An Evaluation of Korean Next Generation Reactor Pressure Vessel Neutron Fluence by Monte Carlo Simulations

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ABSTRACT

Ensuring the integrity of rector pressure vessel (RPV) material is one an important task for economical reasons as well as for safety confirmation during nuclear power plant lifetime. Fast neutron bombardments on the RPV degrade the structural integrity of the vessel. The assessment of RPV integrity requires accurate prediction of neutron fluence over the reactor lifetime. The fast neutron fluence at the RPV of Korean Next Generation Reactor (KNGR) has been calculated by a full-scope explicit modeling of Monte Carlo simulation for an accurate assessment of the structural integrity. The KNGR cycle 8 core geometry has been modeled on three-dimensional representations of the one-sixteenth of the reactor in-vessel components by MCNP4B. All the fuel rods of each fuel assembly have been modeled explicitly. In the Full-Scope Monte Carlo simulation, the maximum RPV flux of $2.738 \times 10^{11} neutrons/cm^2 \cdot sec$ has been obtained by tallying neutrons crossing the beltline of inner wall surface of the RPV. In the ROCS (core analysis code developed by ABB-CE)+MCNP4B calculation, the maximum flux of $2.769 \times 10^{11} neutrons/cm^2 \cdot sec$ at the inner vessel beltline has been obtained.

INTRODUCTION

The lifetime of nuclear power plants is strongly related to the preservation maintenance of the healthy structural integrity of reactor pressure vessel (RPV). The RPV cannot be replaced during the plant lifetime, and therefore its integrity must be ensured over the design lifetime. The primary concern of aging mechanism for RPVs is irradiation-induced embrittlement of the vessel material. Irradiation embrittlement causes to decrease both the cleavage fracture toughness and ductile tearing toughness of RPV materials [1]. Fast neutrons are the primary cause of RPV embrittlement. The RPV materials in pressurized light water reactors are mainly embrittled by the bombardment of neutron with energies greater than approximately 1 MeV during reactor operation [2]. Korean Next Generation Reactor (KNGR) is under design, with 60 year lifetime goal. Therefore, if the fast neutron fluence at the RPV is accurately predicted, the design lifetime of the reactor can be clearly determined

The Discrete Ordinates S_N method has been generally used for the calculation of RPV fluence [3]. This method is used to determine a synthesized three-dimensional flux distribution based on one-dimensional and two-dimensional transport calculations. S_N transport calculation also contains uncertainties associated with the multigroup cross-section libraries, multi-dimensionality, geometric approximations, and angular discretization [4, 5]. This study requires an explicit description of the modification of peripheral fuel assembly neutron source. It has been recognized that an alternative method, the Monte Carlo, is available recently for accurate results.

The benefits associated in employing Monte Carlo method in reactor core calculation are almost geometry-free modeling, and the use of the continuous energy nuclear cross-section data and angular distribution, which are the sources for accurate prediction of neutron behaviors. The utilization of this method makes possible to model almost real core geometry and to use explicitly the original pointwise nuclear data. In recent days, Monte Carlo method has been utilized in the reactor core and RPV fluence calculations, and the accuracy and consistency of this method are well established [5, 6, 7].

A full-scope explicit modeling of KNGR core by Monte Carlo method is carried out for RPV fluence calculation. In this study, MCNP4B, Monte Carlo particle transport code [8] is employed to model the KNGR cycle 8 core, equilibrium core and to calculate its RPV fast neutron fluence. In order to generate the burnup-dependent material compositions data of each fuel assembly, depletion calculations are carried out by using CASMO-3, lattice physics code [9].

MCNP MODELING OF KNGR

Geometric Model

KNGR is a pressurized light water reactor (LWR) based on the System 80+ design of ABB-CE to be planned to produce 3983 MW thermal core power [10]. The reactor has been designed by Korea Power Engineering Company (KOPEC) and KEPCO (Korea Electric Power Corporation) Nuclear Fuel Company (KNFC). The KNGR core characteristics are summarized in Table 1.

