*Transactions of the Korean Nuclear Society Autumn Meeting*
*Gyeongju, Korea, October 29-30, 2015*

# Neutron Fluence Evaluation of Reactor Internal Structure Using 3D Transport Calculation Code, RAPTOR-M3G

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## 1. Introduction

Aging management for Pressurized Water Reactor (PWR) internals components has been recently issued, since the plants would have life extension as well as long term operation. Age-related degradation mechanisms are including the irradiation-assisted stress corrosion cracking(IASCC), void swelling, stress relaxation, fatigue, and etc. Especially, the threshold value of IASCC, which is related to the fast neutron(E>1.0MeV) exposure for the PWR internal structures, which are composed of austenitic stainless steel, is $2x10^{21}$ $(n/cm^2)$ of fast neutron according to Material Reliability Program(MRP)-175, US EPRI.[1]

Kori unit 2 – the second oldest PWR plant in Korea – is Westinghouse 2 loop commercial plant and has big issue of the integrity of reactor internals; baffle, barrel, and former plate, due to long term operation. Also a lot of Baffle Former Bolts(BFBs) was installed at the former plate ends between baffle and barrel structure. These would undergo severe experiences, which are high temperature and pressure, bypass water flow and neutron exposure and have some radioactive limitation in inspecting their integrity.

The objectives of this paper is to evaluate fast neutron fluence($n/cm^2$, E>1.0MeV) for PWR internals using 3D transport calculation code, RAPTOR-M3G, and to figure out a strategy to manage the effects of aging in PWR internals.

## 2. Neutron transport methodology

In this section, the neutron transport methodology is discussed in detail. The transport calculation for determining neutron exposure at the PWR internals includes geometrical modeling, core source distribution and neutron cross section library.

### *2.1 Transport calculation code*

RAPTOR-M3G(RApid Parallel Transport Of Radiation-Multiple 3 dimensional Geometries)[2], which was developed by Westinghouse and KRIST(Korea Reactor Integrity Surveillance Technology), is parallel, deterministic radiation transport code using three dimensional discrete ordinate method. It can decompose the problem angular domain into up to 8 octants while simultaneously decomposing the spatial domain into a user-specified vector along Z-axis to reduce calculation time and required memory allocation. And It implements theta-weighted[3] and directional theta-weighted[4] solution methodologies in XYZ and RθZ geometries.

### *2.2 Modeling, Source and Cross-section*

3D modeling for RAPTOR-M3G was carried out through BOT3P[5] code, which is a series of codes that can be utilized to generate, view, and plot model geometries including XYZ or RθZ coordinates. An octant(1/8) symmetry reactor model, which is commonly applicable for the radiation shielding calculations, was constructed as shown in Fig. 1.

Core source distributions per nuclear fuel cycle were prepared by SORCERY[6] code. The SORCERY input data such as fuel enrichment, assembly burn-up rate, fuel loading pattern and axial relative power distributions were extracted from the plant- and cycle-specified Nuclear Design Report(NDR). The reactor core was treated as homogeneous mixtures, which are composed of uranium dioxide, inconel-718, zirconium, water, boron, and stainless steel-304, in SORCERY.

BUGLE-96[7] cross-section library which provides 47 neutron and 20 gamma-ray energy group cross-section data set for light-water reactor applications was used for the transport calculations.

### *2.3 Application for Kori unit 2*

In the reactor geometry modeling, if the as-built data were available, these as-built data were used for more precise results. However if these as-built data were not available, the design data were used.

Fig. 1 shows the octant 3D transport calculation model for the cylindrical coordinates and R-θ horizontal cross-section view at core middle plane of Kori unit 2.

Kori unit 2 has 8 former plates and BFBs are connected between baffle and barrel in each former plate. With the interest in aging management of internal structures, cracked BFBs were observed sometimes and the cracks propagated along an intergranular path that is indicative of IASCC.

