the Japan Atomic Energy Research Institute (JAERI); (2) review the existing data for these and other ferritic-martensitic steels and to compare the U.S. DOE/JAERI (U.S./J) data with the existing database; (3) develop an understanding of the general dose-temperature regimes where these materials undergo irradiation hardening and loss of uniform elongation; 4) investigate the dose dependencies at various low temperatures. This report summarizes the progress in this effort to date.

Experimental Procedures

The alloy compositions and heat numbers of the ferritic/martensitic steels in the U.S. DOE/JAERI program are given in Table 1. Type SS-3 flat tensile specimens with gage sections 7.6 mm long by 0.76 mm thick were irradiated in either target or removable beryllium (RB) positions in the High Flux Isotope Reactor (HFIR) at ORNL. The irradiation doses ranged from 3 to 31 dpa and the temperatures ranged from 90 to 600°C (see Table 2). Uncertainty limits for the irradiation temperatures are also shown in Table 2. After irradiation, the specimens were tensile tested at the irradiation temperature, in vacuum, at strain rates of 5.6×10^{-4} - 1.1×10^{-3}/s. Comparisons of yield strength data taken from twin specimens from this and other experiments, combined with practical limitations of the testing system, reveal uncertainty limits on yield strength measurements of about ±20 MPa.

Results and Discussions

Representative F82H tensile curves from the U.S. DOE/JAERI program (200-600°C, 3-34 dpa) are shown in Figure 1. For irradiation temperatures less than or equal to 300°C, F82H irradiation hardens and undergoes a severe loss of uniform elongation (E_u) and strain hardening capacity. The maximum increase in yield strength (YS) due to irradiation under these conditions is less than 100%. The E_u is less than 3% for all doses and temperatures in this matrix (200-600°C, 3-34 dpa); the E_u for the same F82H material in the unirradiated condition is less than 5% in this temperature range (200-600°C). Higher dose irradiations at both 400 and 500°C result in slightly higher YS,

Table 1. Alloy Compositions in U.S. DOE/JAERI Program.

	Composition wt. %													
	Fe	Ni	Cr	Mo	Mn	Si	C	N	V	W	Ta	Nb	P	S
F82H (150 kg heat)	Bal	0.05	7.65	---	0.49	0.09	0.093	0.002	0.18	1.98	0.038	---	0.001	0.001
F82H (Heat No. 8091)	Bal	0.01	7.65	tr.	0.49	0.09	0.093	0.0019	0.18	1.98	0.038	---	0.005	0.001
9Cr-1MoVNb (Heat No. C9820)	Bal	0.11	8.62	0.98	0.36	0.08	0.090	0.050	0.209	0.01	---	0.063	0.008	0.004
HT-9 (Heat No.9607-R2)	Bal	0.51	12.1	1.04	0.57	0.17	0.20	0.027	0.28	0.45	---	---	0.016	0.003
HT-9 (Heat No.C9817)	Bal	0.43	11.99	0.93	0.50	0.18	0.21	0.02	0.27	0.54	---	0.018	0.011	0.004

Table 2. U.S. DOE/JAERI Program HFIR Irradiation Conditions.

