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Characterization of Factors Affecting IASCC of Stainless Steels for Core Internals in PWR

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1. Introduction

Stress corrosion cracking (SCC) of reactor core internals resulting from the material degradation and the water chemistry change due to irradiation has been reported in a fuel rod in 1960's, a control rod in 1970's and a baffle former bolt in recent years of light water reactors (LWR). Under the circumstances, a lot of works have been performed on this irradiation-assisted SCC (IASCC) in boiling water reactors (BWR) [1, 2]. Recent efforts have also been focused on IASCC in pressurized water reactors (PWR), but the mechanism in PWR is not fully understood yet as compared with that in BWR due to a lack of data from laboratories and fields. Therefore it is strongly needed to review and analyze recent researches of IASCC in PWR as well as BWR for establishing a proactive management technology for IASCC of core internals in Korean PWR.

2. State-of-the-art research on IASCC of core internals in PWR

Since 2000 the recent works on IASCC of core internals have been performed in a primary water chemistry of PWR for irradiated 304SS and 316SS specimens of baffle-former bolts, baffle-former plates and flux thimble tubes removed from many PWRs.

Suzuki et al. [3] reported slow strain rate test (SSRT) results from cold worked (CW) 316SS tube and solution annealed (SA) 304SS bar specimens in the primary water chemistry of PWR at 290, 325 and 340 ^oC, irradiated to 3×10^{22} n/cm^2 in PWR. Conermann et al. [4] performed SSRT for CW 316SS and 347SS bolts and 304SS lock bar irradiated to 1.4×10^{22} n/cm^2 from 3 units of U.S. PWR. Furutani et al. [5] also carried out SSRT for CW 316SS thimble tube as a function of dissolved hydrogen (DH), irradiated to 5×10^{22} n/cm^2 from Japanese PWR. Busby and Was [6] conducted a constant elongation rate test (CERT) for 304SS and 316SS specimens, proton-irradiated to 3.7×10^{21} n/cm^2 at 360 ^oC. Massoud et al. [7] also reported SSRT results from SA 304L baffle plate and CW 316 baffle bolt, irradiated to 3.3×10^{21} n/cm^2 in Research Institute of Atomic Reactors (RIAR, Russia).

On the other hand, Takakura et al. [8] carried out a constant load test to find the relationship between the applied stress and time-to-failure of IASCC for the baffle-former bolts and flux thimble tubes removed from Japanese PWR. Freyer et al. [9] also performed an O-ring test to evaluate the effect of the applied stress on time-to-failure of IASCC for CW 316SS thimble tubes from Beaver Valley 1, H.B. Robinson 2 and Ringhals 2.

3. Characterization of factors affecting IASCC of core internals in PWR

3.1 Irradiation effect on SCC and its threshold

Fig. 1 presents the irradiation effect on SCC of CW 316SS, SA 304SS and 347SS specimens measured in primary water environments of PWR at various temperatures. %IGSCC is the well known indicative of the susceptibility of materials to IASCC. %IGSCC increases exponentially with increase of the neutron fluence in dpa (displacement per atom). It seems that there is no remarkable difference of IASCC susceptibility among various materials reported by many researchers [3-7]. The prominent effects of temperature on IASCC susceptibility are also observed in Fig. 1.

Fig. 2 exhibits the temperature dependence of the threshold fluence of IASCC determined from Fig. 1. It is obvious that the threshold fluence of IASCC decreases significantly with increasing temperature, that is, the threshold fluences are about 9, 4, 3 and 2 dpa at 290, 320, 325 and 340 ^oC, respectively.

3.2 Effect on water chemistry on IASCC

The absorption of radiations by water leads to its dissociation into several molecules, ions and radicals forming final products, H_2, O_2 and H_2O_2. This radiolysis of the water coolant in LWR increases the corrosion potential of core internals in anodic direction due to the formation of the oxidizing species, O_2 and H_2O_2. In the case of BWR, therefore, there have been many reports of significant effects of water radiolysis on IASCC susceptibility [1, 2]. The primary water chemistry of PWR is much simpler because the water decomposition by radiolysis is effectively suppressed by a hydrogen overpressure condition. However, there is still unsolved issue of the effect of DH on IASCC susceptibility, i.e., increase of %IGSCC with increasing DH [5].

