Journal of NUCLEAR SCIENCE and TECHNOLOGY, Supplement 1, p. 125–129 (March 2000)

Evaluation of Vessel Fluence for SMART by DORT and MCNP Codes

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In Korea, an advanced reactor system of 330 MWt power called SMART (System integrated Modular Advanced ReacTor) is being developed by KAERI to supply energy for seawater desalination as well as electricity generation.

The conventional method using S_N transport codes such as DORT is currently used to evaluate vessel fluence. In this study, the discrete ordinates method is applied to estimate the neutron fluence at the pressure vessel of SMART in the conceptual design stage, and the Monte Carlo method is used to verify results estimated by the discrete ordinates method. According to the estimation by the above two methods, it was shown that the fast neutron fluence at the reactor pressure vessel of SMART is less than 1.0×10^{20} n/cm^2 and meets the design requirement of SMART.

KEYWORDS: SMART, Vessel Fluence, DPA, MCNP, DORT

I. Introduction

A project for the development of a system integrated modular advanced reactor of 330 MWt called SMART[1] was started in Korea in 1977 for supplying energy for seawater desalination as well as for electricity generation. The reactor assembly of SMART is shown in **Fig. 1**. In the stage of conceptual design, the fast neutron fluence at the reactor vessel of SMART was evaluated by using a conventional method, which is S_N transport code such as DORT[2]. The method contains uncertainties associated with multigroup libraries, geometric approximations, and pin power generation in the core. In this study, a coupled Monte Carlo analysis method was also applied for the verification of the SMART reactor vessel fluence using DORT. The coupled Monte Carlo method includes two parts of the Monte Carlo calculations. One is to determine the pin power distribution in the octant full core, another is to evaluate fast neutron fluence using pin power distribution from the first Monte Carlo result as a source term.

The estimation of vessel fluence provides core power distributions, material compositions, and geometry. In the case of the discrete ordinates method, MASTER[3], which is a nuclear design code system developed by KAERI estimates core power distributions.

The Monte Carlo simulation has advantages over the discrete ordinates method because of its decreased number of uncertainties resulting from the problems of the nuclear cross-section library, source term, and geometrical modeling.

Atom displacement per atom (DPA) is also estimated to review the verification of traditional fluence calculations.

II. Source Term

The SMART core consists of 57 fuel assemblies of a rectangular cross-section based on Korean optimized fuel assembly (KOFA) design technology[4]. The cross-sectional view of SMART core is shown in **Fig. 2**. The height of the fuel region of KOFA is 365.8 cm, but the reduced active height of 200 cm is to be applied in the SMART reactor. 17x17 KOFA design is chosen as the basis of the SMART fuel assembly.

The design parameters for KOFA are as follows:

(1) pitches of fuel assembly and fuel rod are 21.504 cm and 1.260 cm, respectively,

(2) number of fuel rods, Al_2O_3-B_4C shim rods and guide thimble/instrumentation tube for fuel assembly type A, C and D are 240, 24 and 25, respectively,

(3) number of fuel rods, Al_2O_3-B_4C shim rods and guide thimble/instrumentation tube for fuel assembly type B are 244, 20 and 25, respectively,

(4) material of the fuel pellet is UO_2 with 8.05 mm of diameter, and

(5) material of fuel cladding is Zircaloy-4 with 9.52 mm of outer diameter.

In the evaluation of fast neutron fluence using the conventional method, the pin power distribution of the SMART core was calculated using CASMO-MASTER code package.

The loading pattern shown in Fig. 2 was rearranged to coincide with the purpose of the vessel fluence calculation. Thus, the rearranged loading pattern has conservatism. The limiting assembly power distribution was selected with a minimum pin-to-box factor because the outer pin power is higher for the flat power distribution.

The assembly's relative power distribution with renormalized pinwise power distribution is shown in **Fig. 3**. The relative power distribution of interior assemblies is 1.3371, which is determined so that the average power density of the whole core is unity.