Table 1. KNGR Core Characteristics

Average Moderator Temperature	310.6 °C	Effective Core Diameter	365.8 cm
Average Fuel Temperature	701.4 °C	Assembly Pitch	20.88 cm
Active Core Height	381 cm	Fuel Rod Pitch	1.285 cm
No. of Fuel Assemblies	241	Fuel Pellet Diameter	0.826 cm
Lattice in Assembly	16×16	Inner Diameter of Cladding	0.843 cm
Maximum Enrichment	5.0 w/o	Outer Diameter of Cladding	0.970 cm
Lead Rod Burnup	60000 MWD_T	Pellet Theoretical Density	10.96 g/cm^3
		Pellet Density (% Theoretical)	95.25 %

The reactor core consists of 241 fuel assemblies. Each fuel assembly consists of a 16×16 array with 236 fuel rods and 5 water holes. The cross-sectional view of one-eighth core model of the KNGR in-vessel components with reflective angular boundaries at 0 and 45 degrees is shown in Figure 1. This model for MCNP4B explicitly represents the rectangular and cylindrical domains in three dimensions. The core baffle, barrel, vessel cladding, and pressure vessel are also clearly modeled. The state of beginning of life (BOL) in cycle 8 is simulated under the conditions of hot full power (HFP), all rods out (ARO), equilibrium xenon and 1180 ppm of boron concentration [10].

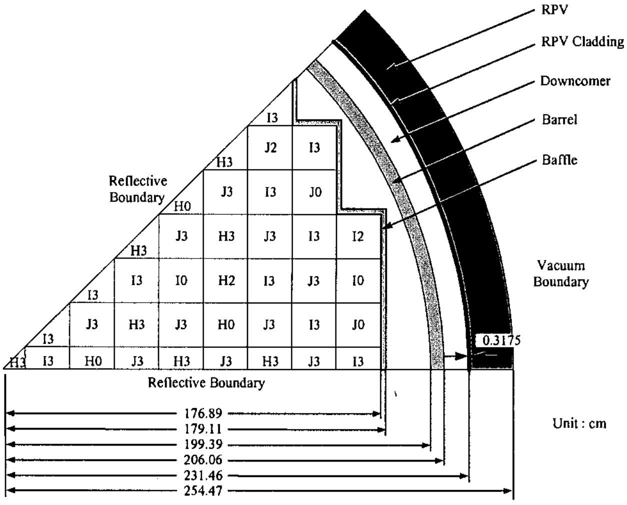

Figure 1. Cross-Sectional View of the KNGR Cycle 8 Core

During the cycle 8, nine types of fuel assemblies, H0, H2, H3, I0, I2, I3, J0, J2 and J3, are burned in the core. All type assemblies contain both 4.7 and 4.2 w/o. In KNGR core, many of the fuel assemblies contain zones of lower enrichment fuel rods to reduce the power peaking within their assembly. The enrichment pattern of fuel assemblies for KNGR cycle 8 core are given in Figure 2. KNGR core uses gadolinia fuel rods for the partial control of excess reactivity.