Capsule	Position	Fast Flux (E>0.1 MeV) $\times 10^{21}$n/cm^2	Thermal Flux (E<0.5 eV) $\times 10^{21}$n/cm^2	Neutron dose, dpa	Temp. °C	Alloy	Heat Number	Heat Treatment
JP13	target	21	31	15.3	400±30	F82H	150 kg Heat	1040°C/0.5hr/AC+740°C/1.5hr/AC
JP14	target	42	71	31	400±30 500±30	F82H	150 kg Heat	1040°C/0.5hr/AC+740°C/1.5hr/AC
JP17	target	3.7	4.8	2.7	275±25	HT-9 F82H	9607R2 150 kg Heat	1050°C/1hr/AC+780°C/2.5hr/AC 1040°C/0.5hr/AC+740°C/1hr/AC
JP18	target	3.7	6.4	2.7	90±10	HT-9	9607R2	1050°C/1hr/AC+780°C/2.5hr/AC
JP20	target	10	17	7.4	300±30 500±30 600±30	F82H	8091	1040°C/0.67hr/AC+740°C/2hr/AC
JP22	target	42	71	31	300±30 400±30 500±30 600±30	HT-9 9Cr-1MoVNb	C9817 C9820	1050°C/1hr/AC+780°C/2.5hr/AC 1050°C/0.5hr/AC+700°C/5hr/AC
200J-1	RB 4.2 mm Hf shield	19	3.5	11.4	200±25	F82H HT-9 9Cr-1MoVNb	8091 C9817 C9820	1040°C/0.67hr/AC+740°C/2hr/AC 1050°C/1hr/AC+780°C/2.5hr/AC 1050°C/0.5hr/AC+700°C/5hr/AC
400J-1	RB 4.2 mm Hf shield	19	3.5	11.4	400±25	F82H HT-9 9Cr-1MoVNb	8091 C9817 C9820	1040°C/0.67hr/AC+740°C/2hr/AC 1050°C/1hr/AC+780°C/2.5hr/AC 1050°C/0.5hr/AC+700°C/5hr/AC

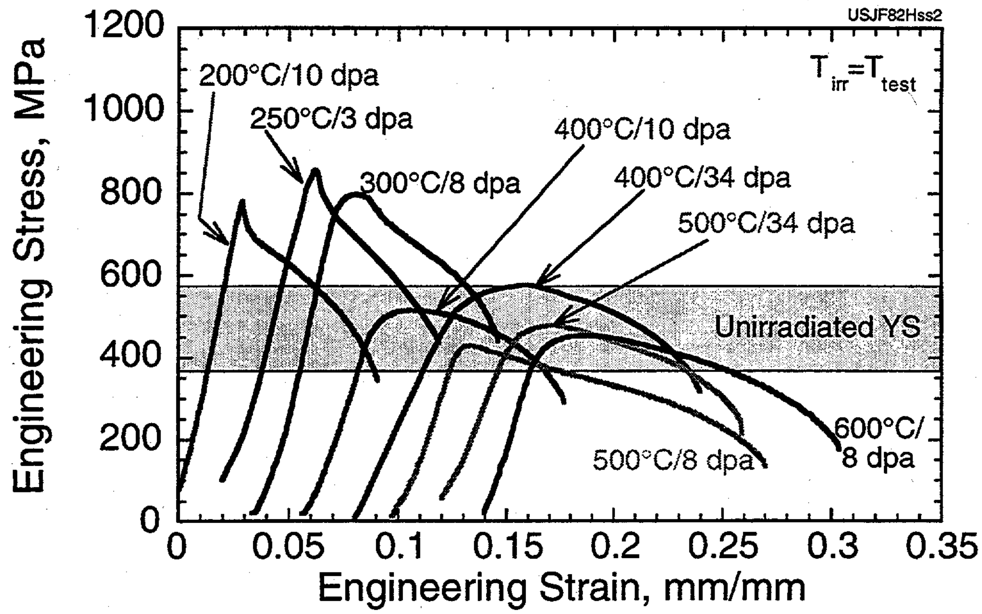

Figure 1. Representative U.S. DOE/JAERI F82H data: 200-600°C, 3-34 dpa.

