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Evaluation of Korean Next Generation Reactor Vessel Neutron Fluence by Full-Scope Monte Carlo Calculations

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ABSTRACT

The fast neutron fluence at the reactor pressure vessel(RPV) of KNGR designed for 60 years lifetime was calculated by full-scope Monte Carlo simulation for reactor vessel integrity assessment. KNGR core geometry was modeled on a three-dimensional representation of the one-sixteenth of the reactor in-vessel component. Each fuel assemblies were modeled explicitly excluding spacer grids, and each fuel pins were axially divided into 5 segments. The maximum flux of $4.3\times10^{10}\,neutrons/cm^2\cdot\sec$ at the RPV was obtained by tallying neutrons crossing the beltline of inner surface of the RPV. The end of life fast neutron fluence at the RPV is satisfactory to achieve 60 years of design lifetime of KNGR.

1. INTRODUCTION

The lifetime of a nuclear power plant(NPP) is strongly related to the maintenance of the structural integrity of the reactor pressure vessel(RPV). The aging mechanism of primary concern for RPVs is irradiation-induced embrittlement of the beltline material. Irradiation embrittlement decreases both the cleavage fracture toughness and the ductile tearing toughness of RPV materials [1]. Most of all, the neutrons with energies greater than approximately 1 MeV are the primary cause of RPV embrittlement. Therefore, if the fast neutron fluence at the RPV is accurately predicted, the design lifetime of the reactor being designed can be clearly determined.

KNGR (Korean Next Generation Reactor) is a nuclear power plant designed for 60 years lifetime, and it is being designed in detail. In this study, the RPV fast neutron fluence of KNGR was estimated, and the vessel integrity was assessed by using Monte Carlo simulation.

2. MONTE CARLO METHOD

The Discrete Ordinates method is generally used for the calculation of RPV fluence [2]. This method is used to determine a synthesized three-dimensional flux distribution based on one-dimensional and two-dimensional transport calculations. This S_N transport calculation contains uncertainties associated with the multigroup cross-section libraries, multi-dimensionality, geometric approximations, and angular discretization [3,4]. Recently, it is recognized that alternative method, Monte Carlo, is available for accurate results.

Monte Carlo analysis of the neutronic behavior of a light water reactor (LWR) in combination with continuous cross-section data is an attractive tool to provide a detailed description of a static LWR core. Very few limitations exist in the area of geometric modeling of a given problem. Furthermore, details of the original nuclear data evaluation can be retained in the cross-section library and self-shielding in the resolved resonance range is explicitly taken into account. In recent days, the calculations of RPV fluence and criticality benchmark were performed by using the Monte Carlo Method, and the accuracy and consistency of this method was well established by the previous workers [4-6]. Therefore, for an accurate estimate of the neutron fluence at the RPV a full-scope Monte Carlo simulation is well suited for this task and was used by this work.

3. MCNP MODELING OF KNGR

KNGR is a LWR to be planed to produce 3983 MW thermal core power and based on the System 80+ of ABB-CE. The reactor has been designed by Korea Power Engineering Company (KOPEC) and Korea Nuclear Fuel Company (KNFC).

Figure 1 shows the cross-sectional view of one-eighth of the KNGR in-vessel components with reflective angular boundaries at 0 and 45 degrees, as modeled by MCNP4B [7]. This model explicitly represents the rectangular and cylindrical domains in three dimensions, and baffle, barrel, and pressure vessel were clearly described. The state of beginning of cycle in cycle 1 was simulated at hot full power, equilibrium xenon, and all rods out conditions. Figure 2 shows the axial view of one-second of the reactor with reflective boundary at core center. To estimate the axial power distributions in fuel rod, each assembly was divided into 5 segments at the axial direction. Top reflector was assumed by three layers of baffle, coolant, and barrel as in the side reflector region. One-sixteenth core was modeled to reduce the size of the input model, allowing computation time faster.

