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State-of-the-art Sodium Compatibility Research on Austenitic and Ferritic Steels for the Fuel Cladding Materials of a SFR

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1. Introduction

The safety of the sodium-cooled fast reactor (SFR) is directly related to the integrity and safety of the nuclear fuel. Especially, the cladding material of the fuel is contacted with liquid sodium, leading to a degradation of their mechanical properties and hence detrimental effects on the safety of a reactor system. Therefore, in order to design cladding materials and to evaluate their sodium compatibility, it is necessary to review the evaluation results of the sodium compatibility that have been reported until now.

This work is aimed at reviewing the sodium compatibility of the austenitic stainless steel and ferritic steel developed for the cladding or structural materials for a SFR. In particular, corrosion and carburization/ decarburization in liquid sodium, and their effects on the mechanical properties will be reviewed and discussed.

2. Degradation of materials in sodium environment

Considering the structural integrity of the austenitic and ferritic steels as the components of a SFR, the main concerns are the material degradation caused by a corrosion and carburization/decarburization in a sodium environment. The main factors that influence the material degradation are the chemical composition and thermo-physical condition of the material, mechanical stress of the material, temperature and flow rate of the liquid sodium, oxygen and carbon concentration as impurities in the liquid sodium, exposure time, and dissimilar materials in contact with the sodium environment.

3. Sodium compatibility of austenitic stainless steel in sodium environment

3.1. Corrosion of austenitic stainless steel

Two predominant corrosion processes applicable to austenitic stainless steel in liquid sodium are a selective leaching and a steady state corrosion. The presence of oxygen even at low levels causes a selective leaching of chromium by forming a ternary $NaCrO_2$ phase [1-3]. For a long term behavior, a steady state corrosion becomes the dominant cause of a degradation, governing the corrosion rate of the austenitic stainless steel.

Cr leached during the initial period of a exposure reacts with Na and O_2 in the liquid sodium forming a $NaCrO_2$ deposit layer, and also a Cr and Ni-depleted ferritic layer are formed. The growth of the ferritic layer is controlled by a diffusion process for a steady state corrosion. The schematic illustration of this process is presented in Fig. 1.

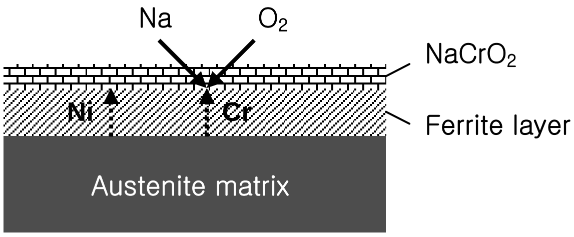

Fig. 1. Schematic illustration of corrosion process of austenitic stainless steel in liquid sodium environment.

Thorley and Tyzack [4] derived the corrosion rate at a steady state in a fast flowing sodium (~7.62 m/s) as written by,

$$\log S = 2.44 + 1.5\log(C_O) - 18{,}000/(2.3RT) \qquad (1),$$

where S is the rate of metal loss (mil/yr), C_O the oxygen content (ppm) and T represents absolute temperature of sodium (K). In a lower sodium velocity, Chopra et al. [5] also reported the corrosion rate as a function of the temperature and the oxygen content in the sodium as shown in Fig. 2. The corrosion behavior of various types of austenitic stainless steel such as 316L [1, 3], 316FR [6], Ti-stabilized No. 1.4970 [7] and 304 [1, 8] were also reported.

3.2. Carburization/decarburization of austenitic stainless steel

The mechanical properties of a fuel cladding material can also be changed as the carbon content of the material surface decreases or increases in a sodium environment depending on the carbon content of the base material and the carbon activity in the liquid sodium. In general, austenitic stainless steels such as type 304 and 316 have been reported to be carburized at a specific carbon activity for the normal operation condition of a reactor below 900 K [9] as depicted in Fig. 3. Due the carburization in the surface of the steel in a sodium environment, the tensile strength was found to increase but the total elongation and ductility to decrease with an the increase in the carbon concentration of the steel.

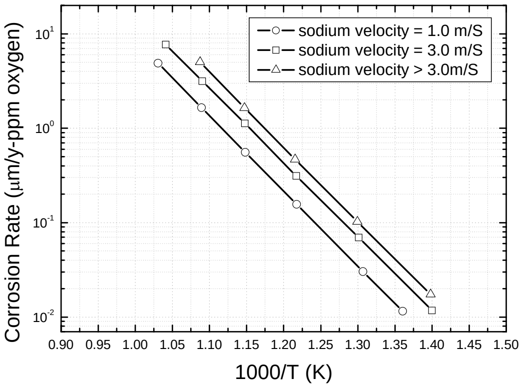

Fig. 2. Effect of sodium velocity on the corrosion rate of Type 316 stainless steel in flowing sodium [1, 3].

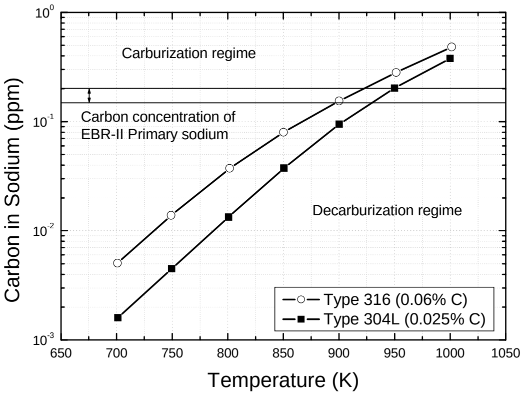

Fig. 3. Carburization-decarburization regimes for type 316 and 304 stainless steel relative to temperature and carbon concentration in sodium reported in EBR-II [9].

4. Sodium compatibility of ferritic steel in sodium environment

4.1. Corrosion of ferritic steel

The ferritic steel exposed in liquid sodium suffered not only a leaching of Cr but also an insertion of Ni from sodium, especially in contact with the austenitic stainless steel. Consequently, the austenitic layer was formed on the steel surface with deposited $NaCrO_2$ and hence a weight gain occurred. Asayama et al. [6] reported that the corrosion rate of Fe-2¼Cr-1Mo steel is similar to that of type 316 steel by

$$\log S = 0.85 + 1.5\log(C_O) - 3{,}900/T \qquad (2),$$

which is valid for oxygen levels of 5 to 25 ppm and temperatures of 673 to 923 K. They also found that the corrosion rate of modified Fe-9Cr-1Mo steel is lower than that of Fe-2¼Cr-1Mo steel by an order of a magnitude.

4.2. Carburization/decarburization of ferritic steel

Fe-2¼Cr-1Mo steel with an inherently high carbon activity tended to be decarburized while Fe-9Cr-1Mo steel tended to be carburized or decarburized a little depending on the temperature and carbon activity in the sodium at the normal operating condition [5, 6, 10]. It was also report that decarburization of Fe-2¼Cr-1Mo steel led to a decrease in the ultimate yield strength of the steels.

5. Conclusion

It was found that the austenitic stainless steel and ferritic steel do not suffer any local corrosion but general corrosion behaviors such as a wall thinning, metal loss, degraded surface layer and $NaCrO_2$ deposit. The mechanical properties were also found to be changed with a change in the carbon content of the steel from its designed value due to a carburization/ decarburization. To design and develop new cladding materials, therefore, the sodium compatibility of the materials should be evaluated well for the integrity and safety of the nuclear fuel in a SFR.

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