In these analyses, 3D model of Kori unit 2 was generated with about 4 millions meshes(R, θ, Z = 194, 99, 193) divided into 30 different areas. Also Forward transport calculations were performed from 1 fuel cycle to 25 fuel cycle and convergence criteria of inner and outer iteration were set to 0.001. Also anisotropic scattering was treated with a $P_3$ Legendre expansion and the angular discretization was set with $S_8$ quadrature.

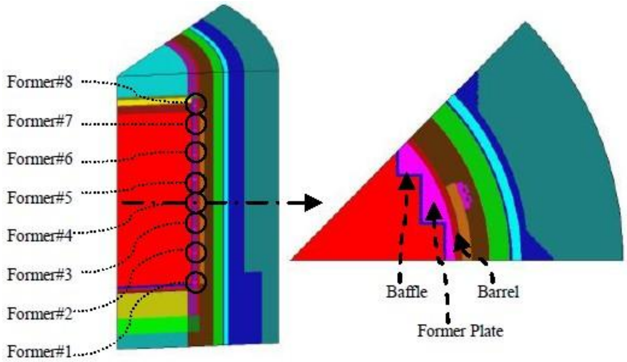

Fig. 1. 3D transport calculation model of Kori unit 2

## 3. Evaluation of internal fluence

In the age-related degradation mechanisms, the threshold value means that the level of susceptibility when an aging effect is first observed. And the screening value means that the level of susceptibility when an aging effect may be significant to functionality or safety. For the IASCC related to the neutron exposure of internal structures, the threshold and screening value are released as $2\text{x}10^{21}$ and $2.7\text{x}10^{22}$(n/cm$^2$) respectively.

Therefore the neutron fluence(n/cm$^2$) generated by RAPTOR-M3G for PWR internal components are to be compared to these threshold or screening values as below.

### *3.1 Neutron fluence of reactor internals.*

There are eight former plates placed in the down comer region of the reactor. They are located between baffle and barrel, and then the BFBs are inserted through the baffle and former plate in order to fix the baffle and former plates together. However, in the neutron transport point of view, these discontinuous structures make the neutron flux increase in these regions. Fig. 2, 3, and 4 show the resultant axial fast neutron fluence profiles of baffle, former plate, and barrel region. In these figures the azimuthal angle(θ) is 16°, where the maximum fluence occurs along the azimuthal angles between 0° and 45°.

As shown in these figures, there are several peaks about 10% higher than the other regions. This is due to the 8 former plates(consist of stainless steel-340) which are less moderation than bypass water.

As shown in Fig. 2, the maximum fluence of baffle region exceeds the screening of $2.2\text{x}10^{22}$ (n/cm$^2$), therefore the aging effect of IASCC may be significant. In conclusion, well-thought-out inspections should be performed in these baffle region comparatively.

