Fast Neutron Fluence Evaluation Of The Smart Reactor Pressure Vessel By Using The Geoshield Code

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Abstract

In Korea, the design of an advanced integral reactor system called SMART has been developed by KAERI to supply energy for seawater desalination as well as an electricity generation. A fast neutron fluence distribution at the SMART reactor pressure vessel was evaluated to confirm the integrity of the vessel by using the GEOSHIELD code.
The GEOSHIELD code was developed by KAERI in order to prepare an input list including a geometry modeling of the DORT code and to process results from the DORT code output list. Results by a GEOSHIELD code processing and by a manual processing of the DORT show a good agreement.

1. Introduction

The SMART[1] is a small-sized advanced integral PWR that produces a thermal energy of 330 MWt under full power operating conditions. The primary components are integrated within a single pressure vessel, in which the arrangement of the components differs from that of the conventional loop-type reactors. New, innovative and highly advanced features adopted in the SMART design provide the reactor with noticeable characteristics of an enhanced safety, reliability, performance, and operability. The SMART is composed of a core, a barrel, 6 side-screens, 3 bottom-screens, 12 steam generators, a reactor vessel, and other internals as shown in Figure 1.

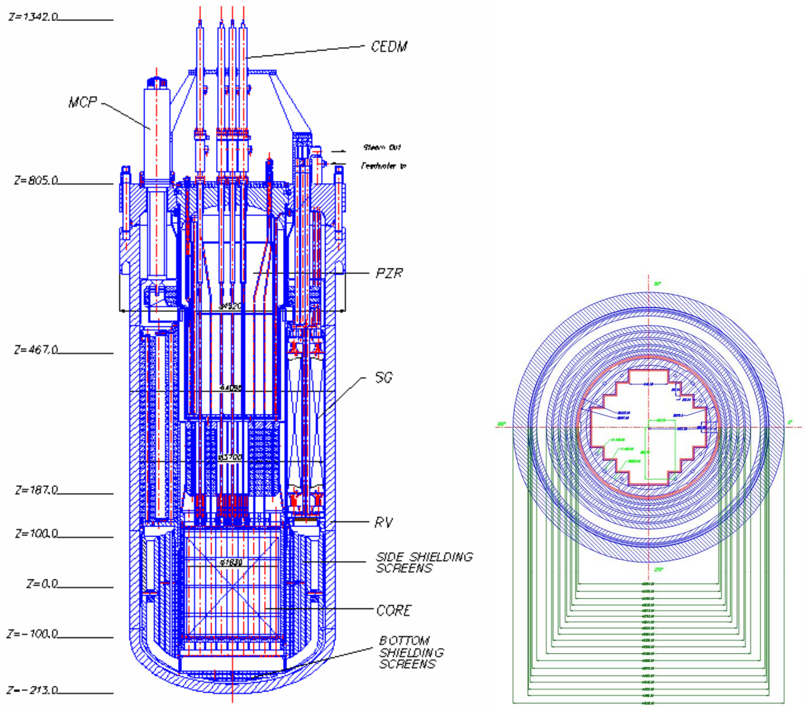


Figure 1 SMART Reactor Assembly and cross-sectional view

The shielding design is concentrated around the core to prevent the reactor components such as the reactor pressure vessel from fast neutron damage. The fast neutron fluence distributions in the reactor assembly around the SMART core are evaluated by using a conventional method in this study, which is a discrete ordinates transport code such as DORT code [2]. Also the time-averaged radial power distributions in the SMART core are evaluated by using the MASTER code [3], which is a nuclear design code developed by KAERI..

Since the DORT code is only able to handle several types of regular meshes, complex reactor core geometry should be converted into regular meshes. It takes a long time and it is cumbersome to generate meshes and to assign a specified composition and source for each mesh, which also may cause an error in preparing the DORT code input. In order to solve these problems we have developed the GEOSHIELD code [4]. The GEOSHIELD code includes various capabilities which are to generate the DORT input file, to process the DORT code output file and to visualize the graphics. The GEOSHIELD program was written in Fortran-95 with a graphic module of Quickwin.