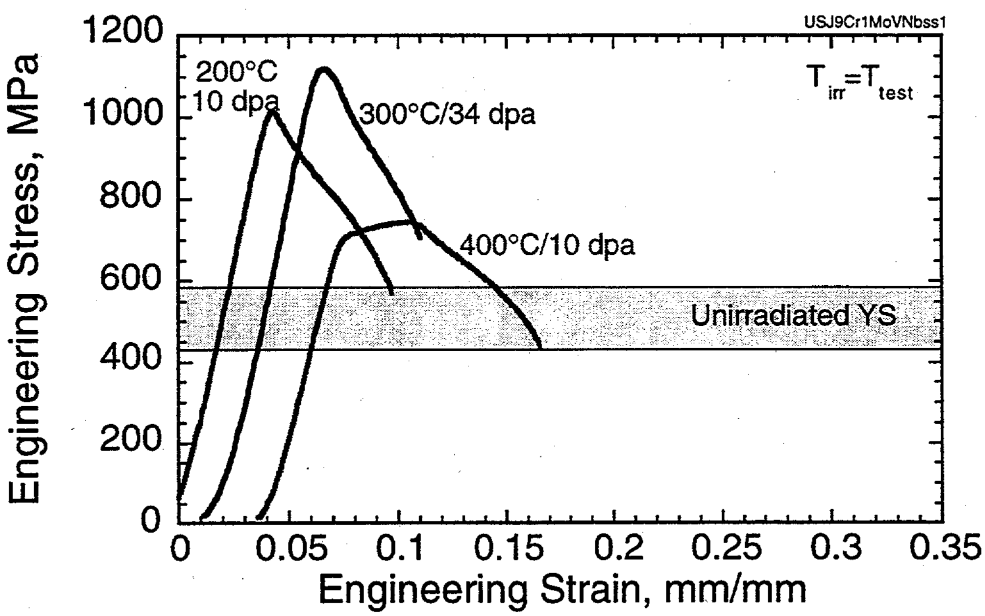

Figure 2. Representative U.S. DOE/JAERI 9Cr-1MoVNb data: 200-400°C, 10-34 dpa.

ultimate tensile strength (UTS), E_u, and total elongation (E_t). The range of unirradiated YS in the temperature range of 200-600°C is shown on the figure.

Representative tensile curves from the 9Cr-1MoVNb alloy tensile tests are given in Figure 2. The 9Cr-1MoVNb irradiation hardens at irradiation temperatures of 200-400°C and undergoes a severe loss of E_u at 200 and 300°C. After irradiation to 10 dpa at 200°C, the 9Cr-1MoVNb has a yield strength about 25% higher than that of F82H. The range of unirradiated YS in the temperature range of 200-400°C is shown on the figure.

Figure 3 shows a set of representative HT-9 tensile curves. Irradiation hardening occurs for all irradiation temperatures in this matrix, 90-400°C, but the increase in YS is less than 100% in each case. The maximum hardening occurs at irradiation temperatures of 200-300°C while the minimum E_u occurs at 90°C. Unlike the F82H alloy, HT-9 retains some strain hardening capacity at irradiation temperatures of 250-400°C. After irradiation to 10 dpa at 200°C, the HT-9 yield strength is almost 40% higher than that for F82H. The range of unirradiated YS in the temperature range of 90-400°C is shown on the figure.

Yield strength data from both the literature [1-25] and the U.S. DOE/JAERI (U.S./J) program are plotted in Figure 4. Examination of the F82H (8Cr) and 9Cr-1MoVNb (9Cr) YS, UTS, E_u, and E_t data as a function of temperature reveals no significant differences between the two alloys and so both the 8 and 9Cr alloys are represented in Fig. 4 by a single symbol after irradiation. (It can be seen in Fig. 4 that the 8-9Cr steel unirradiated tensile data fall in the same band as a function of temperature.) Irradiation increases the yield strength for irradiation temperatures (T_{irr}) less than 400°C, and the radiation hardening is temperature independent up to about 350°C, decreasing sharply in the range 400-450°C. Because of the strong temperature dependence in the range 400-450°C, uncertainties in irradiation temperature result in large differences in irradiated YS in this regime. The U.S. DOE/JAERI F82H data fit in well with the existing data base. The hardening dependence on temperature is clearly visible even with the very large dose range plotted here.

For the 8-9Cr steels, the irradiated E_u is very low (< 1%) for $T_{irr} \leq 300$°C, as shown in Fig. 5. Uniform elongations recover slightly (up to 2-3%) for temperatures above 400°C. Unirradiated 8-9Cr alloys show large scatter in E_t, with no apparent temperature dependence -- E_t values vary from 5 to 30% for T_{irr} from 25 to 700°C. Yield strength is plotted as a function of dose in Fig. 6. For T_{irr} = 500-600°C, there is an initial increase in YS with dose, and then the 8-9Cr steels undergo some softening with increasing dose. For T_{irr} = 430-460°C, there is no significant change in YS with dose up to 94 dpa. For 8-9Cr steels irradiated at temperatures below 300°C, the yield strength approaches saturation at about 5 dpa; the saturation YS value is approximately 1000-1100 MPa.