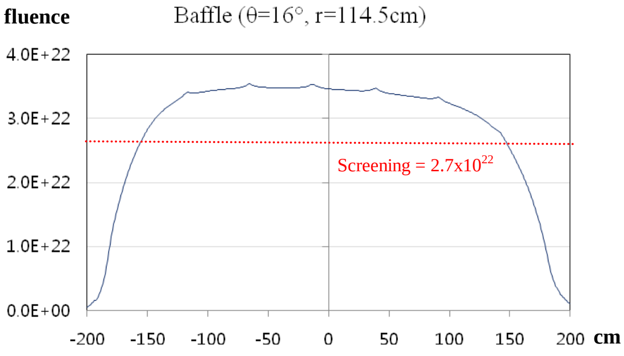

Fig. 2. Axial fast neutron fluence(E>1.0MeV, n/cm$^2$) at baffle

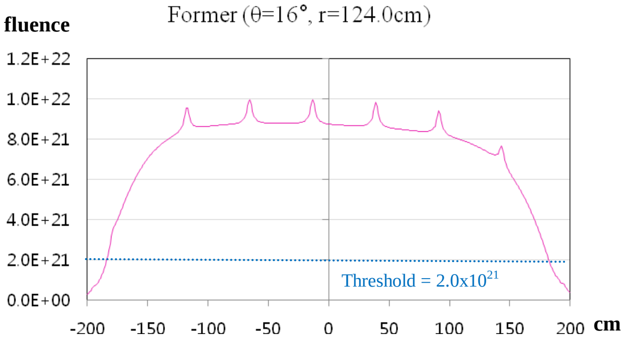

Fig. 3. Axial fast neutron fluence(E>1.0MeV, n/cm$^2$) at former plate

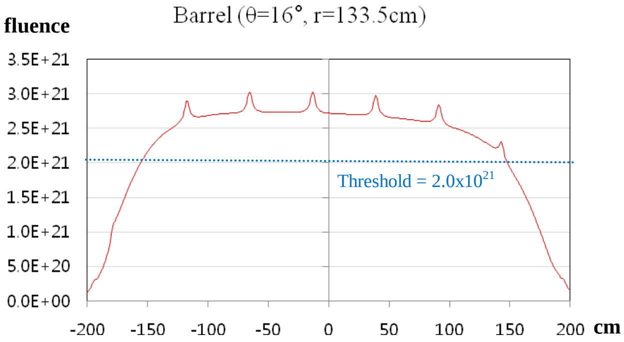

Fig. 4. Axial fast neutron fluence(E>1.0MeV, n/cm$^2$) at barrel

### *3.2 Neutron fluence at baffle region.*

Baffle is the closest structure to reactor core having rectangular geometry. Accordingly, damages by fast neutron related to aging would be the most remarkable in PWR internals. Fig. 5 shows azimuthal neutron fluence profile at the 8 former plate locations.

There are two peaks that fast neutron fluence level is higher than others in Fig. 5. Each of the locations was approximately 16° and 38° near the baffle corner.

Maximum fluence level was evaluated as $3.55\text{x}10^{22}$(n/cm$^2$) at baffle corner(θ=16°) of former plate # 3(Z=-65.437cm) location. Also assuming that operation rate equals to 80% and current flux level keeps going to the future, expected maximum fluence would increase to about $4.57\text{x}10^{22}$(n/cm$^2$) at the end of plant life(32 EFPY). Also it is definitely expected that the regions, which exceeds the screening value, would be widen.

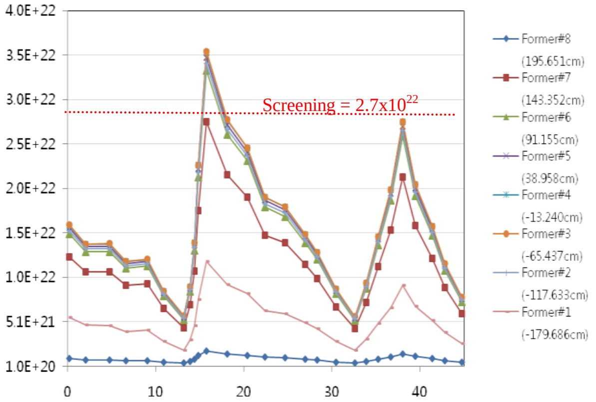

Fig. 5. Azimuthal fast neutron fluence(E>1.0MeV, n/cm$^2$) at baffle

## 3. Conclusions

One of age-related degradation mechanisms, IASCC, which is affected by fast neutron exposure rate, has been currently issued for PWR internals and has $2\text{x}10^{21}$(n/cm$^2$) of the threshold value by MRP-175. Because a lot of BFBs was installed around the internal components, closer inspections are required.

As part of an aging management for Kori unit 2, 3D transport calculation code, RAPTOR-M3G, was applied for determining fast neutron fluence at baffle, barrel and former plates regions. As a result, the fast neutron fluence exceeds the screening or threshold values of IASCC in all of baffle, barrel and former plate region. And the most significant region is the baffle because it is located closest to the core. In addition, some regions including former plate tend to be more damaged because of less moderate ability than water.

In conclusion, IASCC has been progressed for PWR internals of Kori unit 2. Several regions of internal components were damaged by fast neutron exposure and increase in size as time goes by. Therefore more effective inspections and various aging managements are required about higher neutron fluence region.

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