# Factors for IASCC initiation of reactor internal materials of PWRs

Seong Sik Hwang [a *], Min Jae Choi [a], Sung Woo Kim [a], Dong Jin Kim [a]
[a] *Korea Atomic Energy Research Institute, 989-111 Daedeok-daero, Yuseong-gu, Daejeon 34057, Korea*
[*] *Corresponding author: sshwang@kaeri.re.kr*

## 1. Introduction

In order to safely operate the nuclear power plants in Korea that are approaching the end of their design life, a strategy for managing material degradation of components is important. For long term operation of the plants, the integrity of the internal structure of the reactors must be checked. As one of the processes, it is important to determine at what point a cracking defect will occur.

In this paper, degradation trend of foreign plants is reviewed. Materials management research to understand the degradation of the internal structure and prepare a management strategy for long term operation is also discussed. In particular, the factors for crack initiation of stainless steels used as a material for the internal structure of a nuclear reactor are reviewed.

## 2. Degradation trend of foreign plants

In 2010, cracks were found in BFB(Baffle former bolt) at USA power plants such as Surry Units 1 and 2, Ginna, and DC Cook unit 2. The defects were confirmed to be IASCC(Irradiation assisted stress corrosion cracking). For the purpose of suggesting management guidelines of the components, a report MRP-227 titled Inspection & Evaluation (I&E) guidelines for the reactor internals, which contains inspection strategies for reactor internal structures, was published in November 2011 in USA.

In case of French type plants, cracks were first discovered at a plant in 1988, and some more cracks have been reported at other power plants since Tihange unit in 1991 [1].

## 3. Materials management for long term operation

Figure 1 shows the trend of damaged components of nuclear power plants in Korea and around the world. In the 1970s, cracks in steam generator tubes were the main degradation site, and in the 1990s, cracks in nozzles and pipes made of Alloy 600 were the main degradation mechanisms. Entering the 2000s, it can be seen that the IASCC of the internal structure of a nuclear reactor has become a major degradation mechanism subject to management.

One approach to addressing this degradation for long-term operation is to utilize irradiated materials from research reactors exposed to sufficiently high neutron doses.

It is a way to evaluate the life of power plant components in operation by obtaining information on the initiation of cracks using this material.

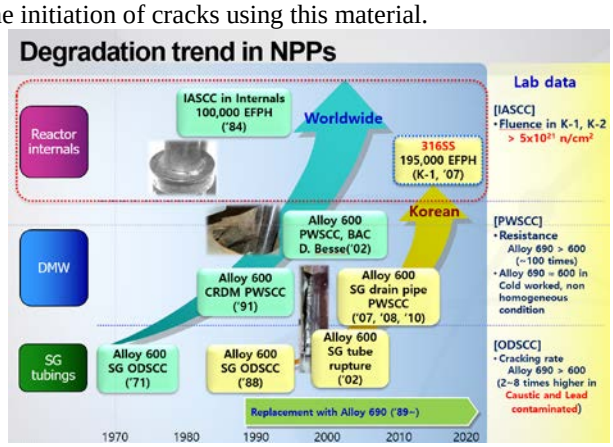


Fig. 1. Trend of damaged parts of major parts of nuclear power plants

Due to differences in operating environments, however, materials irradiated in research reactors may not actually represent materials irradiated in commercial pressurized water reactor (PWR) conditions. Some research related to this is already being conducted at home and abroad [2]. Finding information about crack initiation of these components is key to ensuring the safety of a continuously operating plant. To this end, the following section summarizes the results of investigations on operating conditions that affect IASCC initiation.

## 4. Influence of operating condition on IASCC initiation

### *4.1 Influence of dissolved hydrogen*

A crack initiation sensitivity model established from the analysis results of crack length per area according to the amount of irradiation has been developed [3]. This study showed that the crack initiation sensitivity increased with increasing from 25 cc/kg $H_2$ to 50 cc/kg $H_2$. Although it is not possible to conclude the effect of dissolved hydrogen because the experiment was conducted with only two conditions, it can be said that the IASCC crack initiation sensitivity increased in type 316 stainless steel, as dissolved hydrogen increased. In the case of Ni base alloy, it is different from the result of the above study [3] in that it shows the maximum crack sensitivity under a certain dissolved hydrogen concentration condition (DH of about 30 cc/kg or higher).

