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Calculation of Reactor Pressure Vessel Fluence Using TORT Code

Chul Ho Shin and Jong Kyung Kim
Hanyang University

Abstract

TORT is employed for fast neutron fluence calculation at the reactor pressure vessel. KORI Unit 1 reactor at cycle 1 is modeled for this calculation. Three-dimensional cycle averaged assembly power distributions for KORI Unit 1 at cycle 1 are calculated by using the core physics code, NESTLE 5.0. The root mean square error is within 4.3% compared with NDR (Nuclear Design Report) for all burnup steps. The C/E (Calculated/Experimental) values for the in-vessel dosimeters distribute between 0.98 and 1.36. The most updated cross-section library, BUGLE-96 based on ENDF/B-VI is used for the neutron fluence calculation. The maximum fast neutron fluence calculated on reactor pressure vessel for KORI Unit 1 operated for 411.41 effective full power days is 1.784×10^{18}n/cm^2. The position of the maximum neutron fluence in RPV wall 1/4 T is nearby 60cm below the midplane at zero degree.

1. Introduction

Reactor Pressure Vessel (RPV) integrity must be ensured over the design lifetime, or even longer if plant life extension is considered. That is strongly dependent on the embrittlement of RPV under fast neutron irradiation. In order to assess RPV irradiation damage, the fast neutron fluence must be accurately determined.

RPV fluence calculation is very complex because of deep penetration and anisotropy that characterize the neutron transport. The discrete ordinates method (S_N method) is most frequently selected to solve this type of problem because it provides good accuracy at acceptable computational effort. In the past, DOT (/DORT)[1] was widely used as a standard tool for RPV fluence calculations. In recent years, as computer performance significantly increases, three dimensional calculation can perform with better accuracy at the acceptable cost of computing. In addition, in two-dimensional calculation, the results do not give any informations about;

- Maximum position of neutron fluence at RPV
- Neutron fluence at RPV welding position

Of course, an indirect method for three-dimensional calculation has been developed and is so called synthesis method, by means of which 3D solution is being synthesized from 2D and 1D solutions of neutron transport equation according to the following formulation:

$$\phi(R,\theta,Z) = \frac{\phi(R,\theta)\times\phi(R,Z)}{\phi(R)} \tag{1}$$

This method is currently used to in-vessel capsule benchmark. This method is, however, difficult to apply to three-dimensional activity calculation and streaming calculation through the void (cavity region), because it is assumed that the flux is weakly dependent on z and θ directions. A more accurate description of neutron flux distributions is expected to be obtained by full-dimensional (three-dimensional) analysis. TORT[2] is, thus, chosen for RPV fast neutron calculation in this work.

2. Development of TORT Model

2.1. Geometric Modeling and Material Components

KORI-1 was chosen because the size is small for modeling and it is the first built pressurized water reactor (PWR) with adequate in-vessel dosimetry data. The TORT model for the KORI-1has $80 \times 52 \times 24$ in radial, azimuthal, and axial intervals. Figure 1 illustrates R-θ and R-Z models. Axial model extends to 25.4cm above and below the active core. Reflective boundary conditions are used for the left, front, and back surfaces, and a vacuum boundary condition is prescribed on the top, bottom, and right surfaces. For the TORT model, the fully symmetric S_8 quadrature set, theta-weighted (TW, θ=0.3) differencing scheme, partial current rebalance (PCR) acceleration, ε = 0.0001, and no scattering source fixup are used.

It has been performed that a series of CASMO-3[3] runs to generate a data containing isotropics of fuel for several enrichments including 2.1, 2.83, and 3.2 wt%, and ten burnup steps including 0.0, 0.15, 1.0, 2.0, 4.0, 6.0, 8.0, 10.0, 12.0, and 14.6 GWD/MTU. Based on these data, atomic densities of materials have been calculated throughout the KORI-1 core at cycle 1 and those were averaged over the cycle by burnup weighting[4]. The coolant carries boron in solution at an average concentration of 546 g B/10^6 g of water for cycle 1. The boron concentration was also averaged over the cycle by burnup weighting. The core baffel, barrel, and thermal shield are Type 304 stainless steel. RPV wall is an SA508 steel. The surveillance capsules are located in the downcomer at octal equivalent locations of 13, 23, and 33 degrees.

2.2. Calculation of Source Distribution[4] Using NESTLE 5.0[5]

The multi-group source distribution is determined by combining the power distribution obtained from the core physics calculations with a power-to-source conversion factor, C factor, and a source spectrum χ. The source distribution of group g is then determined by

$$S_{ig} = \chi_g C P_i , \tag{2}$$

where

i = fuel assembly or node index
g = energy group index
χ_g = source spectrum of group g
P_i = assembly or node power
C = power-to-source conversion factor, C factor.