2. GEOSHIELD Methodology

The overall procedure of the GEOSHIELD code is shown in Figure 2. This program includes the following functions:

(i) Original geometry construction by using various structures with a given composition and source
(ii) Mesh generation to be used in the DORT code calculation
(iii) Composition and source assignment to each mesh
(iv) DORT code input generation
(v) DORT code output processing
(vi) Graphical visualization

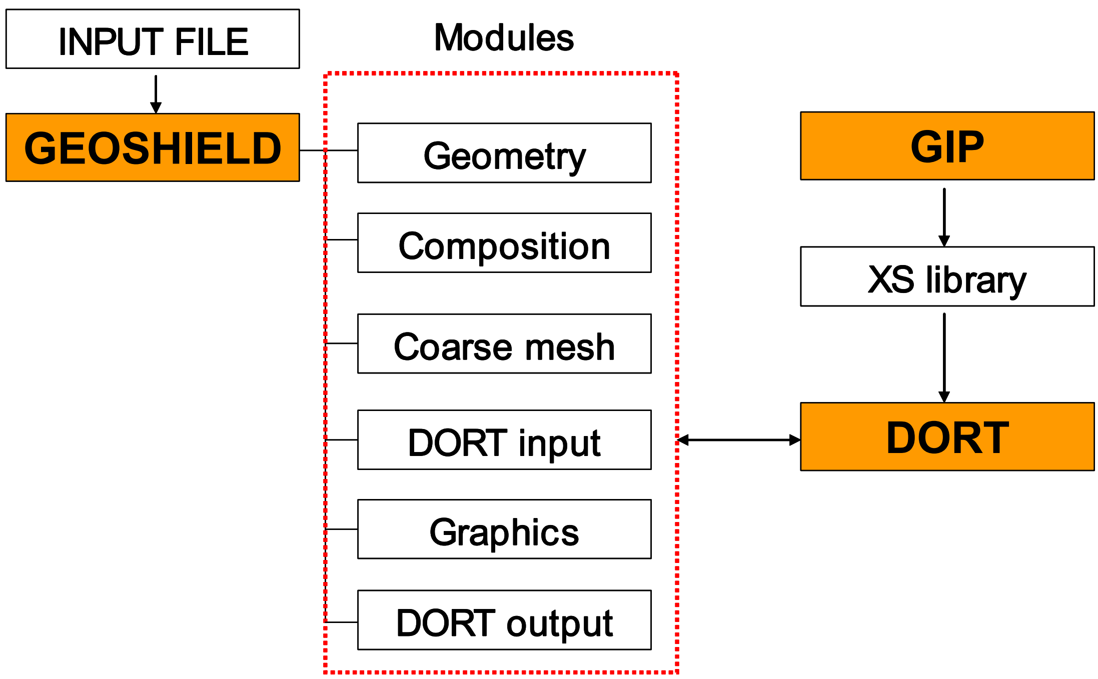


Figure 2 Flow chart of the GEOSHIELD calculation

The original geometry can be constructed by a combination of structures which consists of basic cells. The basic cell types are polygon, ellipse, and pie geometries. A structure is constructed by a combination of the basic cells, and especially rectangular or hexagonal lattice arrays can be constructed as shown in Figure 3.

The fundamental idea for the mesh generation includes three steps. The first one is to construct the original core geometry with various combinations of structures consisting of basic cells including polygon, circle, ellipse and pie geometries. Compositions and sources are assigned to each structure. The second one is to construct the regular meshes to be used in the DORT code transport calculations and to determine the centre points of the regular meshes. The last step is to establish which structures the centre points of the regular meshes are located in. The composition and source of a specified structure are assigned to regular meshes as shown in Figure 4. These processes are to be visualized by the GEOSHIELD code. All the other DORT inputs are also controlled by the GEOHIELD code input. The DORT outputs are also controlled by the GEOSHILD code. Activities resulted from adjoint and forward calculations Assembly weighting factors and shape annealing functions can be edited automatically and other activities can be visualized by the GEOSHIELD code.

