*Transactions of the Korean Nuclear Society Spring Meeting*
*Jeju, Korea, May 6-8, 2015*

# Irradiation-Induced Degradation Effects on Baffle-Former-Barrel Assembly of Reactor Vessel Internal

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## 1. Introduction

In pressurized water reactor power plant, reactor vessel internal (RVI) is designed to hold a core, fix fuel assemblies in their positions, distribute coolant flow in a reactor, support and protect control rods, decrease the neutron effects onto reactor pressure vessel. Components of RVI are consisted of austenitic stainless steel materials which are known to have good strength, ductility, toughness and corrosion resistance. However, when placed in a PWR environment, these properties undergo changes due to long-term exposure to neutron irradiation, high temperature, reactor water, and loading. Especially irradiation aging in the reactor vessel internal components is characterized by a decrease in ductility and fracture toughness, increase in yield and ultimate tensile strength, and potential volume changes due to void formation. Therefore, the demonstration that the effects of degradation in the components of RVI are adequately managed is essential for maintaining a healthy fleet and ensuring the continued functionality of the reactor internals for the long term reactor operation beyond design life. Also, it is very important to determine when and where irradiation susceptibility may occur for the continued operation. This paper introduces the aging management strategy for the pressurized water reactor internal and investigates the effects of the irradiation-induced degradation of the baffle-former-barrel (BFB) assembly of the pressurized water reactor operating in Korea.

## 2. Analysis Procedure, Methodology and Results

### *2.1 Aging Management Strategy*

Fig. 1 illustrates conceptually the process of the aging management [1, 2, 3]. Fig. 1 shows the links between the categorization based on screening criteria, functionality assessment, the aging management strategy, and the inspection & Examination guidelines. As shown in Fig. 1, the component identified as C components are required to perform the functionality analysis which includes the evaluation of the selected components that are judged to be susceptible to irradiation-induced degradation of mechanical and physical properties. This analysis is used to determine when and where irradiation susceptibility may occur for the continued operation. This process permitted further categorization of the pressurized water reactor internals into functional groups: Primary, Expansion, Existing Programs, and No Additional Measures groups with appropriate recommendations to support aging management program. Through this process, a team of experts evaluates the reactor internals and made the appropriate recommendations for aging management actions specific to each component.

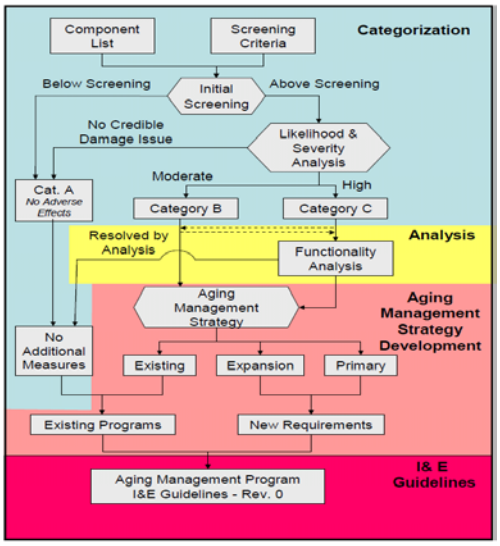


Fig. 1 Strategy for Aging Management Guideline.

### *2.2 Finite Analysis Model*

For the development of aging management and inspection strategies in the pressurized water reactor internal, the functionality analysis is performed on a number of components identified as Category C, which include baffle-former bolts, barrel-former bolts, upper and lower core barrels, and associated welds shown in Fig. 2.

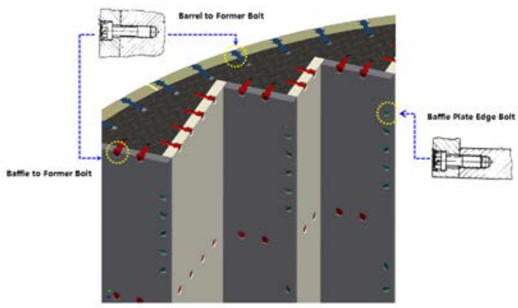


Fig. 2. BFB Assembly with Baffle, Edge and Former Bolts.

Finite element models are developed for each of the components, which are indicated in Fig 3. The analysis simulates selected components by subjecting them to representative core thermal loading and irradiation loading.