In the case of experiments conducted by the CEA laboratory in France[4] as shown Fig.2, no increase in crack growth rate was observed under the condition of 100 cc/kg dissolved hydrogen. Compared to this, the growth rate is significantly reduced to 1/10 level. The results of this study are consistent with the experimental results by Arioka [5] for a limited range of dissolved hydrogen.

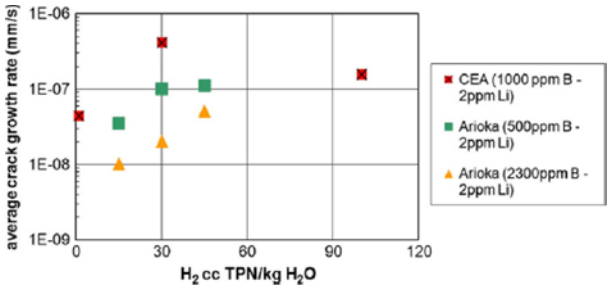


Fig. 2. Relation between dissolved hydrogen and crack growth rate of stainless steels [4]

*4.2 Effect of chemical composition on crack initiation*

According to Stephenson et al. [6], the IASCC susceptibility was very sensitive to the parent alloy composition. The addition of Hf and Si was detrimental, Mo and Ti had little effect, and Ni or Ni + Cr completely lowered the IASCC susceptibility in high-purity model type 304L SS steel. It was also found that the increase in grain boundary Si due to radiation induced segregation (RIS) was well correlated with the increase in sensitivity to IASCC.

*4.3 IASCC Initiation Stress and Critical Irradiation*

Takakura showed the dependence of the SCC initiation stress on the fluence and applied stress[7]. He showed the results of the SCC initiation test and explained the relationship between the irradiation level and the applied stress. His results clearly indicate that the stress required for SCC generation decreases with increasing irradiation dose. According to a study by Nishioka et al. [8], as shown Fig. 3, crack initiation sensitivity according to the reduction in added stress decreases up to 30 dpa, but crack initiation does not decrease even if the stress decreases under the condition of 30 dpa or more.

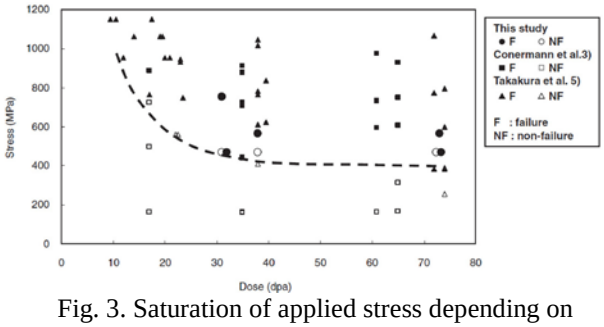


Fig. 3. Saturation of applied stress depending on fluence [8]

## 5. Conclusions

1. Defects found in the BFB were confirmed to be IASCC in USA and French NPPs.
2. IASCC of the internal structure of a nuclear reactor has become a major degradation mechanism subject to management for long term operation.
3. Up to DH concentrations around 50 cc/kg, the IASCC crack initiation sensitivity increases with increasing DH. Regarding higher concentration conditions, researchers have different opinions.
4. IASCC susceptibility was very sensitive to the parent alloy composition. The addition of Hf and Si was detrimental, Mo and Ti had little effect, and Ni or Ni + Cr completely lowered the IASCC susceptibility. Grain boundary Si was well correlated with the increase in sensitivity to IASCC.
5. Stress required for SCC initiation decreases with increasing irradiation dose.

## ACKNOWLEDGEMENTS

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