The source power in the R-Θ meshes was obtained from the renormalized pinwise power distribution by area weighting of the pins that belonged to each mesh. The relative power distribution and its energy spectrum normalized to the core averaged thermal power density of 62.60 W/cc give the actual

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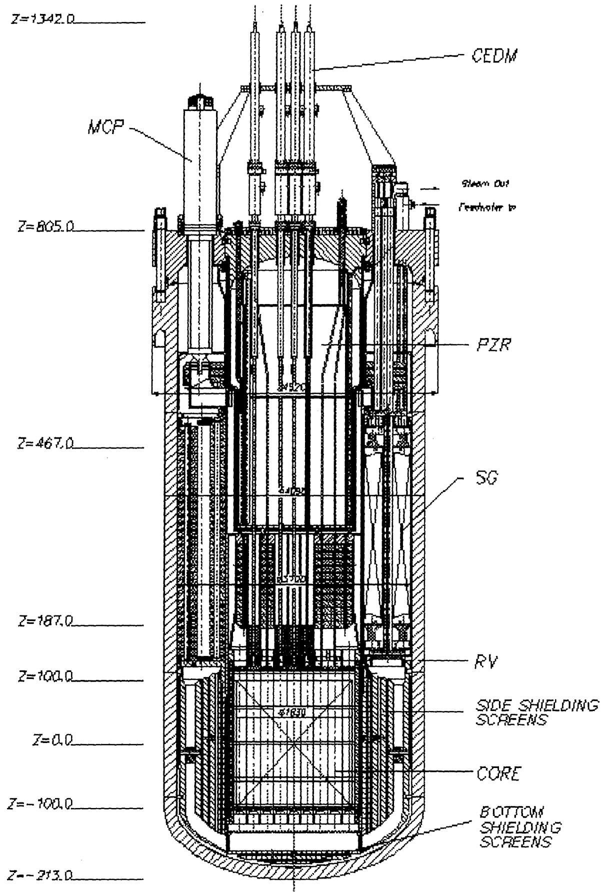

Fig. 1 Reactor Assembly of SMART.

	A	B	C	D	E	F	G	H	J
1				D	D	D			
2			B	A	B	A	B		
3		B	B	A	D	A	B	B	
4	D	A	A	D	A	D	A	A	D
5	D	B	D	A	C	A	D	B	D
6	D	A	A	D	A	D	A	A	D
7		B	B	A	D	A	B	B	
8			B	A	B	A	B		
9				D	D	D			

A 4.95w/o U-235 / 24 Al_2O_3-B_4C Shim
B 4.95w/o U-235 / 20 Al_2O_3-B_4C Shim
C 4.95w/o U-235 / 24 Al_2O_3-B_4C Shim / 4 Gd_2O_3-UO_2
D 4.95w/o U-235 / 24 Al_2O_3-B_4C Shim / 12 Gd_2O_3-UO_2

Fig. 2 Core Loading Pattern

neutron source rate per unit volume. The cycle maximum core average axial peaking factor is 1.6820[5], which will be used for only DORT calculation.

2. Monte Carlo Method

MCNP4A[6] code was used to evaluate pin power distribution in the SMART core. Fuel assemblies and fuel rods are modeled in MCNP input as they are in design. In the input of MCNP, a repeated structure by lattice option was used for 13,844 fuel rods and 57 fuel assemblies. The geometrical model for MCNP is considered as an octant of the core shown in **Fig. 4**, which was generated by MCNP directly.

	E	F	G	H	J
5	-	-	-	-	0.71
6		-	-	0.91	0.56
7			1.00	0.63	

Fig. 3 Relative Power Density Distribution

The cross-section libraries are taken from the MCNP recommended cross-section set, which are continuous energy cross-sections based on ENDF/B-V. However, the libraries of gadolinium isotopes are taken from ENDF/B-VI. The criticality source and track-length estimate tallies are used to pick up fission rate distribution from MCNP run. The track-length estimate of the cell-averaged fission rate is based on particle track-lengths through a given cell.

The tally information file option was used to generate the pin power distribution for each fuel rod. Then the pin power distribution for each fuel rod in an octant core was normalized using a small program from the tally information file. The normalized pin power distribution for 13,844 fuel rods of 57 fuel assemblies was used as the source distribution of the second MCNP calculation.

III. Transport Calculation

The SMART is composed of core, barrel, 6 side screens, reactor vessel, and other internals. DORT and MCNP models for transport calculation are shown in Fig. 4 and Fig. 5, respectively. **Figure 5** shows the cross-sectional diagram of a SMART and the dimensions for the internal structure of the pressure vessel.

The DORT was selected for the shielding design of SMART because it has been used extensively for the shielding design of power reactors over the years and it has been proven to be reliable for the reactor shielding design. The result from DORT evaluation of the fast neutron fluence was verified by using MCNP analysis because MCNP has been extensively useful as the benchmark for the results by discrete ordinates analysis. The fluence level was derived based on 60 years lifetime with 90% capacity factor from the fast neutron flux for energy more than 1.0 MeV.