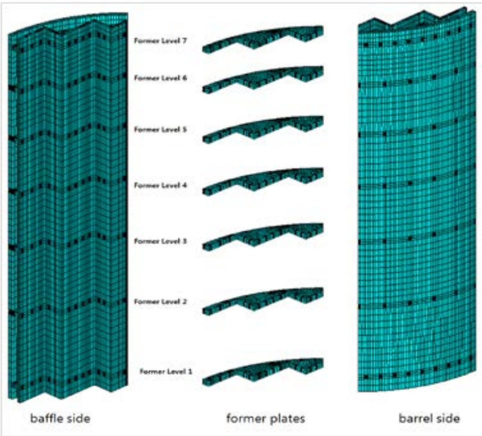


Fig. 3 Finite Element Analysis Model for BFB

*2.3 Functionality Analysis*

Material models used in commercial-purpose finite element codes are inadequate for the analysis of aging degradations of stainless steel components in a pressurized water reactor environment. In this paper, functionality analysis is performed using ANSYS [4] based subroutine USERMAT [5, 6], which utilizes material properties and constitutive models that are a function of irradiation dose and temperature for irradiated austenite stainless steels. For determining the radiation environment for the pressurized reactor vessel internals, irradiation analysis was performed that three-dimensional spatial distributions of neutron exposure in terms of displacement per atom (dpa) and of nuclear heat deposition rates within the RVI structure are generated as input of thermal and structural evaluations.

The structural analysis considers the combined thermal and irradiation loadings that represent a 60-year simulation to determine the aging degradation on the BFB assembly. The thermal loading and irradiation loading consist of two core designs, high leakage and low leakage loading patterns. The thermal loading is generated by applying the heat transfer boundary coefficients and bulk temperature on the surface exposed to the reactor coolant and, also applying the heat generation rates. The irradiation loadings are generated to apply the dose rate. For each core loading pattern, three sets of temperatures and irradiation doses determined with respect to the beginning of cycle (BOC), the middle of cycle (MOC) and the end of cycle (EOC) respectively. The ANSYS model for the BFB assembly consists of a one-eighth symmetry solid model using mostly 20-noded hexahedral elements.

*2.4 Results and Discussion*

The structural analysis loading steps consist of a series of preloads followed by the core loading sequence. The preload begins with the pre-tensioning of bolts. Next, operating condition pressure is applied. Finally the BFB assembly is subject to a uniform temperature of zero power per project direction. The core loading follows and is applied in cycles. Each cycle consists of the sequential application of temperature distributions and dose rates for six-month BOC, six-month MOC, and six-month EOC, followed by a return to zero power. Fig. 4 shows the load sequence for one fuel cycle. Cycles 1 through 20 represent the high leakage core pattern. Cycles 21 through 40 represent the low leakage core pattern.

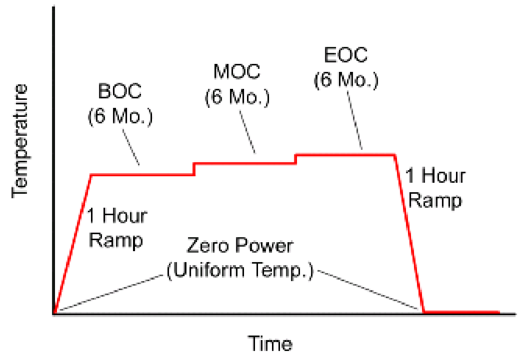


Fig. 4 Load Sequence for one fuel cycle.

Fig. 5 and Fig. 6 shows contour plots of heat generation rate and dose rate distribution each, where the most highly irradiated areas are closed to the core such as internal corners of the baffle plates. Coincidentally, the largest distortion occurs between these baffle plates. The distortion is carried by the bolts holding these plates: baffle-former bolts and baffle-edge bolts. Fig. 7 and Fig. 8 show the behavior of the baffle plates and bolts due to the irradiation loading condition.

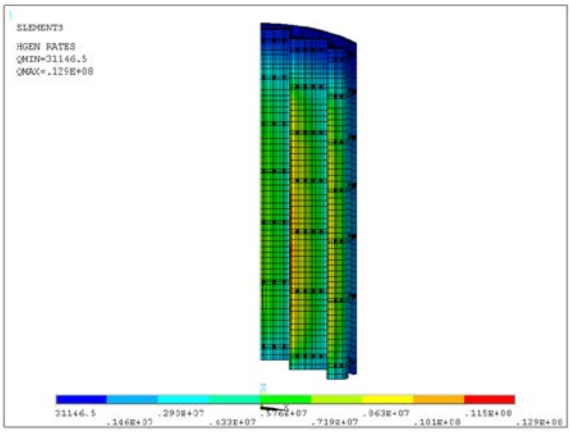


Fig. 6 Contour Plot of Heat Generation Rate (W/㎤)

