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# Material properties of Grade 91 steel at elevated temperature and their comparison with a design code

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## 1. Introduction

Mod.9Cr-1Mo (ASME Grade 91; Gr.91) steel is widely adopted as candidate material for Generation IV nuclear systems as well as for advanced thermal plants. In a Gen IV sodium-cooled fast reactor of the PGSFR (Prototype Gen IV Sodium-cooled Fast Reactor) being developed by KAERI (Korea Atomic Energy Research Institute), Gr.91 steel is selected as the material for the steam generator, secondary piping, and decay heat exchangers [1-2]. However, as this material has a relatively shorter history of usage in an actual plant than austenitic stainless steel, there are still many issues to be addressed including the long-term creep rupture life extrapolation and ratcheting behavior with cyclic softening characteristics.

In this study, the material properties of tensile strength, creep properties, and creep crack growth model for Gr.91 steel at elevated temperature were obtained from material tests at KAERI, and the test data were compared with those of the French elevated temperature design code, RCC-MRx [3]. The conservatism of the material properties in the French design code is highlighted.

## 2. Tensile and creep properties of Gr.91 steel

Tensile and creep tests were conducted for Gr.91 steel at 600°C. The chemical compositions for Gr.91 steel used in the present material tests, as shown in Table 1 and a schematic of plate specimen for the tensile test and welded joint are shown in Fig. 1.

Table 1. Chemical composition of Gr.91 steel

| C | Si | Mn | P | S | Ni | Cr |
|---|---|---|---|---|---|---|
| 0.115 | 0.23 | 0.415 | 0.012 | 0.0014 | 0.22 | 8.9 |

| Mo | Cu | V | Al | N | Nb |
|---|---|---|---|---|---|
| 0.869 | 0.038 | 0.194 | 0.02 | 0.0513 | 0.073 |

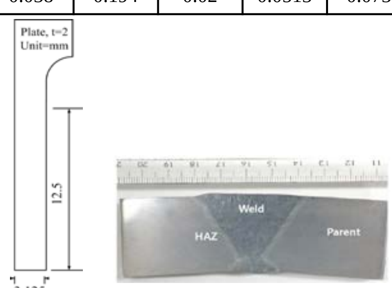


Fig. 1 Schematics of tension test specimen and weld- joint

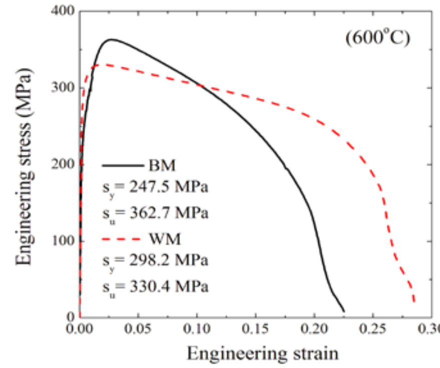


Fig. 2 tensile test results for Gr.91 steel at 600°C

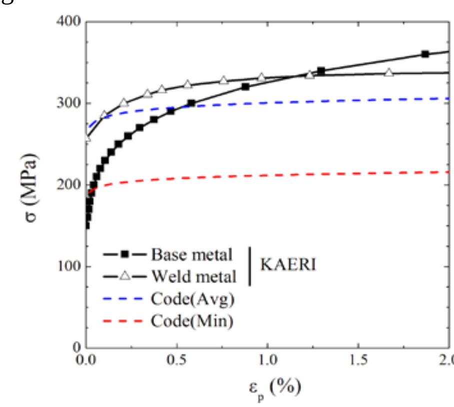


Fig. 3 Comparison of tensile test results for base, weld metal, and RCC-MRx

The tensile test results at 600°C with the values of yield strength (YS) and ultimate tensile strength (UTS) data for the base and weld metal are shown in Fig. 2. It is shown that base metal has a higher YS but lower UTS compared to the weld metal.

When the tensile data were compared to the RCC-MRx code, Fig. 3 shows that hardening for the base metal is higher than that of the code (RCC-MRx), and the hardening of the weld metal is similar to that of the code.

Creep tests for the base and weld metal of Gr.91 steel were conducted with a round bar with a diameter of 6mm at the central part. Power law creep properties were obtained from creep tests for the base and weld metals and the properties of A and n based on the minimum creep rate in the power law creep are shown in Fig. 4. The creep strain behavior under a load of 150MPa is shown in Fig. 5. The RCC-MRx code properties are located between the weld and base metals.

