Transactions of the Korean Nuclear Society Autumn Meeting
Gyeongju, Korea, October 29-30, 2015

An Overview of Irradiation Creep of Stainless Steels

Woo Seog Ryu
Advanced Fuel Development Division, Korea Atomic Energy Research Institute, Daejeon, 305-353, Republic of Korea
**Corresponding author: wsryu@kaeri.re.kr*

1. Introduction

An irradiation creep model of SFR core materials is necessary to apply to the fuel cladding and assembly materials of domestic SFR reactor system. The document of in-reactor irradiation creep has been obtained by investing a long time and large-scale cost using limited experimental research reactors. This paper reviewed systematically a state-of-art of irradiation creep for stainless steels to provide a background information for performing irradiation creep tests and establishing the creep model for advanced domestic steels effectively.

2. Technical Status

A history of irradiation creep is reviewed starting with the documents (first report in 1966) issued by the Dounreay team in UK, as summarized in table 1[1,2]. In main, the series of irradiation test results are collected and summarized which a National Cladding/Duct Materials Development Program under US Atomic Energy Commission (US NCD prgram) carried out through various national institutes. Moreover, the mechanisms of irradiation creep proposed by the test results in worldwide are surveyed. Finally the models of irradiation creep established by the US National Program are reviewed.

Table 1 History of irradiation creep

material	year	authors
Uranium	1955	Roberts and Cottrell
Zirconium alloys	1966	Fidleris and Williams
Modeling	1967	Gilbert, Hesketh
Stainless steels	1967	Mosedale
High fluence effects	1973	Boltax et al.
LMFBR Cladding & Duct US Program	1974	ANL, HEDL
In-reactor creep eq. for 316SS	1982	Puigh et al.

Irradiation creep was linearly dependent on stress and neutron fluence, as shown in figures 1&2. but it was't dependent on temperature below 540℃, as shown in figure 3. A general equation of irradiation creep[3～5] was proposed as a function of stress(σ) and neutron fluence(∅);

$$\dot{\epsilon}_{irr.creep} = (B\phi + D\dot{S}_o)\overline{\sigma}.$$

The mechanisms of SIPN(stress-induced preferred loop nucleation) and SIPA(stress-induced preferred absorption) were suggested as the mechanism of irradiation creep. Investigates of microstructures after irradiation creep tests observed dislocation loops of interstitial atoms produced by neutron collisions. The dependence of interstitial loops with stress informed that growth rate of loop was more resonable than nucleation, so SIPA mechanism was accepted for irradiation creep of austenitic stainless steels[6～8].

At high temeprature above 540℃, thermal creep has a dominant role, so irradiation creep was only a component of in-reactor creep. A terminology of irradiation creep was used as same meaning as in-reactor creep at low temperature below 540ºC, because thermal creep was negligible. But at high temperature above 540ºC, thermal creep has a dominant role in in-reactor creep, so irradiation creep was handled as an extension of the behavior of low temperature irradiation creep.

US NCD program developed an empirical equation of austenitic stainless steels as a following[9].

$$\begin{aligned}\overline{\epsilon} = {} & A_1[1-\exp(-3\Phi)]\overline{\sigma} \\ & + A_2(2.78\times10^{-4}t)^{1/2}\,\overline{\sigma}^{4.5} \\ & + A_3(\sinh\frac{\overline{\sigma}}{\sigma_1 H})^3(2.78\times10^{-4}t)^3 \\ & + A_4\Phi\overline{\sigma} \\ & + \frac{2.2A_5}{A_6}\Omega^2[\ln(\cosh\frac{\Phi}{\Omega})]\overline{\sigma}\end{aligned}$$

In-reactor creep was consisted of both irradiation terms and thermal terms based on experimental data. Three terms of A1～A3 were the primary, steady-state and tertiary creep behaviors of thermal creep, and two terms of A4～A5 were steady-state and tertiary creep behaviors of irradiation creep.

3. Conclusions

Irradiation creep curves of the cladding material should be used to evaluate the strain of SFR fuel rod, and the irradiation creep rupture times of cladding material be necessary to calculate the cumulative damage factor of the rod. This paper will provide the knowledge to understand the irradiation creep and to obtain the background informations of advanced domestic steels, so that it hopes to practically apply for timely producing the documents of irradiation creep of advanced domestic steels necessary for the national SFR program.

Acknowledgement

This project has been carried out under the Nuclear R&D program by Ministry of Science, Ict & Future Planning.