The HT-9 steel shows moderate irradiation hardening (change in YS less than 100%) for T_{irr} up to and including 400°C (see Fig. 7); hardening then decreases with increasing temperature. The hardening regime extends to higher temperatures (about 50°C higher) compared to the 8-9Cr steels. Radiation hardening is almost independent of temperature up to 400°C and then decreases sharply in the range of 450-500°C. The HT-9 data from the U.S. DOE/JAERI program are consistent with previous data. Unirradiated 8-9Cr and HT-9 yield strengths are the same for $T \leq 400$°C. For HT-9, the irradiated E_u is very low (<1%) for $T_{irr} \leq 200$°C; it recovers up to 5% at temperatures of 250-400°C (see Fig. 8). The unirradiated E_u of HT-9 is slightly higher than that for F82H and 9Cr-1MoVNb. As observed for the 8-9Cr alloys, the unirradiated HT-9 specimens show large scatter in E_t, with no apparent temperature dependence; E_t values vary from 8 to 25% for T_{irr} from 25 to 600°C. For irradiation temperatures below 365°C, the dose dependence of HT-9 is similar to the 8-9Cr steels.

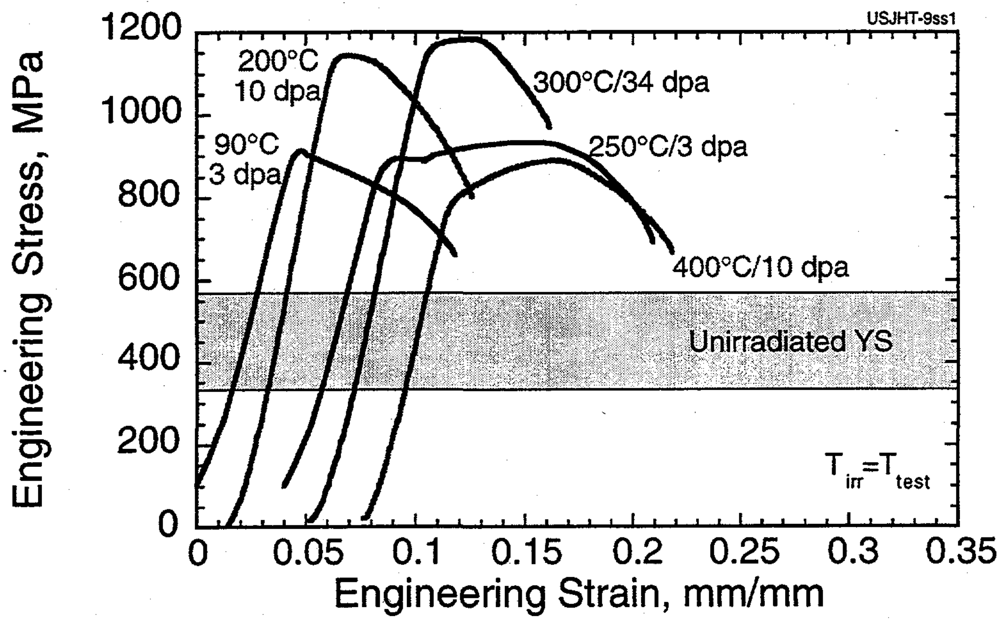

Figure 3. Representative U.S. DOE/JAERI HT-9 data: 90-400°C, 3-34 dpa.