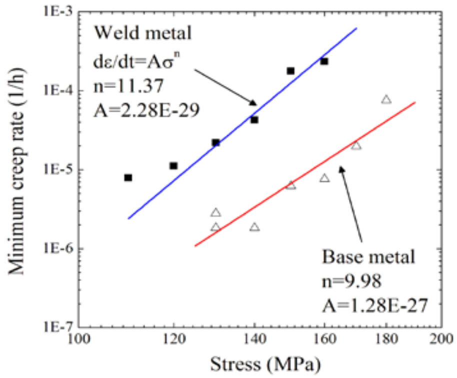


Fig. 4 Power law creep properties of Gr.91 at 600°C

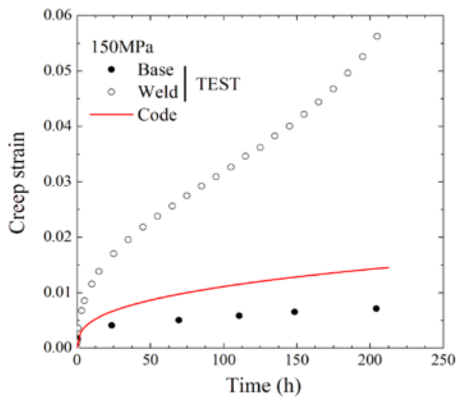


Fig. 5 Creep strain behavior of Gr.91 under 150MPa

### 3. Creep crack growth properties

Creep crack growth tests were conducted for Gr.91 steel at 600°C with a specimen half inch thickness, as shown in Fig. 6.

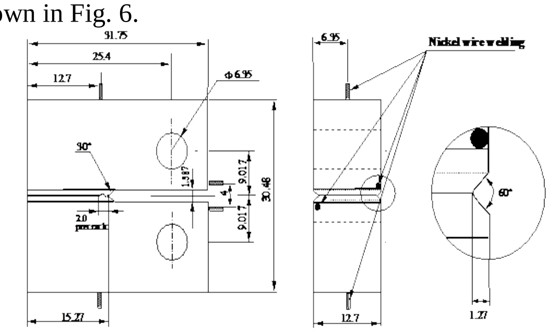


Fig. 6 Configuration of creep crack growth test specimen

Creep crack growth behaviors against a normalized creep rupture life are shown in Fig. 7 for the base metal, and Fig. 8 for the weld metal.

The final comparison of the creep crack growth for the Gr.91 base and weld metal with the RCC-MRx code is shown in Fig. 9. It should be noted that RCC-MRx provides an unconservative CCG model in a dangerous manner based on this test result, which means that the code property needs further validation for this CCG material property.

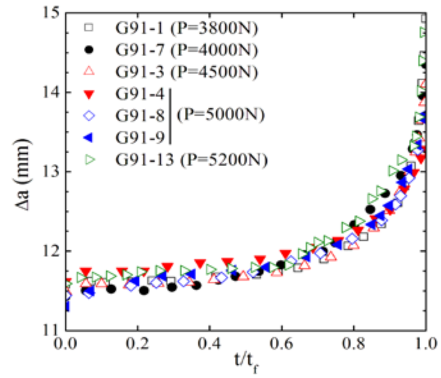


Fig. 7 Creep crack growth test results for base metal

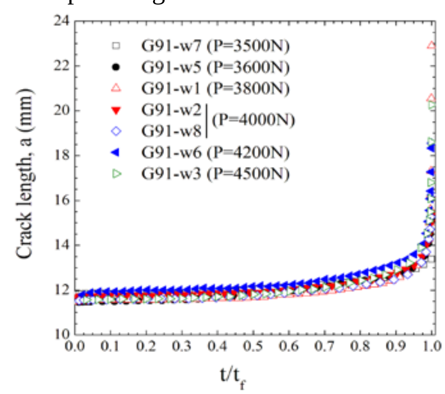


Fig. 8 Creep crack growth test results for weld metal

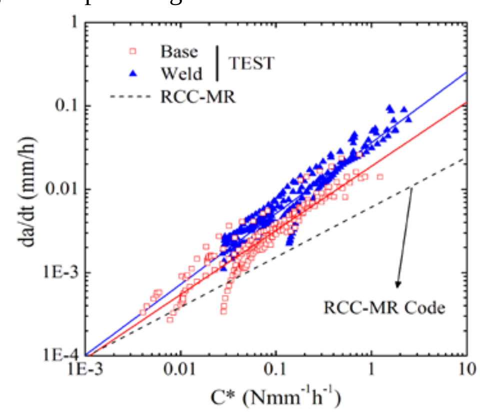


Fig. 9 Comparison of test results RCC-MRx code

### Acknowledgements

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### REFERENCES

[1] Y.I.Kim, et al., Conceptual design report of SFR Demonstration Reactor of 600MWe capacity, KAERI/TR-4598/2012 , 2012.

[2] Lee, H.Y, et al., "*Construction of a High Temperature Grade 91 Sodium Component Test*," Materials Science Forum. Vols. 416-419 528-531,2011.

[3] RCC-MRx, Section III Subsection Z, Appendix A3, 2012, AFCEN, 2012.