3.3 Effect of microstructure of materials on IASCC

Among the important metallurgical, mechanical and environmental aspects that are believed to play a role in IASCC process, Cr-depletion in grain boundary (GB) due to radiation-induced segregation (RIS) has been focused on as the main cause of IASCC of core internals in BWR [2]. However, it does not fully explain the higher dose failures and those occurring in hydrogenated water environments in PWR where corrosion potentials are far below the critical cracking potential. One of minor alloying elements, S segregation to GB is believed to act synergistically with Cr-depletion, even though it is still necessary to add a caution that general conclusion concerning GB segregation as IASCC mechanism.

3.4 Effect of stress on IASCC and its threshold

The crack initiation of CW 316SS specimens irradiated in various PWRs were measured to find the time-to-failure of IASCC as a function of the applied stress by the O-ring and constant load tests in the primary water chemistry [8, 9]. From the previous results, the threshold stress of IASCC was established in this work. Fig. 3 shows the threshold stress as % of yield strength (YS) of the specimen irradiated at various neutron fluences, and the dependence of the threshold stress, σ_{th}, on the neutron fluence, *dpa*, (dotted line in Fig. 3) is given by,

$$\sigma_{th}/YS_{irradiated}(\%) = 39.0 + 138.3\times\exp(-dpa/16.0) \quad (1).$$

From the results, it can be easily estimated that core internals of PWR irradiated to 2 and 10 dpa would be failed by IASCC with the stress of 161 and 113 %YS, respectively, applied to components due to over-fastening by bolts, over-load by adjacent components, etc.

4. Conclusion

This state-of-the-art report of irradiation effects on SCC of core internals of PWR in the metallurgical, mechanical and environmental aspects is applicable to establish a proactive management technology of IASCC of core internals in Korean PWR.

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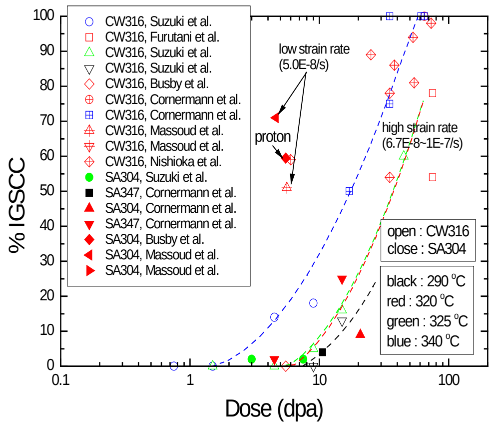

Fig. 1. Plots of %IGSCC vs. dose as a function of temperature measured from CW 316SS, SA 304SS and 347SS by SSRT, reproduced from various literatures [3-7].

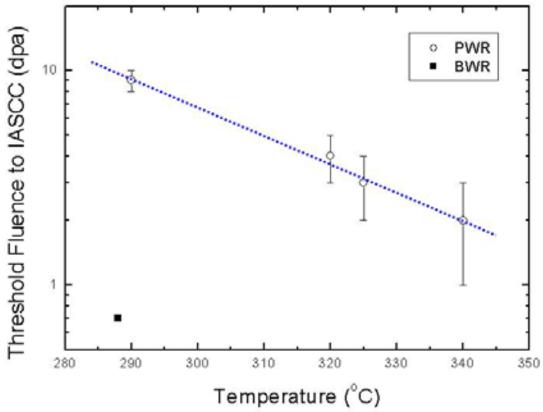

Fig. 2. Threshold fluence to IASCC at various temperatures estimated from Fig. 1.

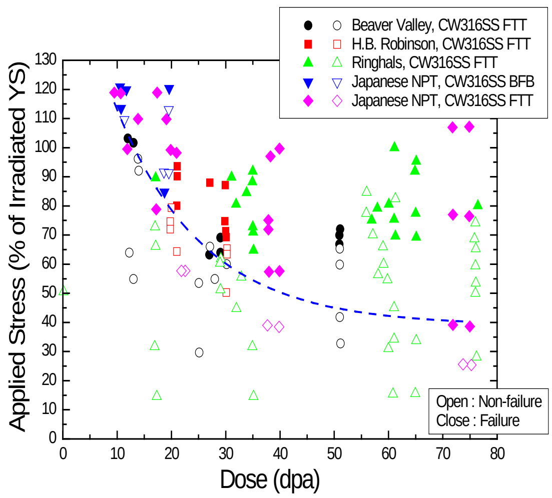

Fig. 3. The relationship between the applied stress and the neutron fluence for CW 316SS irradiated in various PWRs [8, 9]. Open symbol means the specimen failed by IASCC, and close one represents the specimen not failed.