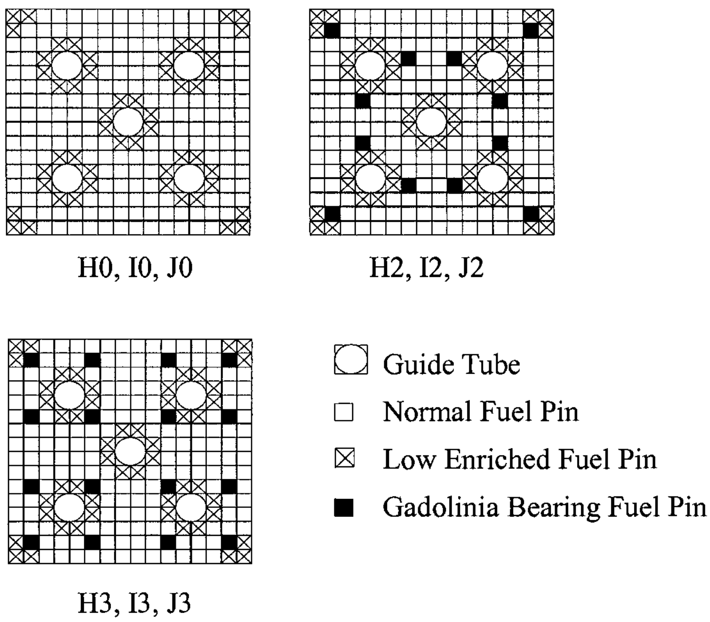

Figure 2. Assembly Configuration for KNGR Cycle 8 Core Design

The axial view of top-half calculation model of the reactor with a reflective boundary at the core center is shown in

Figure 3. In order to estimate the axial power distributions in the fuel rods, each assembly is divided into five segments in axial direction. Top reflector is assumed to consist three layers of baffle, coolant, and barrel, as in the case of the side reflector region. One-sixteenth core model with reflective boundary reduces the size of the problem, and therefore allows reduction of computing time with negligible change of calculated results.

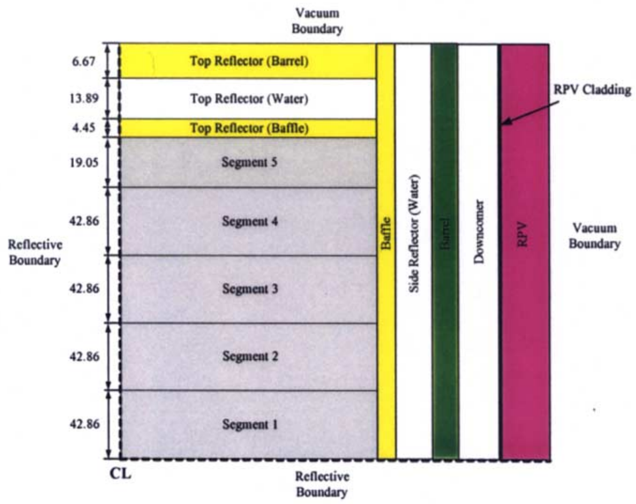

Figure 3. Axial View of KNGR Core

Cross-Section and Depletion Model

For taking into consideration of burnup of the KNGR cycle 8 core, it is necessary to know the change of the concentrations of uranium, plutonium and fission products destroyed and created following burnup. For burnup information, a depletion calculation is performed by CASMO-3 code. The isotopic inventories in each assembly are generated and converted into MCNP material input for each burnup step.

For the consideration of Doppler broadening effect at the real core temperature, a new cross-section library, named as KNGRXS, has been generated by using NJOY97 [11] and ENDF/B-VI cross-sections data [12] scinse the core internal temperature is higher than the library temperature. MCNP4B model for fission products have been evaluated by using Lawrence Livermore National Laboratory model [8] and termed as "average fission product." The consideration of all the fission products would require considerable computing time and effort, and thus, a lumped fission product model has been used.

RESULTS AND DISCUSSION

Neutron Source Calculation

In order to estimate fast neutron fluence at the RPV, two step calculation have carried out in this study. One is a neutron source term calculation for RPV fluence, and the other is a fluence calculation. In the Full-Scope Monte Carlo simulation, the criticality calculation has been employed the KCODE option of MCNP4B [8] to obtain k_{eff} eigenvalue of the system and relative power distributions for the neutron source information. In the KNGR cycle 8 core, by confirming k_{eff} converged to unity and the relative power distribution consistent with that of the nuclear design report [10], the validation of the model is carried out. The relative power distribution has been obtained by F4 tally (track length estimate of cell flux) of MCNP4B [8].