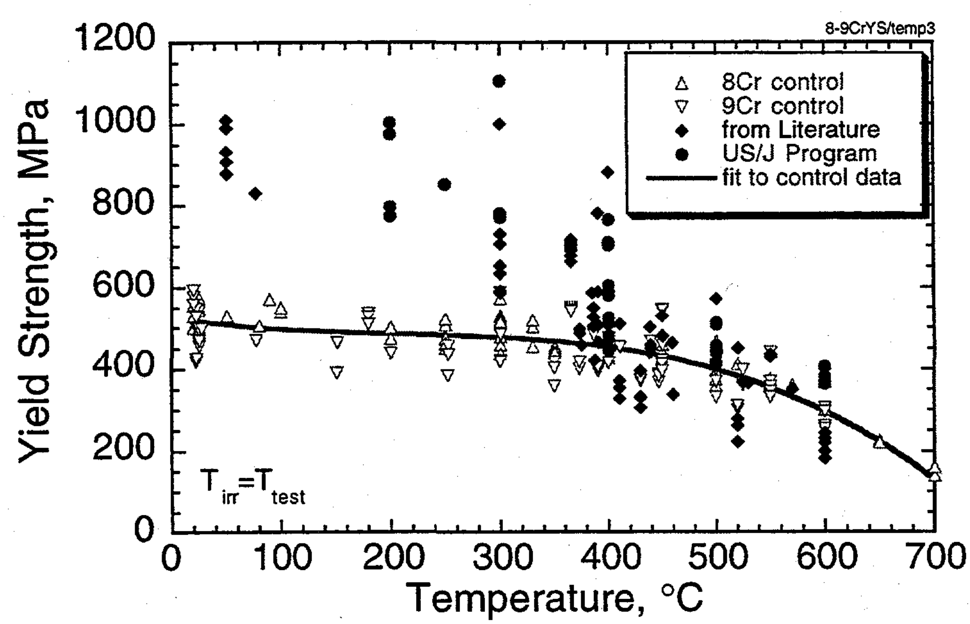

Figure 4. Yield strength as a function of temperature for 8-9Cr steels (0.1-94 dpa) [1-25].

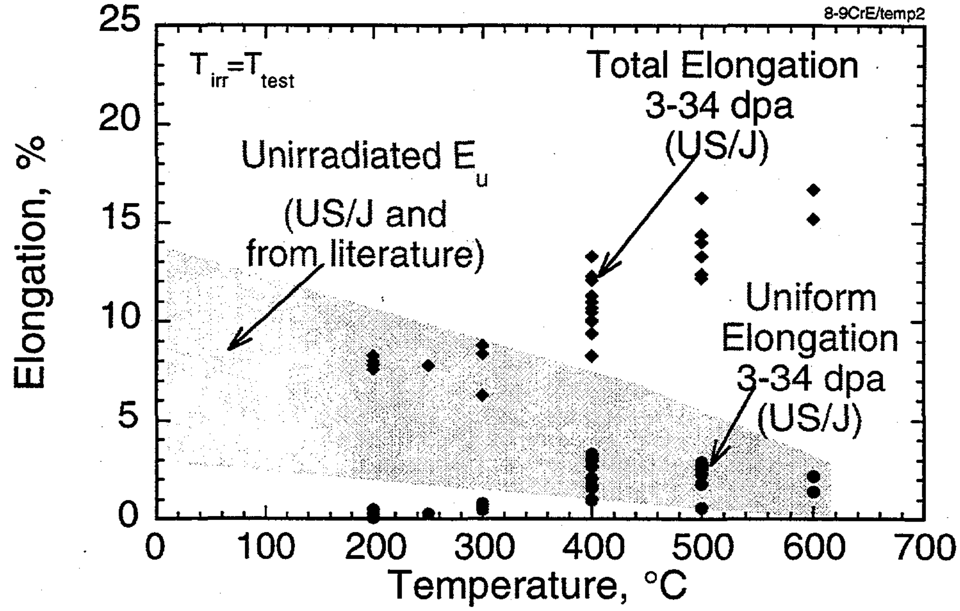

Figure 5. Uniform and total elongation for 8-9Cr steels as a function of temperature [1-26].

