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Evaluation on Irradiation Aging Effects for Reactor Vessel Internals

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1. Introduction

For the long term reactor operation more than 40 years, licensee needs to demonstrate that the effects of degradation of the internal components in pressurized water reactor (PWR) are adequately managed to ensure the continued functionality of the reactor internals. Among the aging degradation mechanisms, the irradiation assisted stress corrosion cracking (IASCC) are especially considered as a significant impact on the assessment and management of the aging degradation of the reactor internals in PWR. The study on the development of the aging management and inspection guideline for the reactor internals in PWR has been currently underway in Korea. This paper presents the procedure and methodology of the evaluation on the effects of the irradiation-induced degradation of the internal components in PWR. Also, the representative results of the baffle-former-bolts in the reactor internal are presented and discussed in this paper.

2. Procedure, Methodology and Results

In this section the process used to develop the aging management strategy for PWR internal are described. This process includes aging management strategy, functionality analysis, and the representative results.

2.1Aging Management Strategy

The aging management flow chart in Fig. 1 illustrates the process conceptualized in MRP-227-A [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 40 fuel cycles. This process permitted further categorization of PWR 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 PWR internals and made the appropriate recommendations for aging management actions specific to each component.

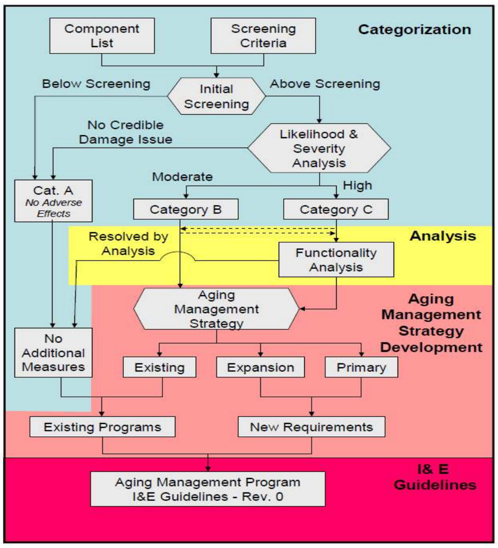

Fig. 1. Strategy for aging management guideline.

2.2 Analysis Model

For the development of aging management and inspection strategies in PWR internal, the functionality analyses were 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. Finite element models are developed for each of the components, which are indicated in Fig 3. The analyses simulate selected components by subjecting them to representative core thermal loading and irradiation loading for a life of 60 years.

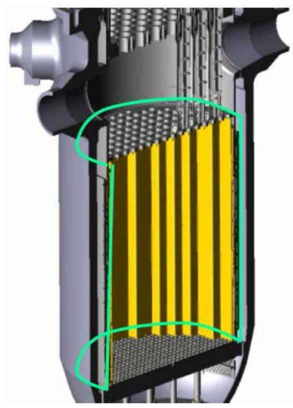

Fig. 2. Baffle-Former-Barrel (BFB) Assembly

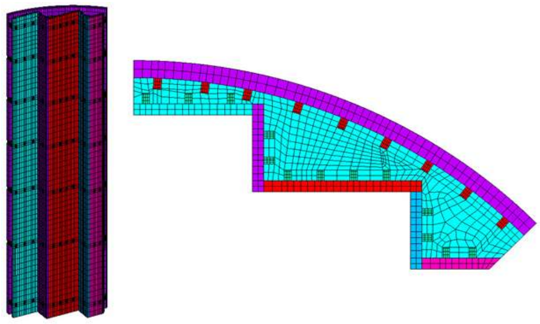

Fig. 3. Finite Element Analysis Model for BFB

2.3 Functionality Analysis

Functionality analyses are 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. The radiation environment for the functionality analysis in PWR internals should be determined using radiation transport methodologies. In the determination of the radiation environment for the reactor vessel internals, calculations are completed for beginning-of-cycle (BOC), middle-of-cycle (MOC), and end-of-cycle (EOC) conditions. The structural analyses examine the aging degradation effects of 60-year life on the baffle-former-barrel (BFB) assembly. The ANSYS model for the baffle-barrel region consists of a one-eighth symmetry solid model using mostly 20-noded hexahedral elements. Contact elements are used in both the structural and thermal models.

2.4 Results and Discussion

Six sets of heat generation data are supplied for the six core loading cases: BOC, MOC, EOC data for both the Out-In and Low-Leakage core loading patterns. Fig. 4 shows the representative heat generation rate distribution for BOC condition of the Out-In loading pattern. The Out-In conditions yield the highest temperatures. The peak temperature occurs in former level at the corner closet to the core. These temperature loadings are applied to the structural model.

The BFB assembly has an L-shaped baffle with a customized corner injection joint between the adjacent baffle plates. The former plates have a spilt near this same location. Contact interfaces are modeled at these locations. The assembly expands outward radially and upward axially. The highest dose rates are at the baffle plates. Coincidentally, the largest distortion occurs between these baffle plates. The baffle plates show the localized indications of IASCC susceptibility due to contact. The former plates show indicated localized indications of IASCC due to thread-to-former coupling. The bolts show greater susceptibility to IASCC at the head shank interface than at the thread-former interface.

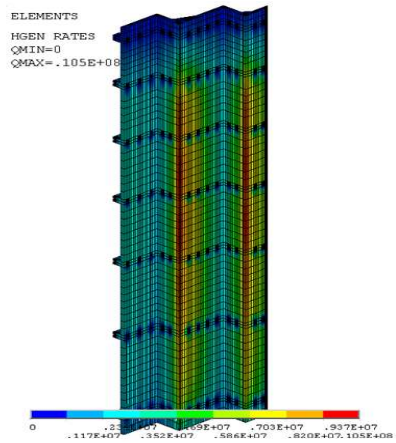

Fig. 4. Heat generation from Out- in loading pattern, BOC [W/m^2]

3. Conclusions

This paper provides the methodologies and results for the functionality assessments on the baffle-former-barrel assembly. For the accurate investigation of the irradiation aging effects on the PWR internals, the three dimensional distributions of neutron exposure and of nuclear heat generation rates within the PWR internal structure are accurately generated. The material behavior model including the essential features of irradiated material should be used for the functionality evaluation of PWR internals.

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