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IASCC Susceptibility of Core Baffle Former Barrel Structure in Korean Pressurized Water Reactor

Jun-Seog Yang[1]

[1] Principal Research Engineer, Central Research Institutive, KHNP, Korea

ABSTRACT

As nuclear power plants age and the cumulative neutron dose increase, the degradation of reactor vessel (RV) internal components become more likely and potentially more severe. Potential cause of material degradation is irradiation assisted stress corrosion cracking (IASCC) that may lead to failure of core baffle former barrel (BFB) structure in RV internal. For maintaining the healthy fleet and ensuring a continued functionality of RV internal components, it is necessary to evaluate the selected components that are judged to be susceptible to the IASCC. In this paper, Irradiation-induced degradations on the core BFB structure of the representative PWR in Korea are predicted up to 60 years with the constitutive model for a typical irradiated austenite stainless steel. The most highly irradiated locations in BFB structure are occurred at the internal corners of the baffle plates. The temperatures of the baffle plates can be significantly higher than the coolant temperature due to the intense gamma heating rates near the core. The swelling in the BFB structure is concentrated in this high-temperature corner. The highest swelling rates occur where the former plates contact the baffle plates. Bolts at all former levels where is nearest to the core experience very rapid stress relaxation and indicate the potential of the IASCC.

INTRODUCTION

The reactor vessel (RV) internal components in pressurized water reactor (PWR) are very important structure to support reactor core. Materials of RV internal components are primarily made of austenitic stainless steels, which have relatively high strength, ductility and fracture toughness. When these materials are placed in a PWR environment, such as prolonged neutron irradiation, elevated temperature, and exposure to the primary coolant, these properties undergo changes [1]. They increase the yield and tensile strength, decrease ductility and fracture toughness, change the potential volume due to void formation and sometimes experience irradiation assisted stress corrosion cracking (IASCC). As nuclear power plants age and the cumulative neutron dose increase, the degradation of RV internal components become more likely and potentially more severe. IASCC is potential cause of material degradation that may lead to failure of core internal structure in PWR [2]. In the present paper, the aging degradations in the baffle-former-barrel (BFB) structure of the representative PWR in Korea are predicted up to 60 years based on functionality analysis with conservative core loading, heat transfer, and mechanical preloads. From these analyses, it is used to determine when and where irradiation aging may occur.

FUNCTIONALITY ANALLYSIS

BFB structure

Among the RV internal components which are affected by radiations, BFB structure with bolted connections is particularly important in terms of assessing their functionality. Fig. 1 shows BFB structure surrounds the reactor core. BFB structure is made up of vertical baffle plates and seven vertically spaced horizontal former plates. The baffle plates are secured to the cylindrical core barrel through a series of

horizontal or former plates. As shown in Fig. 1, the baffle plates are bolted to the formers by the baffle-former bolts, and the formers are attached to the inside surface of the core barrel by barrel-former bolts. And, at selected corners, some of the baffle plates are bolted to each other by baffle-edge bolts. The region between the baffle plates and core barrel is called as the bypass region. Coolant flow bypasses the core and flows into this region serving as a means to cool the components. The BFB structure is primarily subjected to core heating and dose rates.