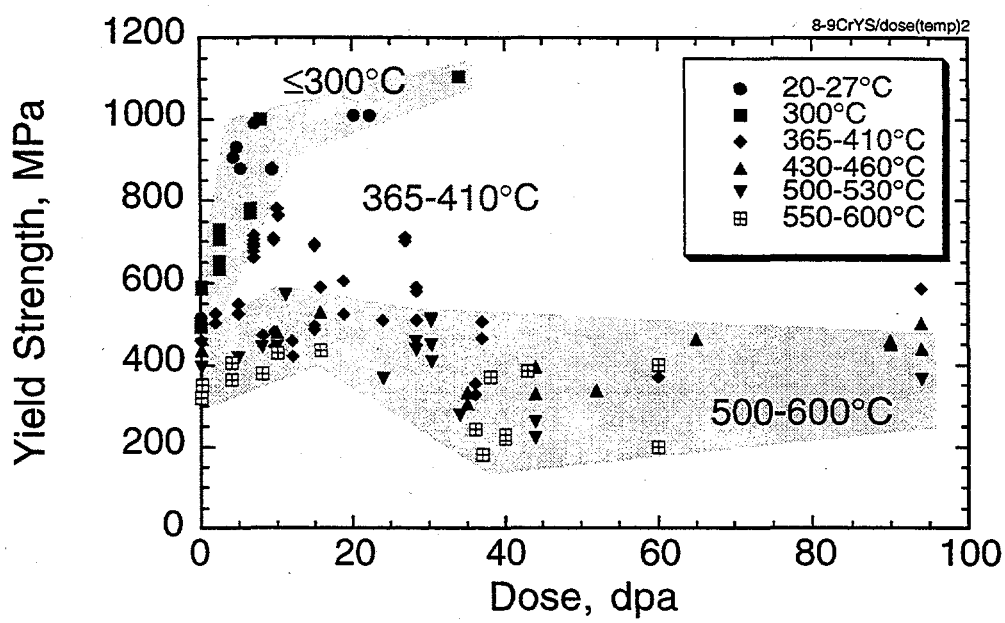

Figure 6. Yield strength as a function of dose and temperature for 8-9Cr steels [1-27].

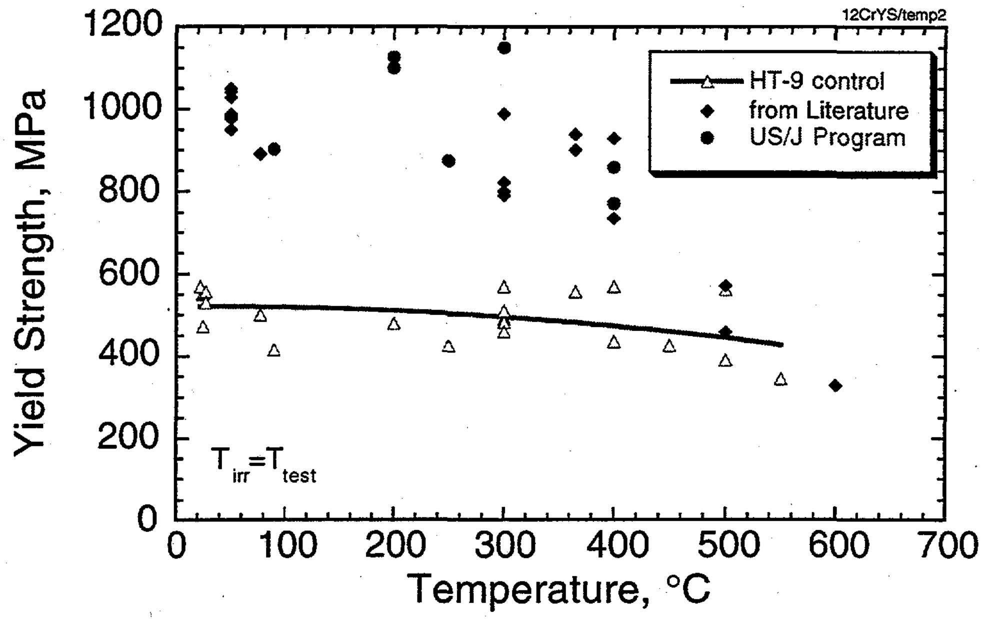

Figure 7. Yield strength as a function of temperature for HT-9 steels [1-27].