To model the core configurations, it is necessary to have uranium, plutonium, and fission product concentrations for representation of the depleted core. For this information, associated depletion calculations were performed by CASMO-3 code [8]. The isotopic inventories in each assembly at the state were generated and used in MCNP material input. For the purpose of considering the Doppler broadening effect, new cross-section library at the core temperature was generated by NJOY [9] and ENDF/B-VI. In this study, the methods chosen to accelerate tally convergence were energy cutoff, geometry splitting with Russian roulette, and weight cutoff. In general, only neutrons with energies above 1 MeV are considered to contribute metal damage [10], and thus this energy was chosen for cutoff-energy in the RPV fluence calculations.

4. RESULTS OF THE RPV FLUENCE

A couple of calculations were carried out to estimate fast neutron fluence at the RPV. One is a criticality calculation, and the other fluence calculation. The criticality calculation employs the KCODE option [7] to obtain k_{eff} eigenvalue of the system and relative power distributions. In the model of the KNGR cycle 1, through confirming that k_{eff} was converged to unity and relative power distribution was consistent with that of initial core design report [11], the validity of the model was examined.

In five segments of each fuel rod, fission reaction densities were calculated. These results were used to be neutron source information for RPV fluence calculations. After 150 cycle calculations (KCODE), the system converged to a k_{eff} value of 1.00338 ± 0.00084. While the criticality calculation was performed, the relative pin power calculations in all rods of five segments were carried out. Thereafter, the averaged relative assembly power was calculated from integrated pin powers. Figure 3 shows the relative assembly power distribution together with the values of the initial core design report of the KNGR cycle 1.

Fast neutron fluence calculations were performed by the fixed source (SDEF) mode [7] in a full-scope MCNP simulation. The relative pin power distribution in the entire core was used as the neutron source probability for the fluence calculation model. Once the position of the starting neutron was determined, its energy was selected from the Watt fission spectrum of the fissile nuclides.

Tally was set around high-flux beltline region, and that was subdivided into 30 sub-cells by 1.5 degrees along the azimuth. The fast neutron angular flux (E > 1 MeV) distribution at the beltline of the core was shown in Figure 4. After the sufficient neutron transport simulations, forty million histories, were done, the relative error of all tallies was made within 5%. It is noted that the beltline flux shows a peak at 17 degree, and the value is given as $4.3 \times 10^{10}\, neutrons/cm^2 \cdot \sec$.

5. CONCLUSIONS

Full-scope Monte Carlo simulation was applied to the RPV fluence calculation of KNGR to investigate quantitatively the lifetime of the reactor. The fuel loading of cycle 1 was explicitly described by pin-by-pin by MCNP4B code, and the validity of modeling was confirmed by criticality calculation. By using neutron source distribution from the criticality calculation, the RPV fluence was calculated around the high-flux beltline region.

The maximum fast neutron flux at the RPV was given as $4.3 \times 10^{10}\, neutrons/cm^2 \cdot \sec$. The end of life fast neutron fluence at the RPV is satisfactory to achieve 60 years of design lifetime of KNGR.