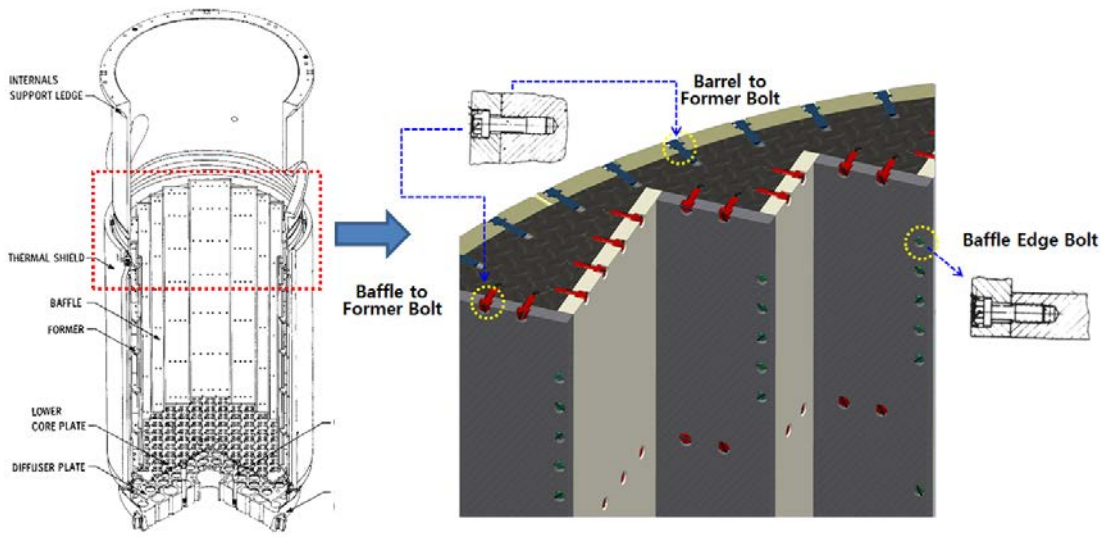

Figure 1. BFB structure and locations of the bolts in BFB structure

Finite Element Model

Fig. 2 shows the finite element model and the identification of BFB bolts of BFB structure, which consists of a one-eighth symmetric solid model using mostly 20-noded hexahedral elements. Two models are developed: a thermal model used to calculate temperatures and a structural model used to calculate displacement and stresses. All components are meshed with solid elements except for the baffle-edge bolts, which are meshed using beam elements in the structural model and thermal link elements in the thermal model. Additionally, surface elements are used for applying the heat transfer coefficients. Contact elements are used in both the thermal and structural models.

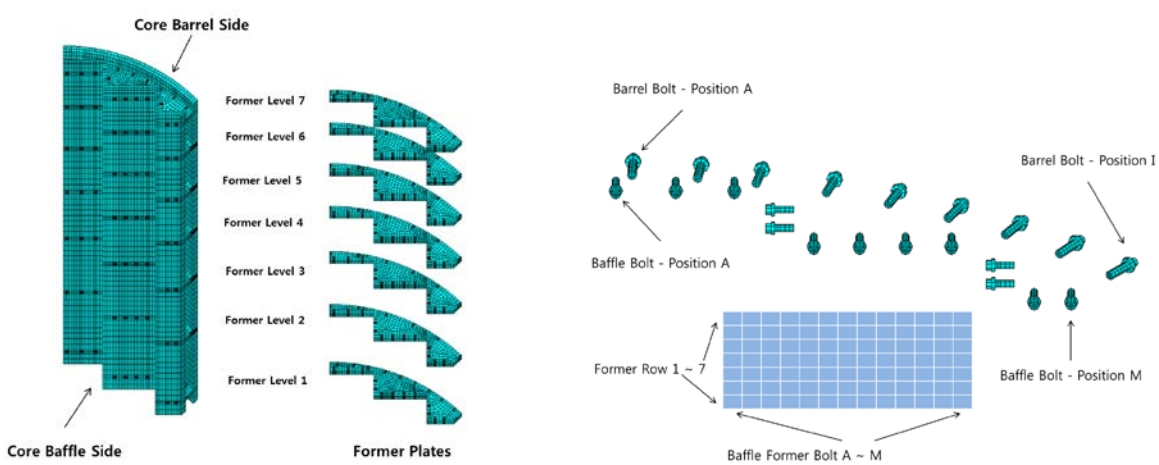

Figure 2. Finite element model of BFB structure and bolting pattern at each former row

Material Properties

The RV internal components require a material constitutive model that allows the material behavior to vary with temperature, strain rate and neutron flux. Constitutive models for typical irradiated stainless steel of RV internals are developed by EPRI, modified by ANATECH Corporation, and implemented with the "USERMAT" subroutine [3,4]. Material input to ANSYS computer code for USERMAT is provided in the form of ANSYS user material variables, which identify the material type (304, 304L, 316, or 316L), cold work ratio, reference (stress-free) temperature, dose and dose rate control flags, and irradiation swelling and creep activation flags. All BFB structure items except for bolts are Type 304 stainless steel and bolts are Type 316 stainless steel. The irradiated material properties for Type 304 and 316 stainless steels are determined using an ANSYS-based subroutine "USERMAT".