The three-dimensional node-power distribution is determined by using core physics codes, i.e., CASMO-3 and NESTLE 5.0 in combination. The burnup averaged power distribution is obtained by averaging several relative power distributions throughout KORI-1 at cycle 1.

The node-power distributions for KORI-1 cycle 1 using CASMO-3/NESTLE 5.0 at HFP (Hot Full Power), ARO (All Rods Out), and equilibrium Xenon condition were calculated. Relative radial assembly power density distributions at BOC (Beginning of Cycle) and EOC (End of Cycle) are shown in Figure 2. In comparison with NDR (Nuclear Design Report), the range of RMS (Root Mean Square) error are from 1.547% (10.0 GWD/MTU) to 4.265% (BOC). The neutron multiplication factor, k_{eff}, is 1.0042 at BOC. It is thus said that the obtained node power distributions are suitable for source of the TORT model.

Neutrons generated in peripheral assemblies, shown as double lined box in Figure 2, compared against core internal assemblies, much affect the in-vessel capsule reaction rate. Therefore, peripheral assemblies were further divided into sixteen equal nodes and used in source of the model.

3. Calculation of RPV Neutron Fluence

3.1. Comparison of Neutron Fluence Calculations

Neutron fluence at various positions was calculated with optimum numerical options and compared with the results from DORT and MCNP calculations. Figure 3 presents angular distributions of fast neutron flux according to TORT, DORT, and MCNP calculations. The results of TORT and DORT are very resembled each other. The both the results are higher than the results from MCNP calculation at all regions except barrel inside. The reason is that MCNP calculation used heterogeneous model; the flux is reduced by 20% than the flux calculated with homogeneous model.[6] At capsule line, a hump of TORT calculation is lower than that of DORT and this introduces the lower lead factors in TORT calculation. Azimuthal flux distributions of TORT and MCNP at capsule line are almost the same except the capsule positions.

Figure 4 shows fast neutron flux distributions at 1/4 T (T = thickness) for fully Z- θ plane, which can not be obtained from two-dimensional methods. Maximum flux appears at zero degree, about 60cm below midplane; relative difference between the maximum flux and midplane flux is 2.5%. It means that the results using two-dimensional methods may be underestimated.

3.2. RPV Neutron Fluence Calculation

For the purpose of RPV neutron fluence calculation, the effective cross-section has to be obtained in capsule and then capsule fluence has to be calculated. Effective cross-sections and capsule flux were calculated by using BUGLE-96[(7)] dosimetry cross-section library and lead factors for in-vessel dosimeters are presented in Table 1. In all values (i.e., effective cross-section, lead factor, capsule flux, C/E), a close agreement with MCNP results[(8)] shown in Table 2 is observed. It is observed that lead factors of TORT for three dosimeters are smaller than those of MCNP and DORT. Along axial direction, the calculated capsule flux shape is similar with the measured flux shape. This means that axial modeling was well developed.

To validate the methodology used and the modeling developed, the ratios of the calculated to experimental (C/E) capsule fluence are calculated and presented in Table 1. Three threshold reactions, including ^{54}Fe(n,p), ^{58}Ni(n,p), and ^{63}Cu(n, α), are considered. The range of C/E values is from 0.98 to 1.36. ^{54}Fe(n,p) and ^{63}Cu(n, α) dosimeters give a good results in this model. The C/E values at top in ^{54}Fe(n,p) dosimeter, middle in ^{58}Ni(n,p) dosimeter, and topmiddle in ^{63}Cu(n, α) are somewhat different from the others.

To calculate flux on RPV wall and within RPV, lead factors and attenuation factors are used, respectively. In this work, two attenuation factors was calculated; 69.1% at 1/4 T, 37.2% at 1/2 T. The maximum values of neutron flux on RPV inner wall and RPV wall at 1/4 T and 1/2 T were calculated, respectively. Table 1 presents the values. In the ^{54}Fe(n,p) dosimeter, the position of the maximum neutron fluence at RPV inner wall is nearby 70cm below the midplane. This result is similar with calculated position at 1/4 T, 60cm below the midplane. The maximum neutron flux at RPV inner wall is calculated as 6.03×10^{10} n/cm^{2}s. Since KORI-1 hd been operated for 411.41 effective full power days, the calculated vessel fluence is 1.784×10^{18} n/cm^2.