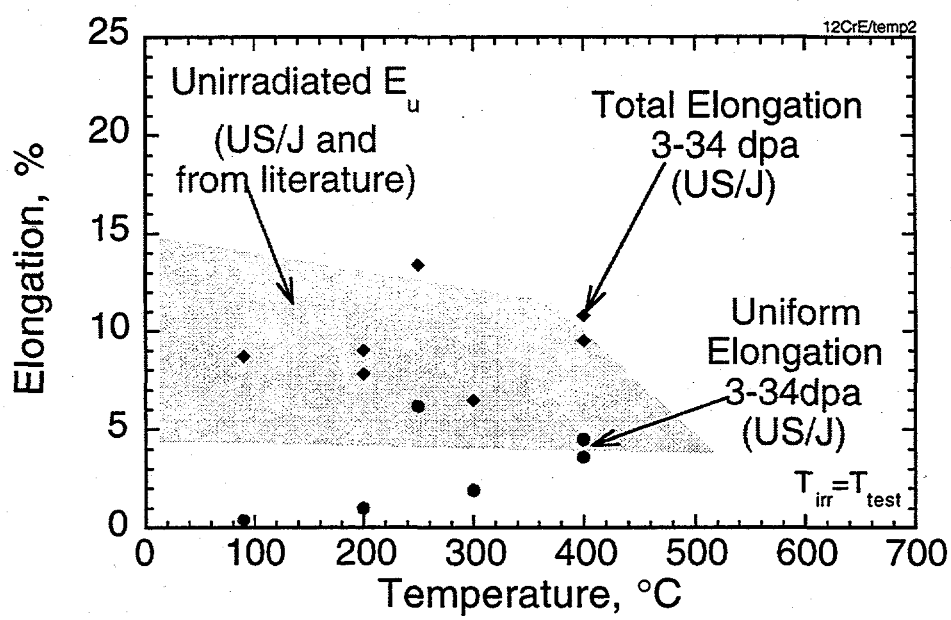

Figure 8. Uniform and total elongation as a function of temperature for HT-9 steels [2, 5, 7, 8, 10, 13, 14, 16, 28].

CONCLUSIONS

The U.S. DOE/JAERI HFIR data for F82H, 9Cr-1MoVNb, and HT-9 fit in well with the existing database; tensile properties for unirradiated 8-9Cr and HT-9 steels fall on the same trend line for $T_{irr} \leq 400°C$. For F82H, irradiation at temperatures less than 400°C results in irradiation hardening and a severe loss of uniform elongation and strain hardening capacity. For T_{irr} = 450-600°C, the irradiated YS are the same as for the unirradiated materials. Radiation hardening is temperature independent up to about 350°C, then decreases sharply at 400-450°C. Because of the strong temperature dependence in the range 400-450°C, uncertainties in irradiation temperature result in large uncertainties in irradiated mechanical properties. For HT-9, irradiation at temperatures less than or equal to 400°C results in irradiation hardening. The hardening regime extends to higher temperatures (about 50°C higher) compared to the 8-9Cr steels. Radiation hardening is almost independent of temperature up to 400°C, then decreases sharply at 450-500°C. The HT-9 retains some strain hardening capacity for T_{irr} = 250-400°C. For 8-9Cr steels irradiated at temperatures below 300°C, the yield strength approaches saturation at about 5 dpa; saturation YS ≈ 1000-1100 MPa. For irradiation temperatures below 365°C, the dose dependence of HT-9 is similar to the 8-9Cr steels.

FUTURE WORK

Future work includes the continued development of the ferritic/martensitic steel database by widening the literature search and testing additional specimens from the U.S./JAERI matrix.