Radiation Analysis

Three-dimensional spatial distributions of neutron flux and nuclear heat generation rate in BFB structure are generated as input to the thermal and structural analysis. Neutron flux has been related to the displacement per atom (dpa) and gamma heating is heat generation rate (W/cm^3). Neutron flux and gamma heating in BFB structure are generated for two representative core loading patterns, which are high leakage loading and low leakage loading. For each core loading pattern, three different boron concentrations are considered as the beginning, middle, and end of cycle (BOC, MOC, and EOC) for 12-month fuel cycle. Fig. 3 through 6 shows the distributions of heat generation rate (W/cm^3) and neutron flux (dpa/s) in the BOC, MOC and EOC with respect to each core loading pattern. The most highly gamma heated and irradiated areas are closed to the core such as internal corners of the baffle plates as shown in Fig. 3 through 6. Although the swelling is localized, it can cause significant distortion in the surrounding and bolts.

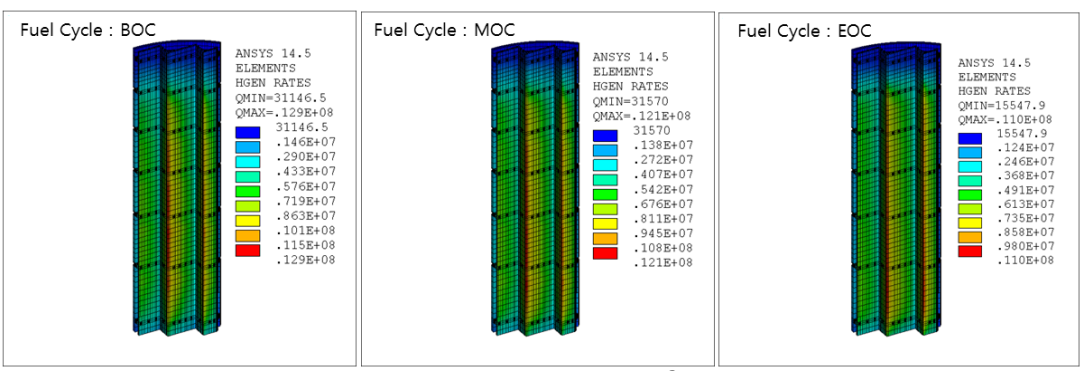

Figure 3. Distributions of heat generation rate (W/cm^3) on high leakage core loading.

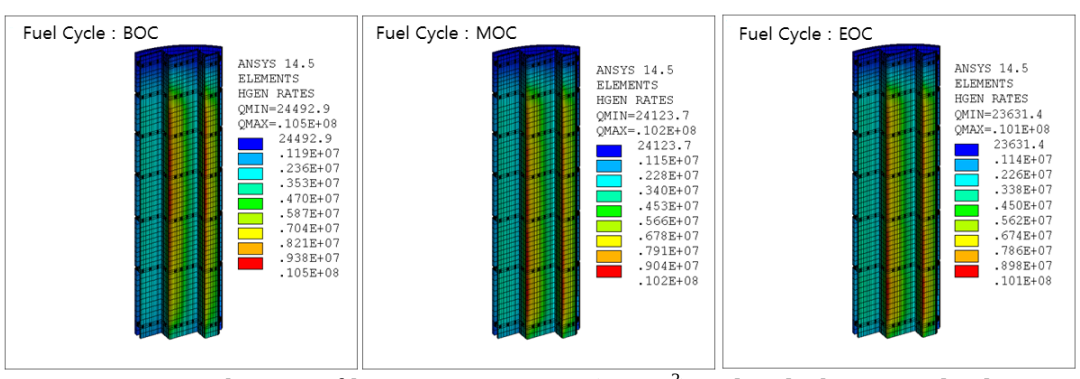

Figure 4. Distributions of heat generation rate (W/cm^3) on low leakage core loading.

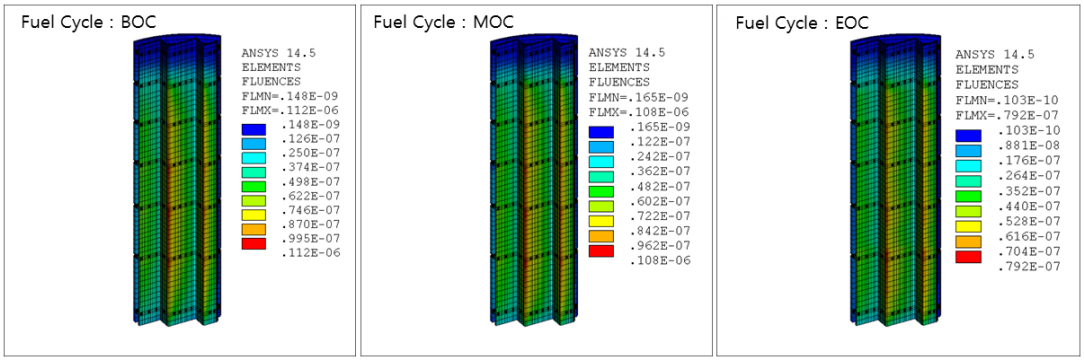

Figure 5. Distributions of neutron flux (dpa/s) on high leakage core loading.