1. Discrete Ordinates Method

The DORT was used to simulate R-Θ geometry of the SMART system and octant symmetry was assumed as shown in Fig. 4.

The 152 radial and 90 azimuthal meshes were applied in

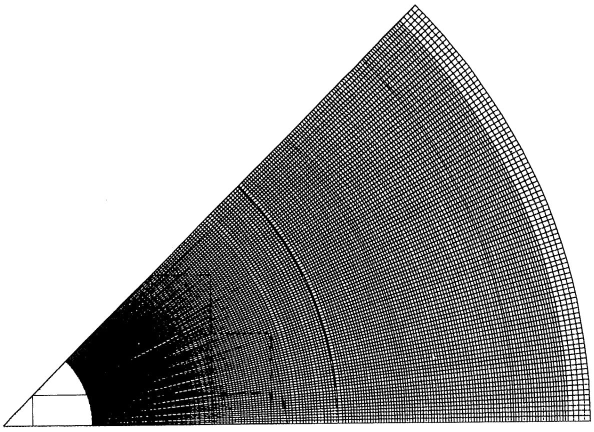

Fig. 4 Horizontal Cross-section of the DORT Model at the Core Mid-plane of SMART

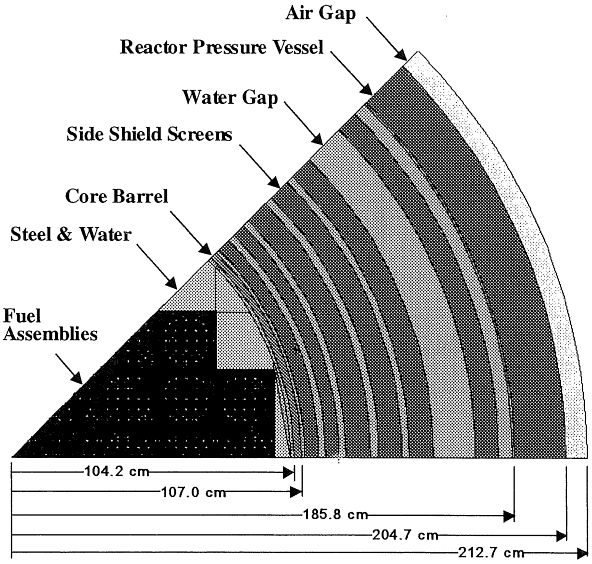

Fig. 5 Horizontal Cross-section of the MCNP Model at the Core Mid-plane of SMART

the model as shown in Fig. 4. The P_3 scattering expansion and S_8 angular quadrature were used for the DORT calculation. Because the neutron fluence on the reactor vessel originates mainly from the peripheral assemblies, the interior assemblies were not included in the model as the conventional procedure. For the energy spectrum, the Watt fission spectrum was used and the energy spectrum was normalized to thermal power density. DLC-23F[7] library was used for 40 group coupled neutron and gamma-ray cross-section data of DORT calculation.

Overall a 30% uncertainty was applied to the fluence at the vessel so that uncertainty due to dimensional tolerance, representation of source distribution and cross-section can be supplemented.

2. Monte Carlo Method

45 azimuthal segments for inner and outer surfaces of the vessel were applied in this modeling. The second MCNP model to calculate fast neutron fluence at reactor vessel contains the core geometry used in the first MCNP model for the power calculation. Therefore, the same surface numbers and cell numbers as in the first model are repeated in the second model.

The normalized pin power distribution taken from MCNP core calculation is used for the source term of the second MCNP input. A general source description was used to replace the pin power distribution with a fixed source term of MCNP. The energy distribution of the neutron source was randomly selected from the Watt fission spectrum of U-235. The tally for surface averaged flux was used to calculate neutron fluence penetrated inner and outer surfaces of the reactor vessel.

3. DPA Calculation

The structural neutron damage is the direct loss of ductility due to atom displacement within the material of a structural component. In this study, atom displacement per atom (DPA) is estimated with a more traditional fluence calculation. DPA is calculated in the following equation:

$$DPA = \sum [\sigma_{DPA}(E_i)\ \phi(E_i)\ t], \qquad i=1, 2, \ldots, 22 \qquad (1)$$

where $\sigma_{DPA}(E_i)$ is DPA cross-section for energy group i by Norgett-Robinson-Torrens model[8], $\phi(E_i)$ is neutron flux for energy group i taken from DORT result, and t is time in second.