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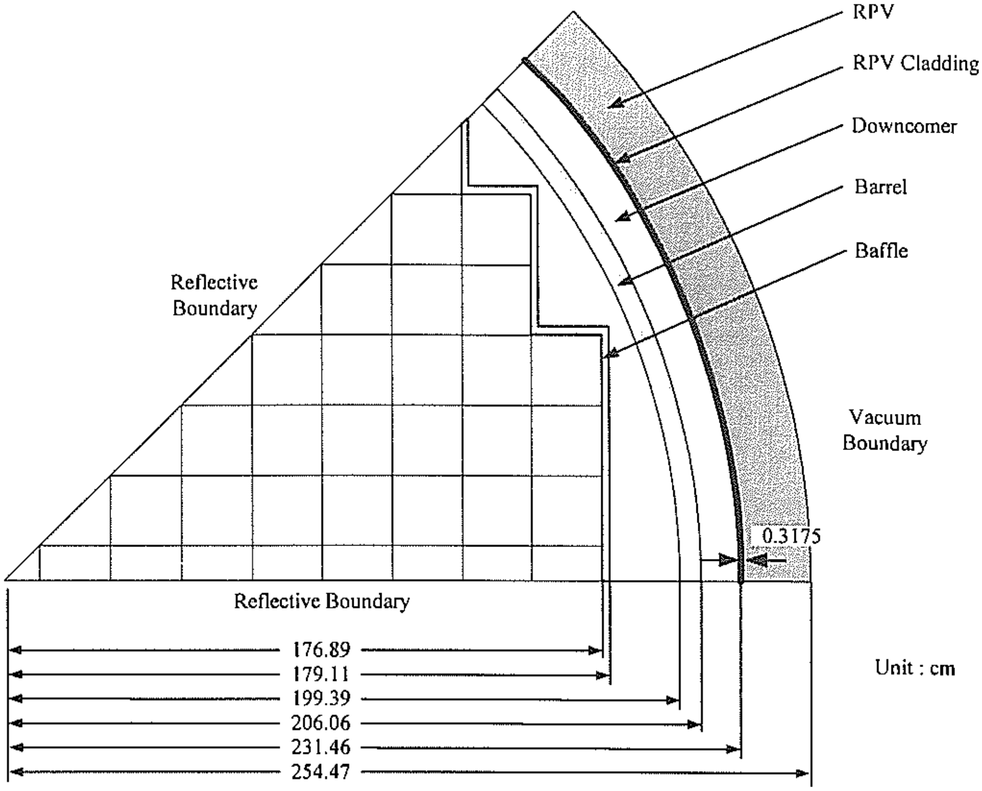