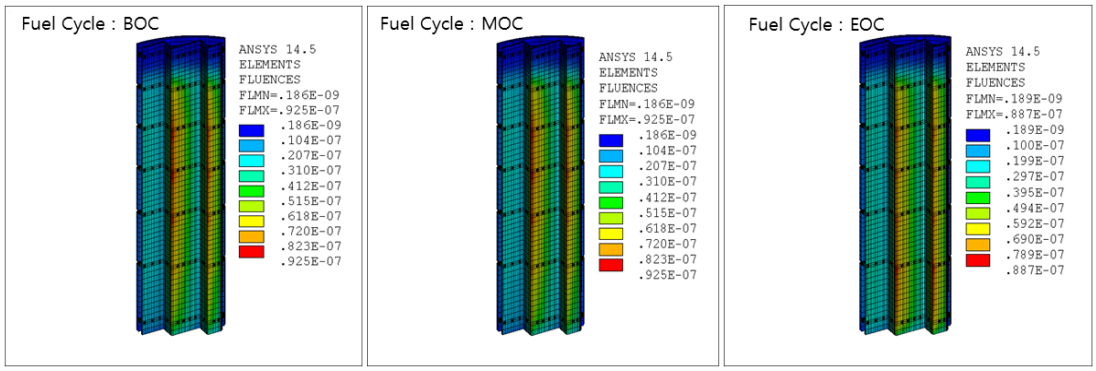

Figure 6. Distributions of neutron flux (dpa/s) on low leakage core loading.

Structural Analysis

The BFB structure is subjected to 60 years of core loading consisting of temperature and dose. The functionality analysis considers the combined thermal and irradiation loadings that represent a 60-year simulation to determine the material degradation on the BFB structure [5]. The thermal and irradiation loadings are generated to apply the temperature and dose rate for two core loadings at BOC, MOC, and EOC conditions, respectively. The important factor that determines the response of BFB structure is the material properties that represent susceptibility to irradiation-induced degradation of mechanical and physical properties. The functionality analysis for BFB structure employs the material behavior predicted from ANSYS-based subroutine “USERMAT”. The structural analysis loading steps consist of a series of preloads followed by the core loading sequence. The core loading follows and is applied in cycles. Each cycle consists of the sequential application of temperature distributions and dose rates for four-month BOC, four-month MOC, and four-month EOC, followed by a return to zero power. The high leakage loading is applicable for the first 30 years, while the low leakage loading is applicable for fuel cycles 31 through 60.

RESULTS AND DISCUSSIONS

Potential Damage Criteria

Under IASCC conditions, potential failure is assumed to occur when the component becomes susceptible to stress corrosion cracking after a certain period of irradiation. Potential for stress corrosion cracking is defined by an IASCC susceptibility stress [6]. The effective stress at which IASCC susceptibility initiates (σ_{IASCC}) is defined for use in functionality analysis as follows :

$$\sigma_{IASCC} = S(d)\sigma_y(T, d, \varepsilon_{eff}^{pl}) \tag{1}$$

where, d is dose, the IASCC yield stress multiplication factor, $S(d)$, is defined as shown in Fig. 7, which relates the applied stress to the cumulative irradiation dose after which the material becomes susceptible to stress corrosion cracking. T and ε_{eff}^{pl} denote the temperature and effective plastic strain, respectively. Initiation of IASCC requires high local tensile stresses. Locations where the local tensile stress exceeds the IASCC stress are potential crack sites. For convenience in analysis, the ratio of the effective stress to the IASCC susceptibility stress is reported as the IASCC susceptibility ratio. When this ratio is less than one, the location is not susceptible to IASCC. To indicate the potential for cracking, the IASCC susceptibility ratio must be greater than one, and the effective stress used to calculate the ratio must be tensile.