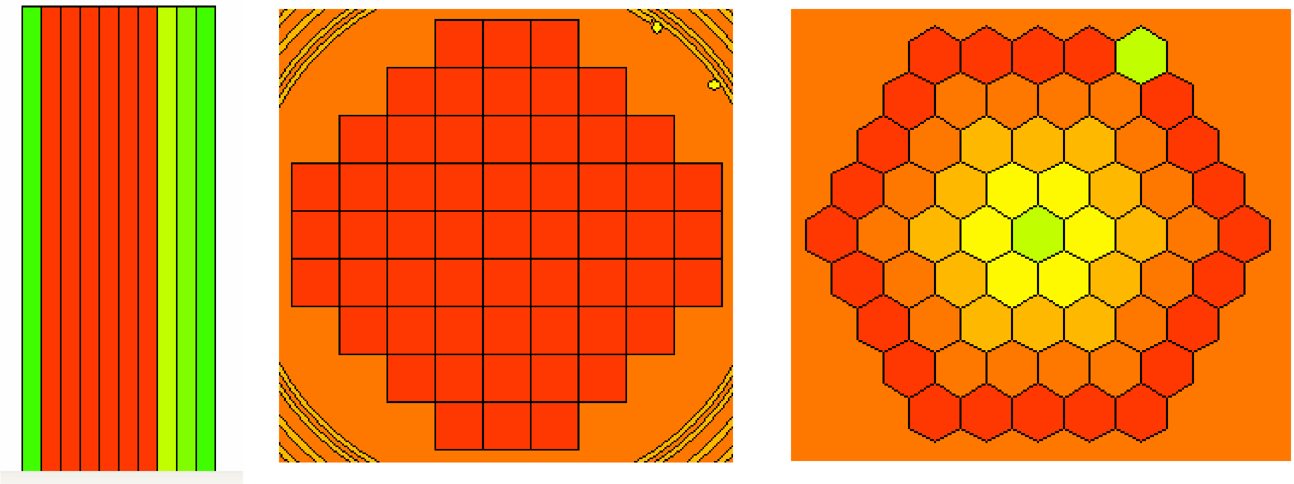


Figure 3 Lattice array

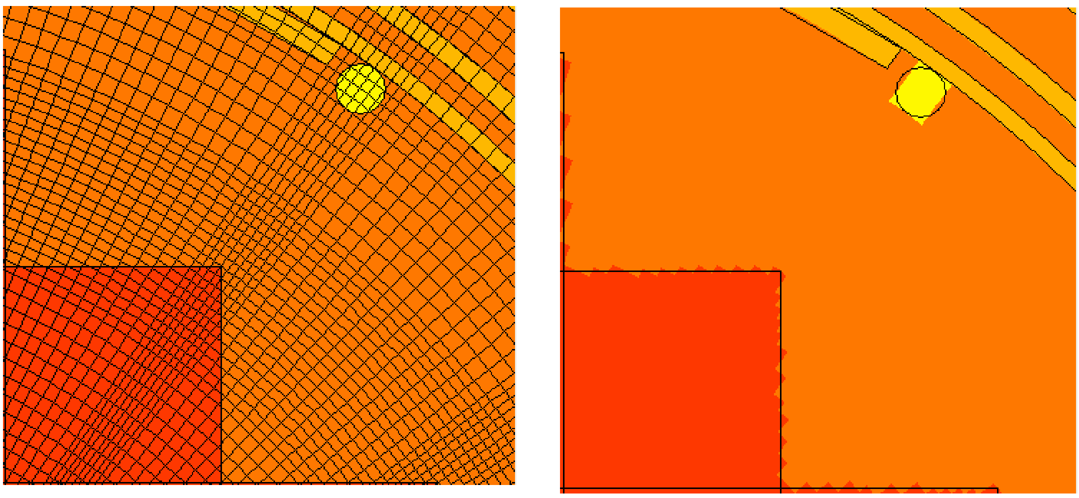
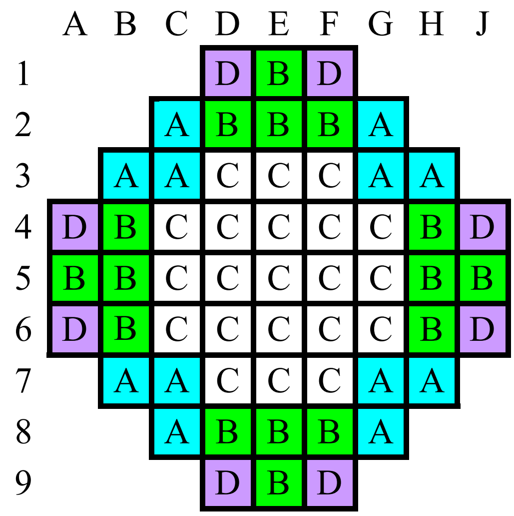


Figure 4 Centre point allocation

3. Source Term Generation

The SMART core consists of 57 fuel assemblies of a rectangular cross-section based on Korean optimized fuel assembly (KOFA) design technology [5]. A cross-sectional view of the SMART core is shown in Figure 5. 17x17 KOFA design is chosen as the basis of the SMART fuel assembly. The height of the fuel region of KOFA is 365.8 cm, but a reduced active height of 200 cm is to be applied to the SMART reactor. The time averaged power distributions of the fuel assemblies in the SMART core are derived from the power distributions calculated by using the MASTER code as shown in Table 1.



