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Evaluation Methodology for Void Swelling Susceptibility of APR1400 Reactor Vessel Internals for U.S. NRC Design Certification

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1. Introduction

KHNP (Korea Hydro & Nuclear Power Co., Ltd.) has been trying to get the Design Certification from U.S. NRC (Nuclear Regulatory Commission) to confirm the enhanced safety designs of the APR1400 (Advanced Power Reactor 1400) to world markets since 2009.

The APR1400 RVI (Reactor Vessel Internals) operates in harsh conditions, such as long term exposure to neutron irradiation, high temperatures, reactor coolant environment, and other operating loads. Therefore, even though the RVI components are mainly made of austenitic stainless steel which is well known to have good mechanical and corrosion-resistive properties, these operating conditions, especially neutron irradiation, cause them to age. The aging is characterized by a chromium depletion along grain boundaries of austenitic stainless steel, a decrease in ductility and fracture toughness of the steel, an increase in yield and ultimate strength of the steel, and a potential volume change due to void formation in the steel. For these reasons, under certain conditions of stress, temperature, and level of irradiation, the void swelling which is one of the challenging degradation mechanisms affecting the integrity of the RVI may appear at specific locations of the RVI, especially due to high neutron fluence and high temperature under localized gamma heating and low velocity of coolant flow [1].

Recently, EPRI (Electric Power Research Institute) started research on aging management for pressurized water reactor internals and published several MRPs (Material Reliability Programs) to provide guidelines on the evaluation of the aging and the aging management methodology and procedures for operating RVI, especially for reactor whose lives had been extended to 60 years [2,3,4,5]. Even though the MRPs have the purpose of providing the methodology of an evaluation or a management for operating RVI, similar evaluation methodologies can be applied to advanced but to-be-built or under construction nuclear power plants, such as the APR1400, in the design stage for the evaluation of neutron irradiation effects on their RVI design.

This paper provides the evaluation methodology for the void swelling susceptibility of the APR1400 RVI based on the EPRI guidelines.

2. Description of the APR1400 RVI

The APR1400 RVI consists of two major structures, referred to as the core support structures and internal structures. The core support structures are those structures or parts of structures which are designed to provide direct support or restraint of the core. Internal structures are all the structures within the reactor pressure vessel other than the core support structures, fuels, control element assemblies, and instrumentations. The core support structures consist of core support barrel, lower support structure, and upper guide structure assembly. For the internal structures, there are core shroud, snubber lugs, alignment keys, inner barrel assembly, heated junction thermocouple tube assembly, and so on. The general arrangement of the APR1400 reactor vessel is shown in Fig. 1. Most of the RVI components are made of stainless steel Type 304 and jointed by the stainless steel Type 308L or Type 347 welds [6].

The main coolant from the four inlet nozzles flows down to the flow skirt through the annulus between the reactor vessel and the core support barrel, and flows upward through the core support region and the reactor core. Finally it exits through two outlet nozzles.

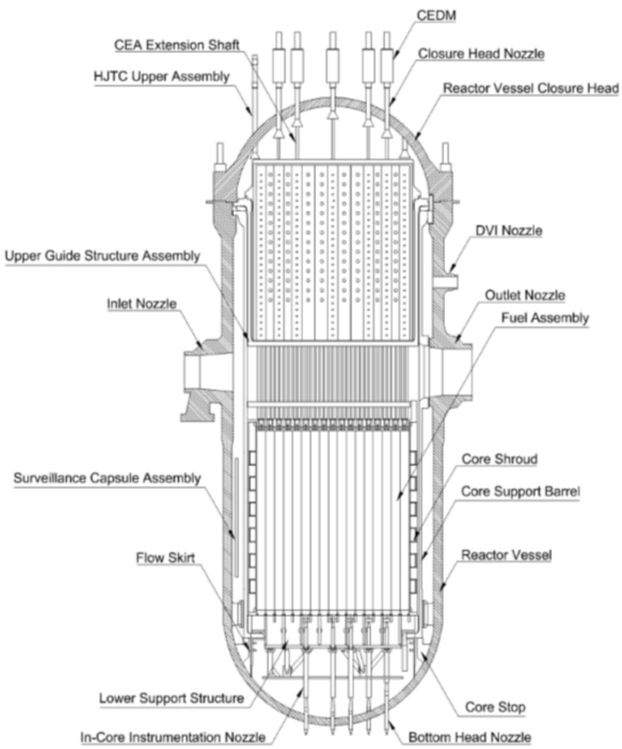

Fig. 1. General arrangement of the APR1400 reactor vessel

3. Methodology for Evaluation on the Void Swelling

To assess the RVI for the irradiation assisted degradation mechanisms such as the void swelling, the radiation transport, temperature distribution and stress analyses are sequentially performed. Fig. 2 provides an activity flow chart generally describing the approach of the evaluation on the void swelling susceptibility of the APR1400 RVI.