4. Conclusions

This study has evaluated the reactor pressure vessel fluence by using TORT code. Three-dimensional cycle averaged source distribution was calculated. The range of C/E values was from 0.98 to 1.36. The maximum fast neutron flux appeared at about 60cm below midplane at zero degree, 1/4 T in RPV wall. The difference between the max flux and midplane flux was 2.5%. Calculated flux shape using TORT code was axially similar with measured capsule flux shape. The effective cross-section, capsule flux, and lead factors gave similar values with the results of MCNP calculations. The azimuthal neutron fluxes are similar with results of DORT and MCNP. The C/E values have also a close agreement within 36%. The calculated vessel fluence for KORI-1 operated for 411.41 effective full power days is given as 1.784×10^{18} n/cm^2.

References

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Table 1. Summary of Neutron Dosimeter Results and Calculated Fast Neutron Flux at 13 Degree

		A_{SAT} (dps/mg)	σ_{eff} (barns)	Capsule Flux (n/cm²s)	Lead Factor	C/E	Fast Neutron Flux (n/cm²s) RPV	1/4T	1/2T
$^{54}Fe(n,p)^{54}Mn$ 158.3 cm	T[a] 279.37[b]	4350	7.304-02[c]	9.522+10	2.17+00	1.34	4.39+10	3.03+10	1.63+10
	TM 243.87	5080	7.324-02	1.109+11	2.21+00	1.18	5.02+10	3.47+10	1.87+10
	M 208.27	5090	7.328-02	1.111+11	2.24+00	1.19	4.96+10	3.43+10	1.84+10
	BM 172.72	5310	7.328-02	1.159+11	2.28+00	1.16	5.08+10	3.52+10	1.89+10
	B 137.16	5470	7.383-02	1.185+11	2.27+00	1.13	5.21+10	3.60+10	1.94+10
$^{58}Ni(n,p)^{58}Co$ 159.3 cm	M 208.27	55600.0	1.041-01	7.627+10	1.76+00	1.36	4.34+10	3.00+10	1.61+10
$^{63}Cu(n,\alpha)^{60}Co$ 159.3 cm	TM 172.72	480	7.012-04	1.044+11	1.73+00	0.98	6.03+10	4.17+10	2.24+10
	BM 172.72	424	7.021-04	9.213+10	1.79+00	1.14	5.15+10	3.56+10	1.91+10

[a]See Figure 1

[b]Read as 279.37cm from bottom

[c]Read as 7.304×10^{-02}

Table 2. Summary of Neutron Dosimeter Results by MCNP Calculation at 13 Degree

	$^{54}Fe(n,p)^{54}Mn$	$^{58}Ni(n,p)^{58}Co$	$^{63}Cu(n,\alpha)^{60}Co$
C/E	0.97	1.35	1.10
σ_{eff} (barns)	7.120-2[a]	9.769-2	7.001-4
Capsule Flux (n/cm²s)	1.111+11	8.084+10	9.073+10
Lead Factor		2.30	