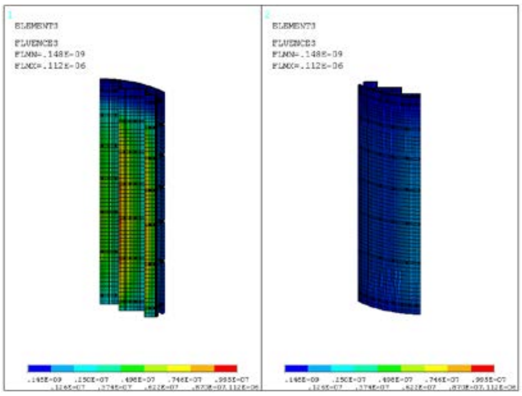

Fig. 7 Contour Plot of Dose Rate (dpa)

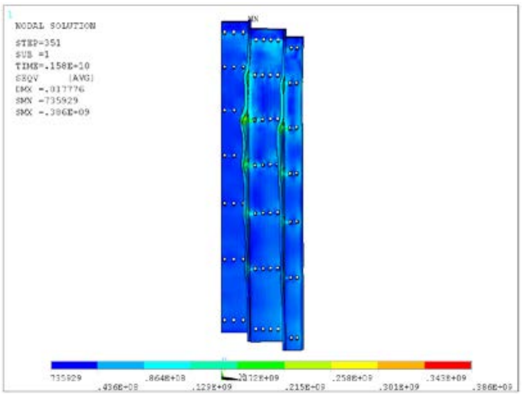

Fig. 8 Distortion and Stress Contour of Baffle Plate

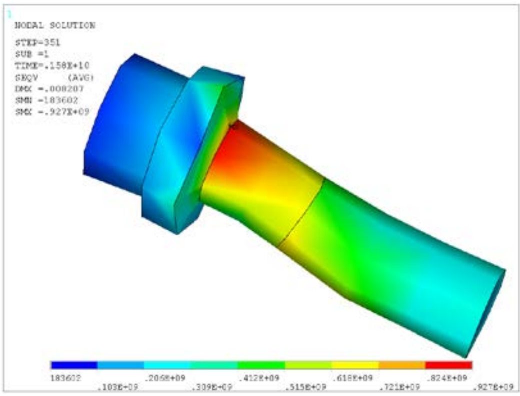

Fig. 9 Distortion and Stress Contour of Bolts

Fig. 10 depicts time history plot of the volumetric void swelling. Each curve shows baffle-former-bolt with the highest void swelling in each former row. The location of the bolts with the highest void swelling is at the internal baffle plate corner closest to the core. These locations are consistent with the maximum temperature and dose rate location in the former low. The maximum void swelling is in the baffle-to-former bolt at former low elevation 5 and its value is 1.5%. Characteristic to each curve is the exponential growth of void swelling with time.

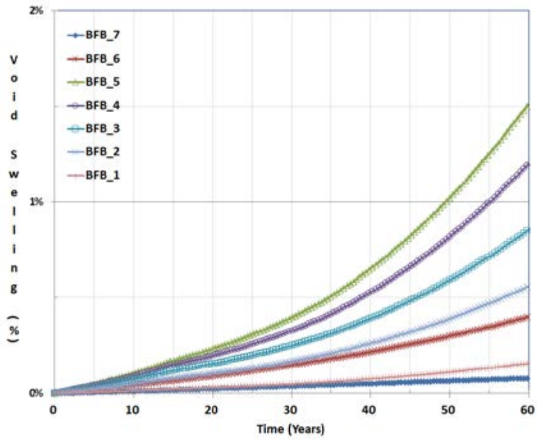


Fig. 10. Time History Plot of Void Swelling (%)

## 3. Conclusions

This paper provides the methodologies and results for the functionality analysis on the baffle-former-barrel assembly. The highest temperatures are yielded from EOC of high leakage loading. The most highly irradiated areas are also closed to the core such as internal corners of the baffle plates. Coincidentally, the largest distortion occurs between these baffle plates. The distortion is carried by the bolts holding these plates. Characteristic to void swelling is the exponential growth with time. At 30 years, when the fuel cycles are switched from high leakage to low leakage, the void swelling rate appreciably decreases resulting in much smaller void swelling at 60 years. Void swelling predictions are therefore highly sensitive to the assumed fuel cycle management

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