At first, an RVI component list of the APR1400 is collected. In addition to the core support structures, the list includes a core shroud assembly, an instrument nozzle assembly, a holddown ring, bolting materials and so on. Then initial screening is performed for the RVI components using their fluence values which were calculated during the RVI design. The screening criterion is the neutron fluence, 5 x 10^{19} n/cm^2 (> 1 MeV). This screening criterion is conservative because it is significantly small compared the void swelling threshold fluences [1]. Based on the initial screening, evaluation scope is determined for a functionality assessment that involves three kinds of computer code analysis: radiation transportation analysis, CFD (Computational Fluid Dynamics) analysis and structural analysis.

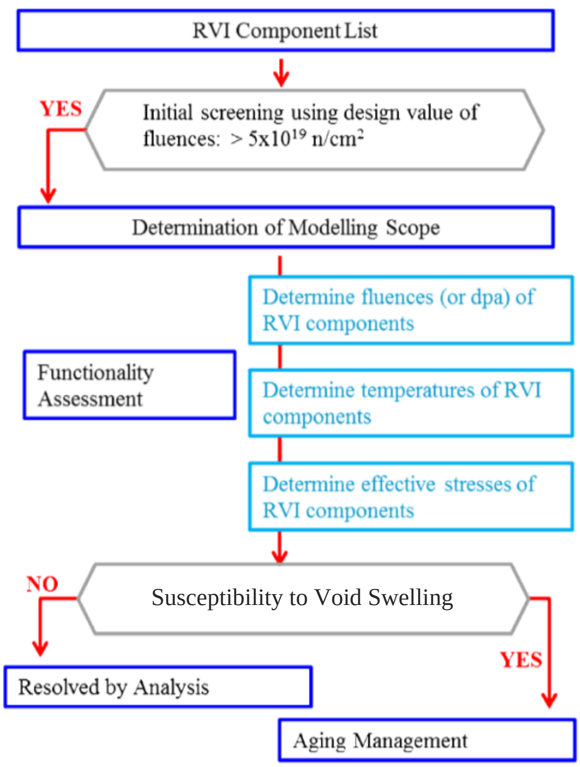

Fig. 2. Flow chart for description of evaluation approach

The model for each analysis is determined to include the RVI components that would be expected to be exposed to neutron fluence higher than 5 x 10^{19} n/cm^2. For the radiation transport analysis, the model includes the RVI components in the range of the fuel alignment plate to the bottom plate of the lower support structure, including the core shroud, the lower part of the core support barrel, snubber lugs, support beans, etc. The scope of modelling is somewhat different for each of the analyses, in accordance with the purpose of each analysis and/or the results of the radiation transport analysis as shown in Fig. 3. The range between the two dotted lines is for the radiation transport analysis, which covers the RVI components that would be expected to be exposed to the neutron fluence above the screening criterion. The range between the two dot-dot-dashed lines is for the CFD simulation, which interfaces with the structural analysis. However, to obtain more accurate temperature and pressure data, the model range is axially extended to include the core support barrel, the support plate of the upper guide structure, the reactor vessel shell and the parts of the hot leg and cold leg. The range between the dot-dashed and the dotted lines is for the structural analysis. A smaller range is selected because the result of radiation transport analysis shows that the neutron fluence of the top plate of the core shroud is below 5 x 10^{19} n/cm^2.