[a]Read as 7.120×10^{-2}

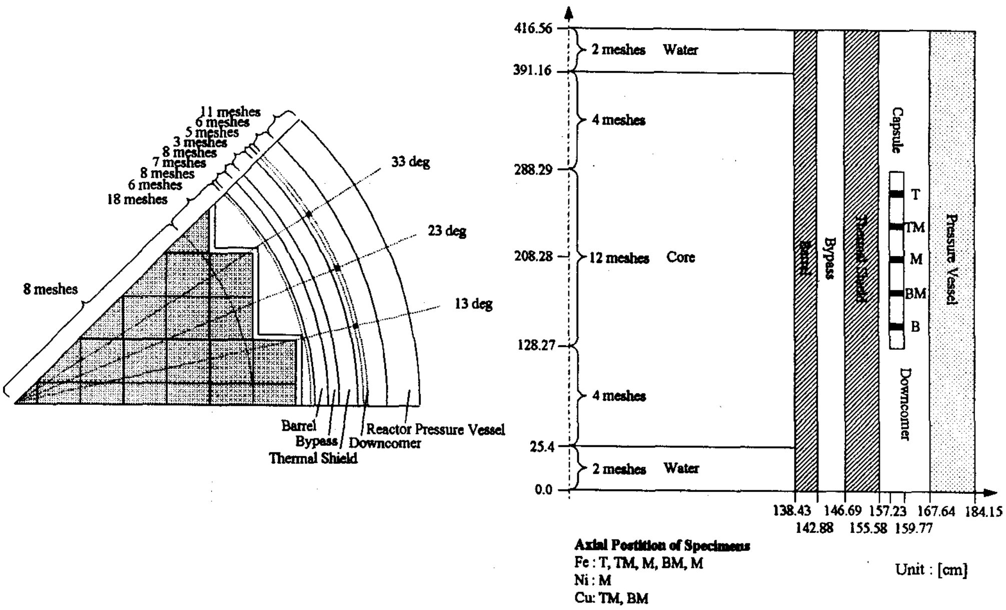

Figure 1. KORI-1 R-θ and R-Z Models

	NDR NESTLE %ERROR			0.722 0.729 0.790		
			1.120 1.158 3.616	1.043 1.042 -0.010	0.608 0.603 -1.053	
		1.146 1.175 2.644	1.179 1.155 -1.934	1.062 1.089 2.627	0.778 0.752 -3.432	
	1.094 1.170 6.819	1.080 1.045 -3.324	1.144 1.164 1.827	1.151 1.113 -3.223	1.132 1.124 -0.618	0.662 0.648 -2.236
1.081 1.204 11.138	0.986 1.045 5.771	1.105 1.151 4.045	1.096 1.043 -4.891	1.145 1.156 1.039	1.064 1.022 -3.910	0.821 0.810 -1.486

Root Mean Square Error = 4.265 % (BOC)

%ERROR = (NDR-NESTLE)×100/NDR

				0.704 0.695 -1.349		
			1.048 1.047 -0.057	1.018 1.0136 -0.432	0.629 0.608 -3.418	
		1.116 1.135 1.729	1.197 1.206 0.735	1.022 1.014 -0.763	0.821 0.802 -2.351	
	1.097 1.126 2.662	1.198 1.224 2.137	1.116 1.131 1.344	1.174 1.174 0.017	1.043 1.020 -2.205	0.633 0.607 -4.076
1.072 1.104 2.948	1.064 1.107 4.070	1.111 1.137 2.367	1.205 1.222 1.403	1.097 1.100 0.246	1.073 1.057 -1.501	0.767 0.743 -3.090

Root Mean Square Error = 2.077 % (EOC)

Figure 2. Relative Assembly Power Density Distributions at BOC and EOC

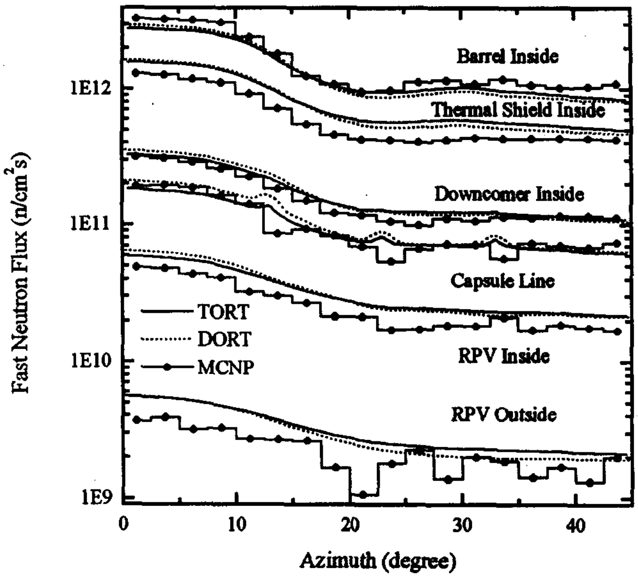

Figure 3. Angular Distributions of Fast Neutron Flux by TORT, DORT, and MCNP Calculations

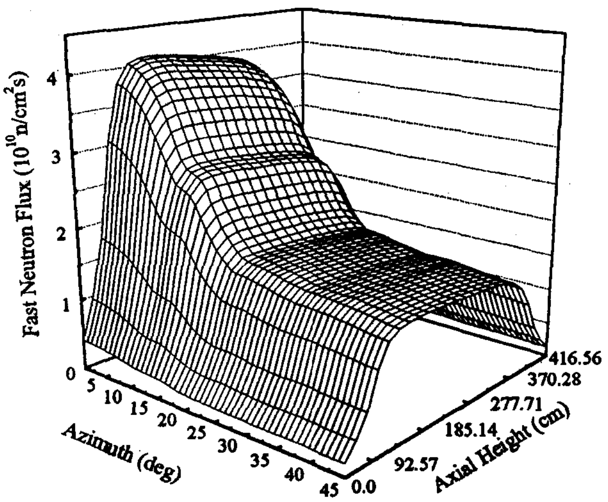

Figure 4. Fast Neutron Flux Distribution of Z- θ Plane at 1/4 T RPV