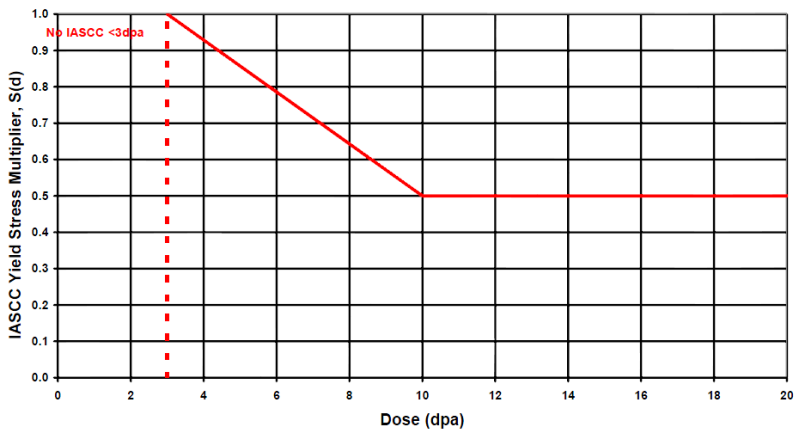

Figure 7. Variation of IASCC multiplication factor with irradiation dose [6]

Bolts

Fig. 8 shows time history plots of the effective stress in the head and shank of the baffle-former bolts at each former row. All of the highest stress locations also occur at external baffle-former plate corner closest to the core. As shown in Fig. 8, bolts in former row (1 through 6) experience very rapid and considerable stress relaxation. The trend then changes at around 7 to 12 years where the IASCC ratio decreases due to stress relaxation in these bolts. Later in years, when relative distortion of the component items exerts more stress on those bolts, the IASCC ratio increases again. Most of bolts show an increasing trend of stress with time, which suggests an increasing bolt load with time. Fig. 9 shows time history plots of IASCC susceptibility ratio in the head and shank of the baffle-former bolts at each former row. At year 32 bolt 5E has reached IASCC susceptibility (1or greater). Baffle-former bolt 4E has been reached susceptibility at year 40. Baffle-former bolt 3E and 6E have been reached susceptibility in year 50. Baffle-former bolt 2E has been nearly reached IASCC susceptibility at 60 year. In the top former row (bolt 7E), IASCC does not occur since bolts at these elevations do not reach the dose threshold (currently 3dpa) for IASCC to be a concern. For the barrel-former bolts, all bolts show no indication of susceptibility. Fig. 9 shows that bolts with the highest void swelling are at the internal baffle plate corner closest to the core. These locations are consistent with the maximum temperature and dose rate location in each former row. The maximum swelling calculated for the baffle-former bolt nearest to the region of swelling is approximately two percent for 60 years. Therefore, swelling in the bolts will not likely be a reasonable indicator of swelling deformation in the BFB structure.