REFERENCES

1. A. Alamo, J. L. Seran, O. Rabouille, J. C. Brachet, A. Maillard, H. Touron, and J. Royer, Effects of Radiation on Materials: 17th International Symposium, ASTM STP 1270, D. S. Gelles, R. K. Nanstad, A. S. Kumar, and E. A. Little, Eds., American Society for Testing and Materials, 1996, p. 761.
2. M. G. Horsten, Proceedings of the IEA Working Group Meeting on Ferritic/Martensitic Steels, Culham, United Kingdom, October 1996, ORNL/M-5674.
3. I. Ioka et al, unpublished data.
4. A. Kimura, T. Morimura, M. Narui, and H. Matsui, J. Nucl. Mater. 233-237 (1996) 319.
5. R. L. Klueh and P. J. Maziasz, J. Nucl. Mater. 187 (1992) 43.
6. R. L. Klueh and J. M. Vitek, J. Nucl. Mater. 132 (1985) 27.
7. R. L. Klueh and J. M. Vitek, J. Nucl. Mater. 150 (1987) 272.
8. R. L. Klueh and J. M. Vitek, J. Nucl. Mater. 161 (1989) 13.
9. R. L. Klueh, P. J. Maziasz and J. M. Vitek, J. Nucl. Mater. 141-143 (1986) 960.
10. R. L. Klueh et al., unpublished data.
11. R. L. Klueh, J. M. Vitek and M. L. Grossbeck, Effects of Radiation on Materials: 11th International Symposium, ASTM STP 782, H. R. Brager and J. S. Perrin, Eds., American Society for Testing and Materials, 1982, p. 648.
12. R. L. Klueh, Proceedings of the IEA Working Group Meeting on Ferritic/Martensitic Steels, Culham, United Kingdom, October 1996, ORNL/M-5674.
13. R. L. Klueh, J. Nucl. Mater. 179-181 (1991) 728.
14. A. Kohyama and Y. Kohno, Proceedings of the IEA Working Group Meeting on Ferritic/Martensitic Steels, Culham, United Kingdom, October 1996, ORNL/M-5674.
15. R. Lindau, Proceedings of the IEA Working Group Meeting on Ferritic/Martensitic Steels, Culham, United Kingdom, October 1996, ORNL/M-5674.
16. A. F. Rowcliffe, J. P. Robertson, R. L. Klueh, K. Shiba, D. J. Alexander, M. L. Grossbeck, and S. Jitsukawa, presented at the 8th International Conference on Fusion Reactor Materials, Sendai, Japan, October 1997, and to be published in the J. Nucl. Mater.
17. J. L. Seran, A. Alamo, A. Maillard, H. Touron, J. C. Brachet, P. Dubuisson, and O. Rabouille, J. Nucl. Mater. 212-215 (1994) 588.

18. J. L. Seran, et al., Effects of Radiation on Materials: 15th International Symposium, ASTM STP 1125, R. E. Stoller, A. S. Kumar, and D. S. Gelles, Eds., American Society for Testing and Materials, 1992, p. 1209.
19. K. Shiba et al., unpublished data.
20. K. Shiba, Proceedings of the IEA Working Group Meeting on Ferritic/Martensitic Steels, Culham, United Kingdom, October 1996, ORNL/M-5674.
21. K. Shiba, M. Suzuki, and A. Hishinuma, J. Nucl. Mater. 233-237 (1996) 309.
22. K. Shiba, M. Suzuki, A. Hishinuma, and J. E. Pawel, Effects of Radiation on Materials: 17th International Symposium, ASTM STP 1270, D. S. Gelles, R. K. Nanstad, A. S. Kumar, and E. A. Little, Eds., American Society for Testing and Materials, 1996, p. 753.
23. M. Victoria, D. Gavillet, P. Spatig, F. Rezai-Aria, and S. Rossman, J. Nucl. Mater. 233-237 (1996) 326.
24. M. Victoria, E. Batawi, Ch. Briguet, D. Gavillet, P. Marmy, J. Peters, and F. Rezai-Aria, Effects of Radiation on Materials: 17th International Symposium, ASTM STP 1270, D. S. Gelles, R. K. Nanstad, A. S. Kumar, and E. A. Little, Eds., American Society for Testing and Materials, 1996, p. 721.
25. N. Yamanouchi, M. Tamura, H. Hayakawa, A. Hishinuma, and T. Kondo, J. Nucl. Mater. 191-194 (1992) 822.
26. K. Shiba et al., unpublished data.
27. A. Koyama and T. Hasegawa, Proceedings of the IEA Working Group Meeting on Ferritic/Martensitic Steels, Culham, United Kingdom, October 1996, ORNL/M-5674.
28. R. L. Klueh, J. M. Vitek, and M. L Grossbeck, J. Nucl. Mater. 103-104 (1981) 887.