IV. Results and Discussion

The azimuthal distributions of fast neutron fluence at the inner and outer surfaces of the reactor vessel are shown in **Fig. 6** and **Fig. 7**, respectively. The fast neutron fluence was integrated over 60 years with a 90% capacity. According to DORT analysis, the maximum fluence at the inner surface and outer surface of the SMART vessel is 2.88×10^{16} n/cm^2 at 3.75° and 1.66×10^{15} n/cm^2 at 1.25° of azimuthal angle from core center.

A coupled Monte Carlo analysis method was successfully applied to the SMART reactor. Forty-five azimuthal segments divided the surfaces of the reactor vessel, each with an angle of 1°. The maximum fluences were taken from the largest value among the values for the segments of inner surface and outer surface, which are 2.33×10^{16} neutrons/cm^2 and 1.77×10^{15} neutrons/cm^2 at the first segment of the inner surface and outer surface. These values are far less than 1.0×10^{20} neutrons/cm^2, the requirement specified by 10CFR50.61[9]. In this study, the pin power distribution was calculated under an assumption that the core is at the beginning of a cycle. The difference of the maximum fluence at the reactor pressure vessel between the two methods was within the 30 %. The difference of pin power distribution generation, geometrical modeling, and cross-section libraries between the two methods caused the difference of vessel fluence distribution. The maximum DPA at the inner surface of the reactor vessel derived by Eq. (1) is 4.10×10^{-5}. Therefore, it is shown that the maximum DPA is far less than 2.4 and meets the design requirement of SMART for the DPA[10].

In this stage, any measurements are currently not available for comparison. Moreover, other techniques besides the coupled Monte Carlo analysis technique are not applicable for benchmark of the discrete ordinates analysis. Therefore, the