REFERENCES

[1] G.W.Lewthwaite and D.Mosedale, Nature 216 (1967) 472-473

[2] J.L.Straalsund, R.W.Powel and B.A.Chin (HEDL), Journal of Nuclear Materials, 108&109 (1982) 299-305

[3] W.G.Wofer and A.Boltax, Stress-Induced Deformation of Metals during Fast-Neutron Irradiation, European Conference on Irradiation Embrittlement and Creep in Fuel Cladding and Core components, London, CONF-721115-7 (1972)

[4] E.R.Gilbert, J.L.Straalsund and G.L.Wire, Irradiation Creep Data in Support of LMFBR Core Design, Journal of Nuclear Materials 65 (1977) 266-278

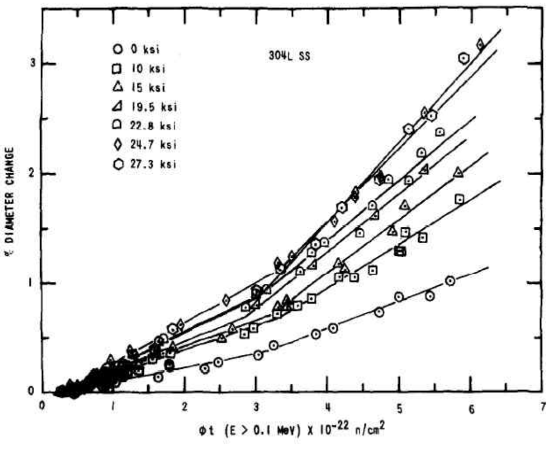
[5] B.A.Chin, J.L.Straalsund and G.L.Wire, Irradiation Creep of Previously Irradiated 304 Stainless Steel in a Fast Flux Environment, Journal of Nuclear Materials, 75 (1978), 274-281

[6] J.R.Mathews and M.W.Finnis, Irradiation Creep Models-An Overview, Jounal of Nuclear Materials 159 (1988), 257-285

[7] K.Ehrlich, Irradiation Creep and Interrelation with Swelling in Austenitic Stainless Steel, Journal of Nuclear Materials, 100 (1981) 149-166

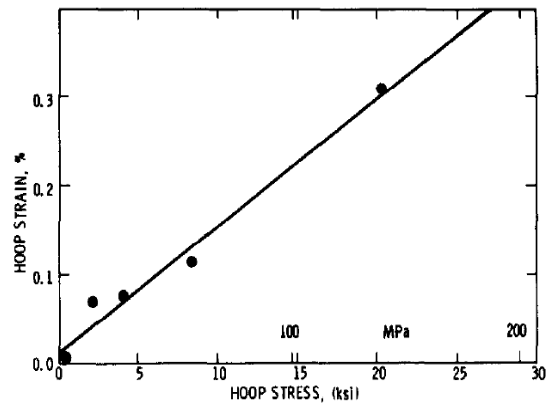
[8] F.A.Garner and D.S.Gelles, Irradiation Creep mechanisms: An Experimental Perspective Journal of Nuclear Materials 159 (1988), 286-309

[9] R.J.Puigh, E.R.Gilbertand and B.A.Chin, An In-Reactor Creep Correlation for 20% Cold-Worked AISI316 StainlessSteel, ASTM 11th Confernce on Effects of Radiation of Materials, ASTM STP782 (1982) 108-120



Composite of all diameter changes vs fluence data from the Type 304L stainless steel creep experiment.

Figure 1. Irradiation creep of type 304 in EBR-II



Stress dependence of irradiation creep 20% CW Type 316 stainless steel 520°C – 3 X 10^{22} n/cm^2.

Figure 2. Stress dependence of irradiation creep of type 316 in EBR-II

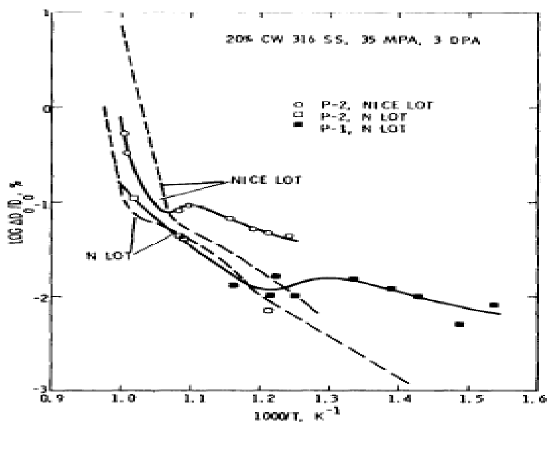


Figure 3. Temperature dependence of irradiation creep for 316CW, relative to thermal creep