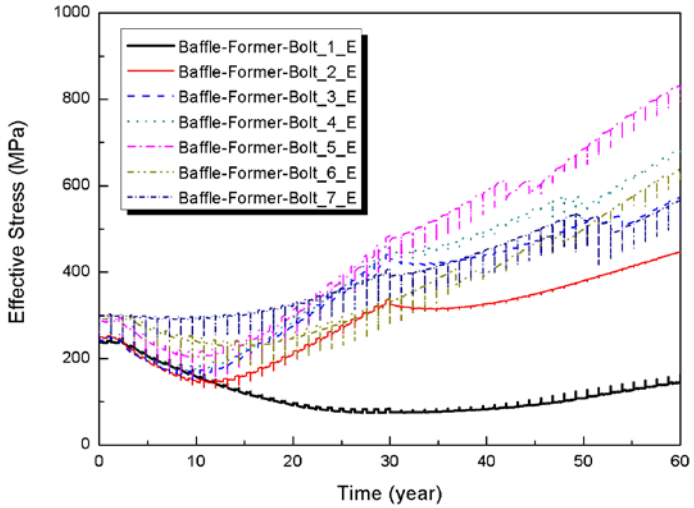

Figure 8. Effective stresses on the baffle-former bolts at each former row

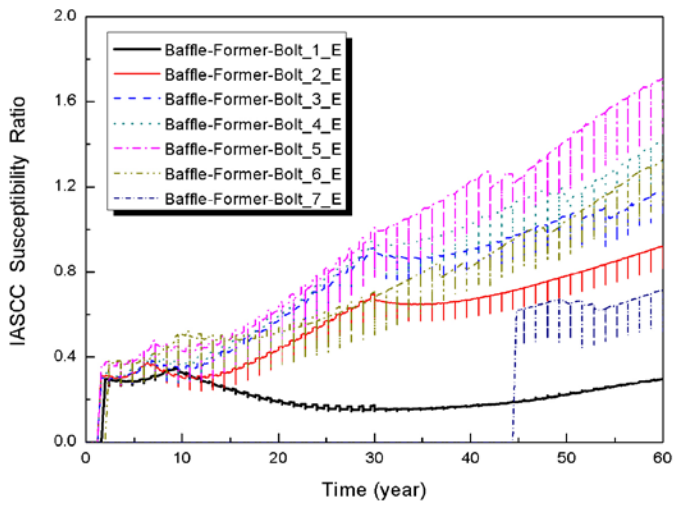

Figure 9. IASCC susceptibility ratio on the baffle-former bolts at each former row

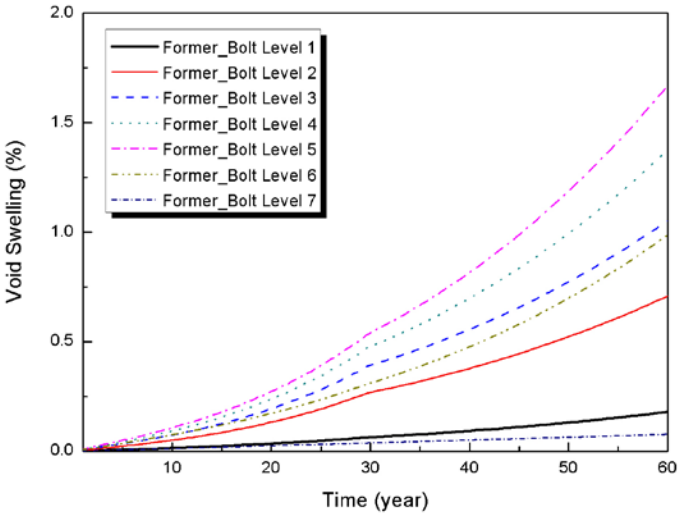

Figure 10. Void swelling on the baffle-former bolts at each former row

Baffle plates, former plates, and core barrel

Fig. 11 shows stress contours for the bypass flow side of baffle plates. The maximum stresses occur at bearing surfaces under the bolt heads. These locations experience compressive stresses and are due to contact between the mating surfaces of the bolt head and bolt hole through the plates. IASCC susceptibility ratios are well below one over the bulk of the baffle plate surfaces. Local peaks in the effective stresses occur at locations where the swelling former plates contact the baffle plates. The peak values of the baffle plates occur at the elevation of former row 6. IASCC susceptibility ratio of the baffle plates is reached in year 45. While the IASCC susceptibility ratio is nearly exceed in year 45 the contact stresses at these locations are compressive; thus, the locations are not identified as potential IASCC initiation site. Fig. 12 shows the maximum stress of the former plates occurs in a threaded baffle-former bolt hole at former level 6. While the former plates show localized indications of IASCC susceptibility due to thread-to-former coupling the contact stress at this location is compressive; thus, the former plates are not as an issue of IASCC. In the present paper, the thread to former interface is not modeled in detail and the results are approximately. The bolts show greater susceptibility to IASCC at the head-shank interface than at the thread-former interface. The core barrel has very low stresses and shows little susceptibility to IASCC. The swelling in the BFB structure is concentrated in this high temperature corner as mentioned before. Fig. 13 shows void swelling time history plots of the plates, and the highest swelling rates occur in the first 30 years, when the high leakage loading patterns are assumed. Although the swelling is highly localized, the calculated peak swelling is about 6 percent at year 30, 12 percent at year 45, and 18 percent at year 60. The swelling also causes significant distortion of the baffle-former and baffle-edge bolts. The bolts in the adjacent baffle plates are connected to a common former plate.