A **4.95 $^w/_o$ U-235 / 20 Al_2O_3-B_4C Shim**
B **4.95 $^w/_o$ U-235 / 20 Al_2O_3-B_4C Shim / 4 Gd_2O_3-UO_2**
C **4.95 $^w/_o$ U-235 / 24 Al_2O_3-B_4C Shim / 4 Gd_2O_3-UO_2**
D **4.95 $^w/_o$ U-235 / 28 Al_2O_3-B_4C Shim / 12 Gd_2O_3-UO_2**

Figure 5 The loading pattern of the SMART core

Table 1 Time Averaged Power distribution from the MASTER code

	A	**B**	**C**	**D**	**E**	**F**	**G**	**H**	**J**
1	0.00	0.00	0.00	0.53	0.70	0.53	0.00	0.00	0.00
2	0.00	0.00	0.74	1.07	1.18	1.07	0.74	0.00	0.00
3	0.00	0.74	1.17	1.24	1.27	1.24	1.17	0.74	0.00
4	0.53	1.07	1.24	1.26	1.19	1.26	1.24	1.07	0.53
5	0.70	1.18	1.27	1.19	1.18	1.19	1.27	1.18	0.70
6	0.53	1.07	1.24	1.26	1.19	1.26	1.24	1.07	0.53
7	0.00	0.74	1.17	1.24	1.27	1.24	1.17	0.74	0.00
8	0.00	0.00	0.74	1.07	1.18	1.07	0.74	0.00	0.00
9	0.00	0.00	0.00	0.53	0.70	0.53	0.00	0.00	0.00

4. Transport Calculation

The GEOSHIELD code was used to simulate the R-Θ geometry of the SMART reactor assembly as shown in Figure 6 and an octant symmetry was assumed as shown in Figure 7. The GIP code [6] and DORT code are selected for the shielding design of the SMART reactor because they have been used extensively for the shielding design of power reactors over the past years and they have been proven to be reliable for a reactor shielding design. Two-dimensional R-Θ geometry models are used in DORT transport analyses. The R-Θ models include most of the reactor components with the consideration of an azimuthally homogeneous core configuration. The geometrical model extends radially from the core centreline to the outer surface of the reactor vessel.

152 radial and 90 azimuthal meshes were used in the R-Θ model as shown in Fig. 8. A P_3 scattering expansion and a S_8 angular quadrature set were used for the DORT code calculation. For the energy spectrum, the Watt fission spectrum normalized to the thermal power density was used. BUGLE-96 library [7] was used for the 67 group coupled neutron and gamma-ray cross-section data for the DORT code calculation, which consists of 47 neutron and 20 gamma energy groups. The fluence level was derived based on a 60 years lifetime with a 90% capacity factor from the fast neutron flux for an energy of more than 1.0 MeV.
Overall a 30% uncertainty was applied to the fluence at the vessel so that an uncertainty due to a dimensional tolerance, representation of a source distribution and cross-section can be supplemented.