Figure 1. Cross-Sectional View of the KNGR Core

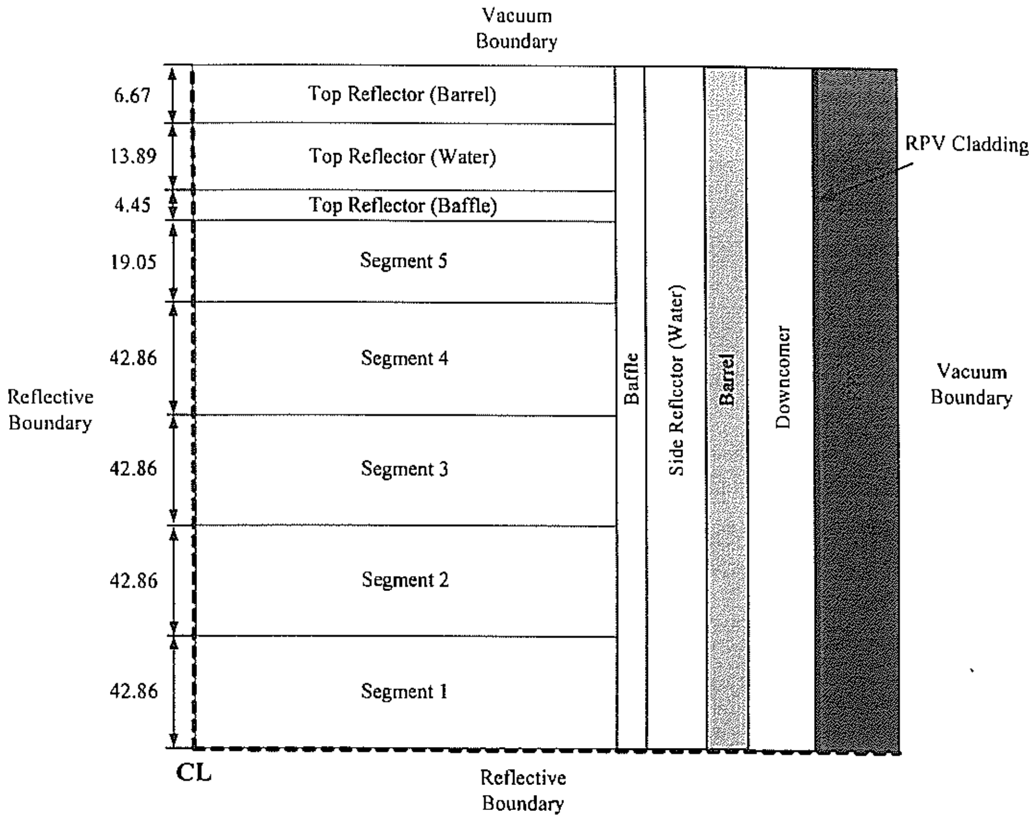

Figure 2. Axial View of the KNGR Cycle 1 Core

Design MCNP Diff.(%)*

						0.695 0.614 -11.6		
					0.863 0.820 -5.0	0.956 0.897 -6.1	0.656 0.639 -2.7	
				0.906 0.889 -1.9	1.154 1.137 -1.5	1.116 1.108 -0.7	1.093 1.074 -1.7	
			0.876 0.878 0.3	1.158 1.166 0.7	0.920 0.924 0.4	1.175 1.210 3.0	1.050 1.078 2.7	0.676 0.722 6.8
		0.888 0.881 -0.8	1.141 1.153 1.1	0.877 0.882 0.5	1.166 1.191 2.1	0.924 0.967 4.7	1.152 1.183 2.7	1.006 1.038 3.3
	0.903 0.894 -1.0	1.138 1.125 -1.2	0.878 0.878 0.1	1.137 1.152 1.4	0.880 0.912 3.6	1.173 1.214 3.5	0.930 0.950 2.2	0.993 1.058 6.6
0.899 0.894 -0.6	1.118 1.107 -1.0	0.895 0.886 -1.1	1.140 1.202 5.4	0.869 0.918 5.7	1.143 1.184 3.6	0.893 0.932 4.4	1.179 1.186 0.6	0.965 1.057 9.5

Root Mean Square Error = 3.68 %

* Difference = $\dfrac{\text{MCNP} - \text{Design}}{\text{MCNP}} \times 100\ (\%)$

Figure 3. Relative Power Distribution of the KNGR Model

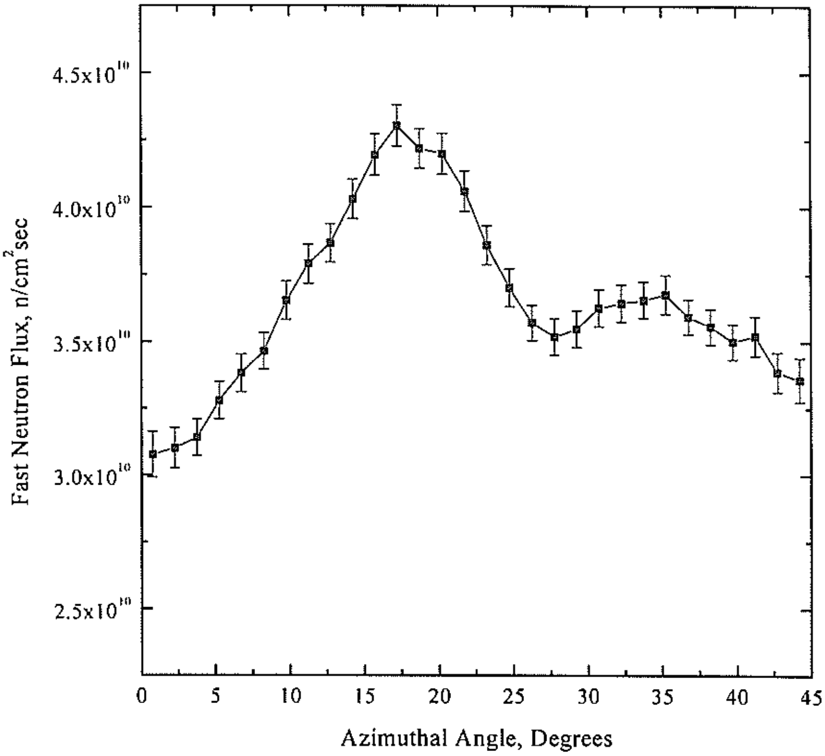

Figure 4. Angular Flux Distribution at the Vessel Beltline