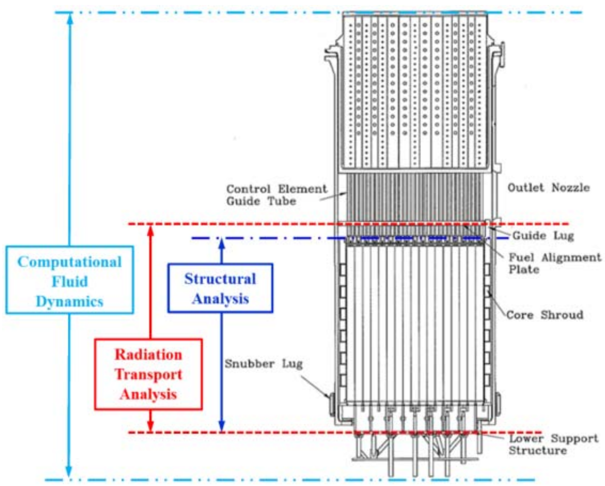

Fig. 3. Model scopes for computer code analysis

The radiation transport analysis provides information on neutron and gamma fluxes in the APR1400 RVI. Neutron and gamma fluxes are calculated both for three different boron concentrations during equilibrium fuel cycles, which represent the beginning, middle, and end of the cycle (BOC, MOC, and EOC) and for conservative fuel cycles considering a conservative radial pin power distribution and axial power shape irrespective of BOC, MOC or EOC. Additionally, a low leakage core is utilized in this analysis. Neutron flux results have been related to the displacements per atom (dpa) occurring as a result of collision of the neutrons in the core shroud, core support barrel and lower support structure regions. Using neutron and gamma flux data, gamma heating is also evaluated in units of W/cm^3 for the RVI. The results of the radiation transport analysis are used as inputs to the CFD analysis and the structural analysis.

The CFD analysis provides temperature and pressure distributions for the RVI. This analysis computes the effects of heat transfer between the metal components and the surrounding fluid and heat generation within the metal components caused by neutron and gamma irradiation. The results of this calculation are metal temperatures and pressures, which are used as boundary conditions by the ANSYS computer code. These conditions correspond to a low leakage core, three states within the equilibrium fuel cycles (BOC, MOC, and EOC) and one state for the conservative fuel cycles. Each case is characterized by a different distribution of heat sources or neutron and gamma fluxes.

The structural analysis of the core shroud, core support barrel, and lower support structure provides effective stress and strain fields during the 60-year plant lifetime. This analysis is a non-linear analysis because the mechanical properties of the material are continuously changing due to neutron irradiation as operation time passes. Using the neutron flux data from the irradiation analysis, and the temperatures and pressures from the CFD analysis, transient analysis of the RVI is performed for 60 years of design life, which includes the initial 8 conservative fuel cycles (12 year operation) and remaining 32 equilibrium fuel cycles (48 year operation). However, it should be noted that if a 93 percent capacity factor of the APR1400 is assumed and considered, structural analysis is only required for 55.8 effective full power years of operation.

After the stress and strain fields, temperatures and the neutron fluence are calculated using the three types of computer code analysis, the susceptibility to the void swelling of the RVI components are determined using the USERMAT module within ANSYS, which is developed by the EPRI [7]. The irradiation growth strain resulted from ANSYS/USERMAT represents a linear directional swelling strain due to irradiation and is equal in all three orthogonal directions. Therefore, its unit is mm/mm or in/in. Meanwhile, the void swelling is commonly measured in a volumetric strain (mm^3/mm^3 or in^3/in^3). If it is larger than 2.5 % for volumetric strain or 0.8265 % for linear strain, the RVI component is assumed to be susceptible to the void swelling [1]. The relationship between the volumetric void swelling strain and the linear irradiation growth strain is as follows:

$$\varepsilon_{volumetric} = (1 + \varepsilon_{linear})^3 - 1 \quad (1)$$

$$\varepsilon_{linear} = (1 + \varepsilon_{volumetric})^{\frac{1}{3}} - 1 \quad (2)$$

4. Conclusions

To assess the effects of operating neutron fluences, temperatures and stresses on the material properties changes and the susceptibility to the void swelling, the evaluation methodology of the APR1400 RVI components for U.S. NRC Design Certification was suggested in this paper. The approach to the evaluation is summarized as follows:

1. RVI component list of the APR1400 is collected.
2. Initial screening to determine the evaluation scope is completed using the design values of fluences.
3. Functionality assessments (radiation transport analysis, CFD analysis, structural analysis) are sequentially performed.
4. Susceptibility to the void swelling is identified through ANSYS/USERMAT module.

KHNP believes that the proposed methodology which is based on the EPRI works for operating reactors is the best way to evaluate the void swelling for new reactors such as the APR1400. The above tasks were submitted to the U.S. NRC and are under review.

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