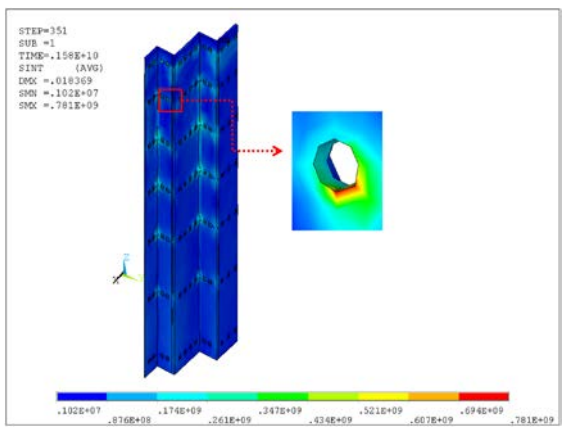

Figure 11. Stress contours on the baffle plates

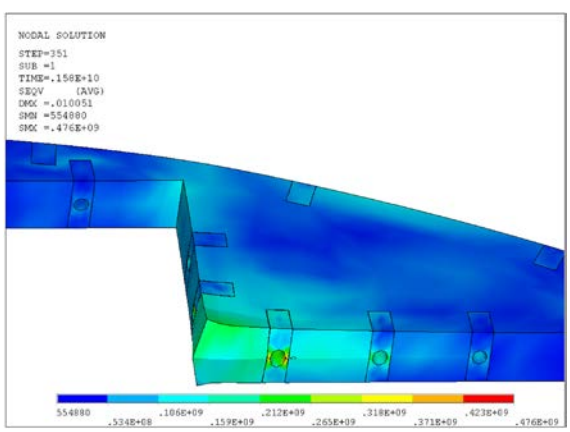

Figure 12. Stress contours on the former plate at former level 6

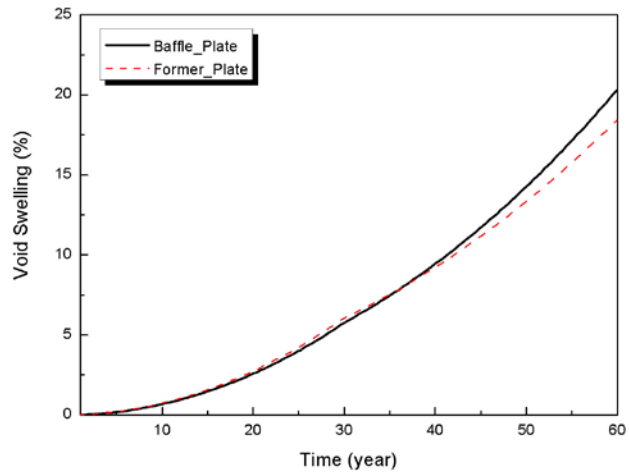

Figure 13. Void swelling time history plots of the baffle and former plates

CONCLUSION

The irradiation-induced degradations of the core BFB structure of the oldest PWR in Korea are predicted up to 60 years using the functionality analysis with the material model of irradiated austenite stainless steel. The most highly irradiated locations in the core baffle and barrel structure are occurred at the internal corners of the baffle plates. The temperatures of the baffle and former plates can be significantly higher than the coolant temperature due to the intense gamma heating rates near the core. The highest temperatures are calculated for the first 30 years due to the higher gamma heating rates associated with the assumed high leakage core power distributions. The peak temperature occurs in former row 5 at the corner closer to the core. The swelling in the core BFB structure is concentrated in this high-temperature corner. The highest swelling rates occur in the first 30 years and higher swelling rates are indicated where the former plates contact the baffle plates. Although the swelling is highly localized, the calculated peak swelling in the plates is 20 percent at year 60. While the core barrel experiences very little swelling, the maximum swelling calculated for the baffle-former bolt is approximately 1.7 percent at year 60. Irradiation-induced stress relaxation is very significant for the baffle-former bolts. The baffle-former blots at former row (2 through 6) experience very rapid and considerable stress relaxation. The trend then changes at around 7 to 12 years where the IASCC ratio decreases due to stress relaxation in these bolts. Later in years, when relative distortion of the plates exerts more stress on those bolts, the IASCC susceptibility ratio increases again. With respect to IASCC cracking potential, the baffle-former bolts at former row (3 through 5) have reached IASCC susceptibility ratio (1 or greater) at 32 ~ 50 years. The baffle-edge bolts between former row 5 and 7 also indicate possible susceptibility at 30 ~ 40 years. For the barrel-former bolts, only one bolt indicates IASCC susceptibility beginning at year 46. All other barrel-former bolts are not susceptibility of IASCC. Bolts also show greater susceptibility to IASCC at the head-shank interface than at the thread-former interface. IASCC susceptibility ratio is well below one over the bulk of the baffle plate surface. The core barrel has very low stresses and shows little susceptibility to IASCC.

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