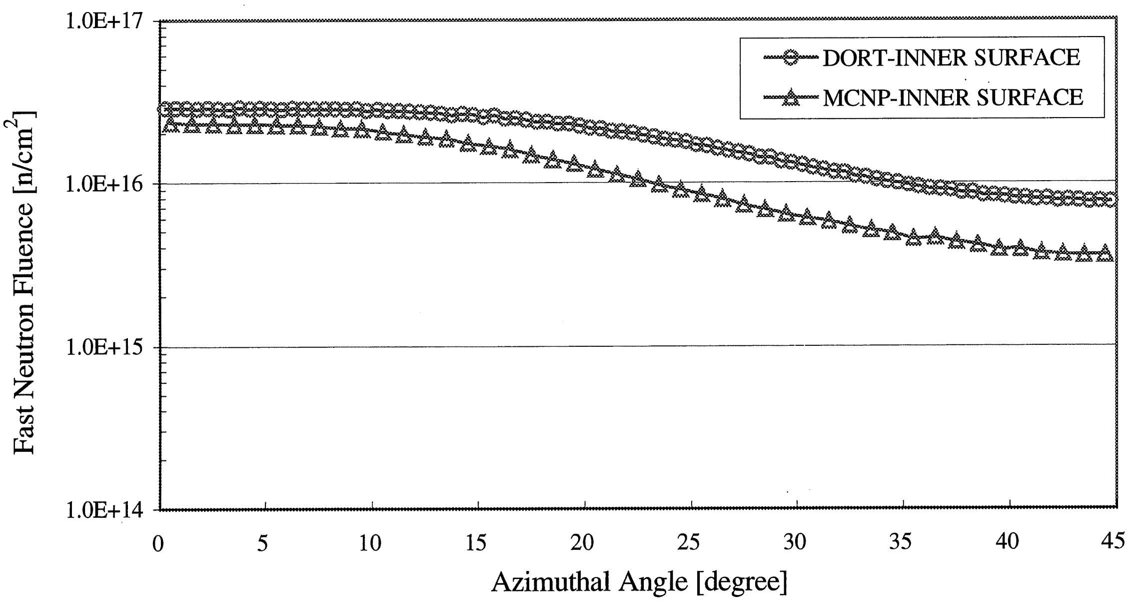

Fig. 6 Fast Neutron Fluence at Inner Surface of SMART

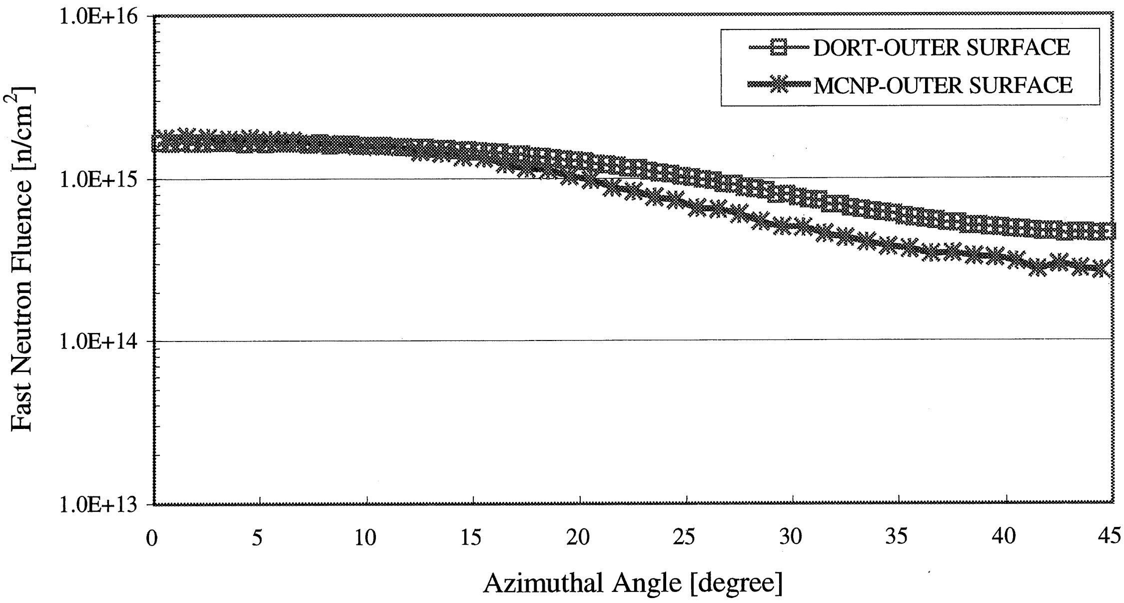

Fig. 7 Fast Neutron Fluence at Outer Surface of SMART

coupled Monte Carlo analysis technique carried out in this study is useful to verify that fast neutron fluence resulted from conventional technology.

V. Conclusions

In this study, the following conclusions are made:

1. The independent calculation of the reactor vessel fluence of SMART performed by Monte Carlo method showed that the design value by the discrete ordinates method met the requirement. Additionally, it was shown that the maximum DPA met the requirement. Therefore, it is concluded that the integrity of the SMART vessel is preserved throughout the lifetime of SMART.
2. The coupled Monte Carlo analysis method, including such calculations as criticality, source terms, and fast neutron fluence, was successfully applied to the vessel fluence evaluation for SMART. Because the MCNP calculation has the advantage of the consistency and accuracy of calculation, it is positively suggested that the code shall be applied to the reactor shielding design.

Acknowledgment

This work has been carried out under the Nuclear R&D Program by MOST.

— References —

(1) Seo, J. K., *et al.* : "Advanced Integral Reactor (SMART) for Nuclear Desalination", *IAEA-SM*-347/40 (1997).
(2) Rhoades, W. A., *et al.* : "The DORT Two-Dimensional Discrete Ordinates Transport Code System", *RSIC-CCC*-484 (1989).
(3) Lee, C. H., *et al.* : "MASTER User's Manual", *KAERI/UM*-3/98 (1998).
(4) KAERI : "Fuel Design Report for 17x17 Assembly", *KAERI/KWU*, (1987).
(5) Lee, C. C., *et al.* : *Proceedings of the Korean Nuclear Society Spring Meeting*, **Vol. I**, 19 (1998).
(6) Briesmeister, J. F., ed. : "MCNP-A General Monte Carlo N-Particle Transport Code", Version 4A, *LA*-12625-*M*, (1993).
(7) RSICC : *DLC-23F* (1978).
(8) Norgett, M. J. : *Nucl. Eng. Design*, **33**, 50 (1975).
(9) Nuclear Regulatory Commission : "Fracture Toughness Requirements for light Water reactor Vessel", 10CFR Part 50 (1994).
(10) Kim, K. Y., *et al.* : "Preliminary Shielding Design Evaluation for Reactor Assembly of SMART", *KAERI/TR*-1252/99 (1999).