In five segments of each fuel rod, the power by fission reactions has been calculated. These results are used for neutron source information in the second step RPV fluence calculation. After 150 cycles (in KCODE run), the system has been converged to a k_{eff} value of 0.99955±0.00084. The number of neutrons per cycle has been chosen as 5000, and 10 cycles have been skipped before k_{eff} data accumulation begin. The relative pin power calculations in all rods of the five segments have been carried out. The averaged relative assembly power has been calculated by integrating pin powers. The calculated relative assembly power distribution of the KNGR cycle 8 is shown in Figure 4 together with the proposed design values, which have been calculated using ROCS code (core analysis code).

MCNP Design Error(%)*

						0.423 0.401 5.534		
					0.699 0.748 -6.615	0.885 0.860 2.899	0.436 0.433 0.592	
				1.031 1.078 -4.362	1.146 1.176 -2.572	0.945 1.007 -6.095	0.961 0.960 0.091	
			0.951 0.896 6.067	1.296 1.271 1.997	1.011 1.049 -3.558	1.299 1.222 6.297	0.831 0.888 -6.444	0.455 0.428 6.316
		1.106 1.122 -1.363	1.101 1.085 1.488	1.244 1.267 -1.827	1.054 1.103 -4.429	1.042 1.125 -7.352	1.183 1.126 5.098	0.736 0.690 6.654
	1.232 1.157 6.464	1.308 1.259 3.873	0.915 0.885 3.398	1.238 1.192 3.861	0.943 0.946 -0.249	1.261 1.226 2.916	1.023 1.054 -3.004	0.910 0.921 -1.216
0.924 0.871 6.101	1.171 1.109 5.537	1.178 1.096 7.524	1.248 1.159 7.756	0.908 0.898 1.095	1.104 1.162 -4.989	0.986 0.972 1.438	1.173 1.160 1.097	0.642 0.686 -6.446

Root Mean Square Error (%) = 4.467

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

Figure 4. Relative Power Distributions of the KNGR Cycle 8 Core

It has been observed that the calculated assembly powers are reasonably consistent with the design values. The effect on the peripheral assembly powers is generally more critical than the others of inner core in the RPV fluence calculation [3, 13].

In the ROCS+MCNP4B calculation methodology, assembly power distribution has been calculated using ROCS code and the fast neutron fluence calculations have been performed using the results of ROCS code.

RPV Fluence Calculation

The fracture toughness of pressure vessel material is directly related to the reference temperature at the end of the license period (RT_{PTS}) which is based on the best estimate of fast neutron fluence at RPV materials. Only neutrons with energies above 1 MeV are considered to contribute to vessel damage, and this energy is thus chosen as the cutoff-energy for the RPV fluence calculation.

The relative pin power distributions of all the rods in the five segments are converted into the neutron source probabilities for the RPV fluence calculation. The probability that a source neutron will be born at a particular position is proportional to the specific power of that position. The probability distribution function of the neutron source is established in such a way that the probability of a neutron to be born in a segment of a rod is proportional to the normalized power fraction of that segment. Once the position of a starting neutron is determined, its energy is selected from the Watt fission spectrum of the fissile nuclides. Because of the small fraction, say less than 2% of Pu-241 at the BOL state, the effect of Pu-241 spectrum has been neglected. By using the fixed source problem (SDEF) of MCNP4B [8], the RPV fluence calculation has been carried out.

MCNP4B fluence results must be multiplied by a factor representing the absolute value of neutron source because the result is different from normal fluence unit as obsered in the deterministic methods. This tally multiplication factor is just the total absolute neutron source value of all fuel pins, namely, the total number of fission neutrons that are included in the system.

Figure 5 shows tally regions divided into 5 segments axially and these are subdivided into 30 sub-cells by 1.5 degrees along the azimuthal direction.

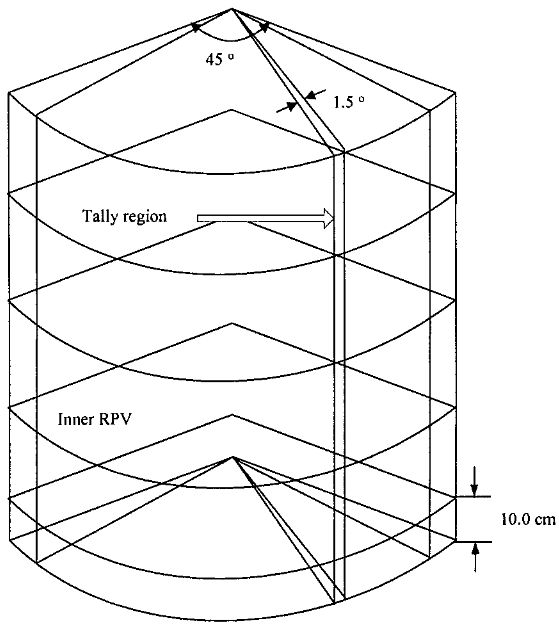

Figure 5. Tally Segments at the RPV Inner Wall

The fast neutron fluence (E > 1 MeV) distribution at the RPV inner wall surface of the KNGR cycle 8 core is shown Figure 6 and compared to the cycle 1 results [14]. Nearly q same trend has been observed in all cycle 8 calculations, except angle of maximum point. The RPV fulence of the cycle 1 has been higher then that of the cycle 8. The high enrichment flesh fuels are loading at all peripheral assemblies of cycle 1 core. In Full-Scope Monte Carlo method, it is observed that the maximum flux of the RPV beltline appears at ~14 degrees in azimuth, due to the peripheral assembly position within the core, and the value is given as $2.738 \times 10^{10} neutrons/cm^2 \cdot \sec$. In ROCS+MCNP4B calculation, the maximum flux of $2.769 \times 10^{10} neutrons/cm^2 \cdot \sec$ at the RPV has been obtained by tallying neutron crossing the inner surface of the RPV and the maximum flux appears at ~34 degrees in azimuth. In the simulations, ten million histories have been carried out. The relative errors of all tallies have been made within 5%.

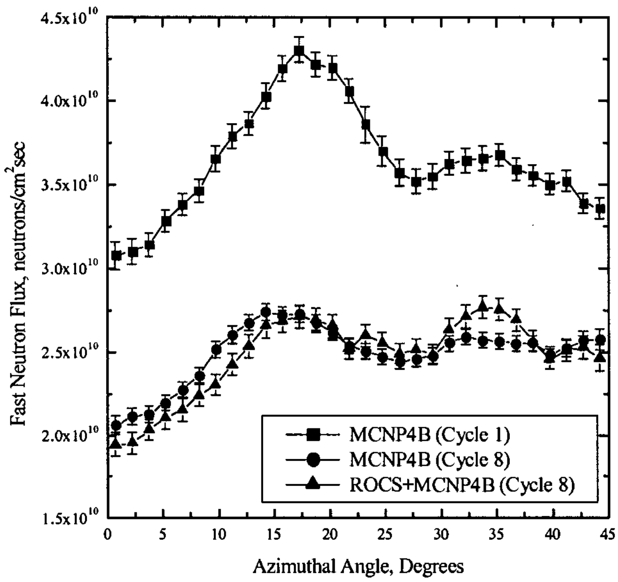

Figure 6. Flux Distribution at the KNGR RPV Beltline

CONCLUSIONS

The Full-Scope Monte Carlo simulation and ROCS+MCNP4B calculation have been applied to the KNGR RPV fluence calculation to assess quantitatively the lifetime of the reactor. The fuel loading pattern of cycle 8 core has been explicitly represented by pin-by-pin, and the calculation model has been verified by the neutron source criticality calculation using MCNP4B code. By formulating the neutron source probability distribution function from the source calculation (KCODE) and ROCS code results, the RPV fluence distributions have been calculated around the high flux beltline region.

The maximum fast neutron flux at the inner wall surface of the RPV has been calculated as $2.738 \times 10^{10} neutrons/cm^2 \cdot \sec$ and $2.769 \times 10^{10} neutrons/cm^2 \cdot \sec$, in Full-Scope Monte Carlo simulation and ROCS+MCNP calculation, respectively. 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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