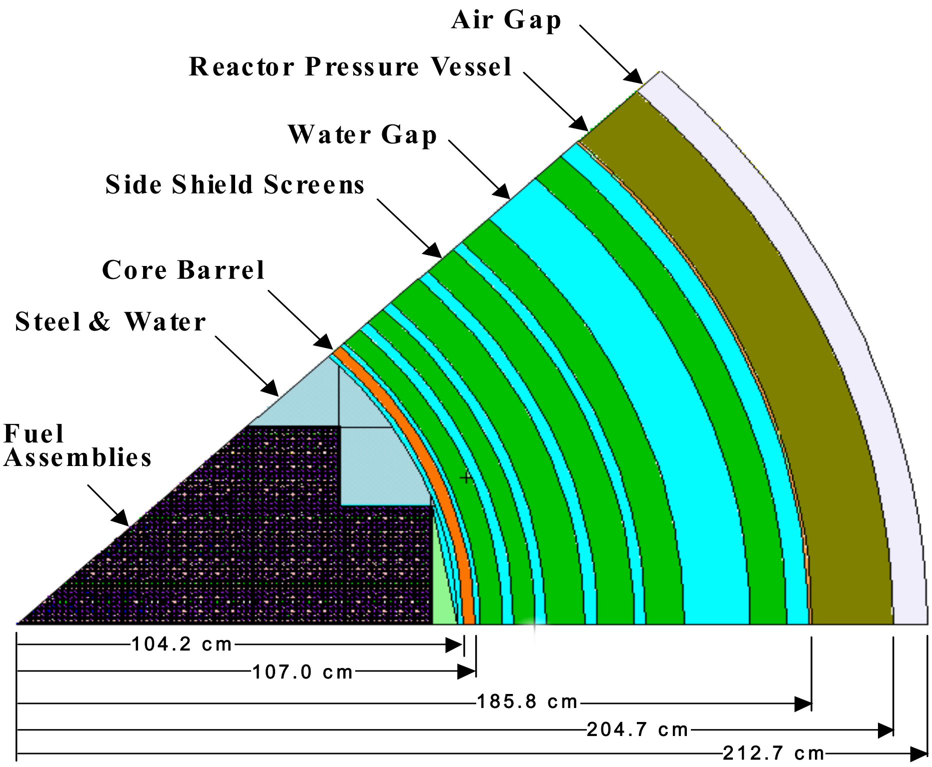


Figure 6 Simplified Reactor Structure of SMART for the GEOSHIELD Modelling

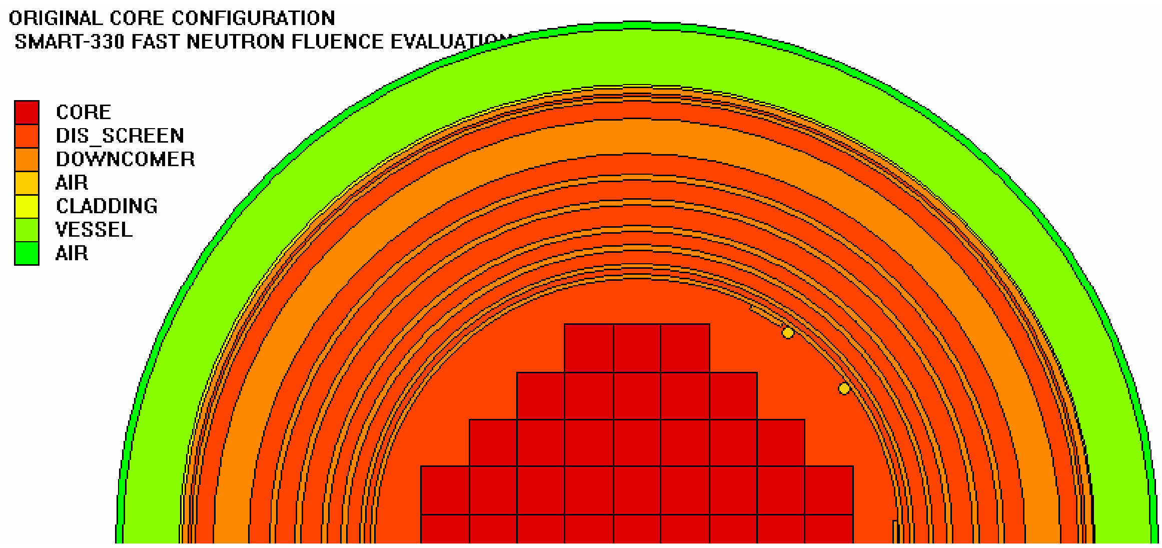


Figure 7 Original reactor configuration for the DORT model using GEOSHIELD

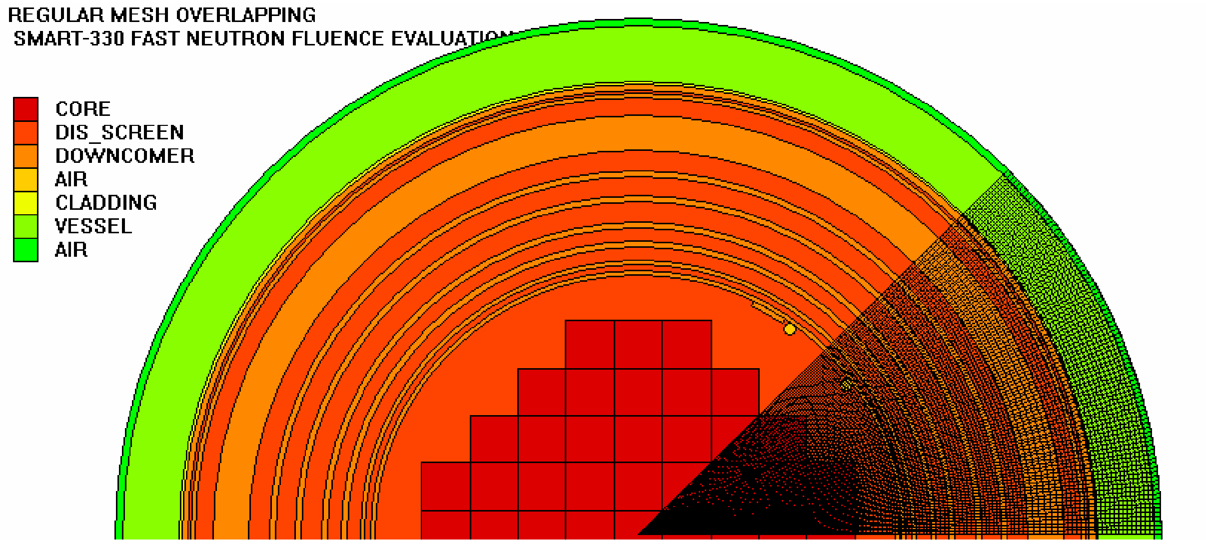


Figure 8 Regular mesh overlapping for the DORT model using GEOSHIELD

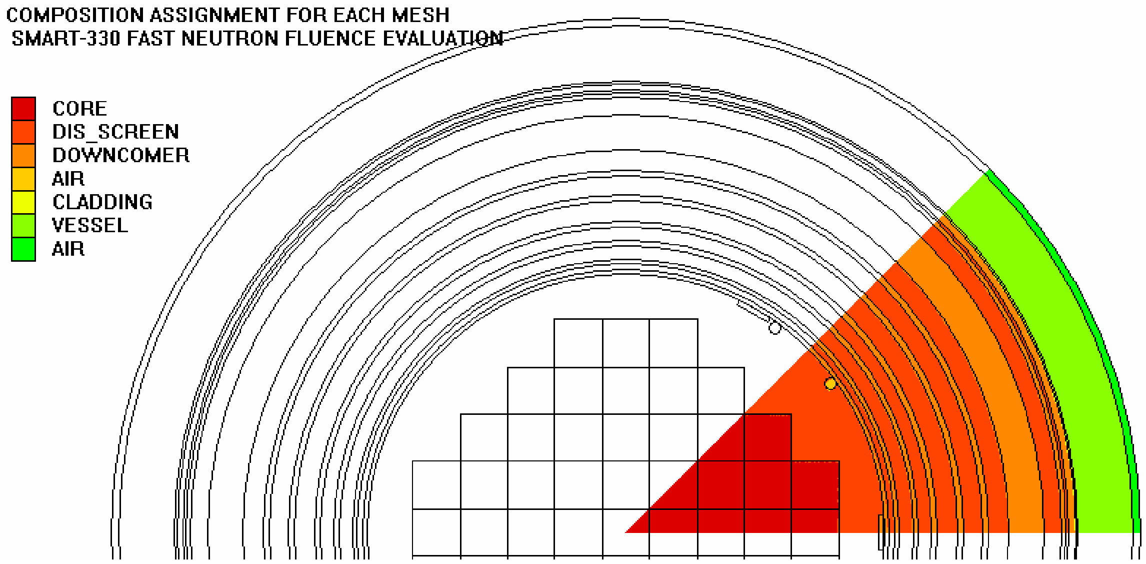


Figure 9 Composition assignment for each mesh of the DORT model using GEOSHIELD

4. Result and Discussion

In order to evaluate the fast neutron fluence distribution in the SMART reactor assembly, the DORT input file was generated automatically and the output file was processed by using GEOSHIELD program, which has been developed recently. Graphical visualization of the GEOSHIELD program helps users to prepare easily for the DORT calculations and to summarize the DORT outputs.

The fast neutron fluence distributions at the inner surface and the outer surface of the reactor vessel generated by the GEOSHIELD code are shown in Figure 10. The fast neutron fluence distributions by using the GEOSHIELD code were compared with those previously generated by a manual method [8]. The fast neutron fluence distributions by the GEOSHIELD code are very consistent with those by the manual method.

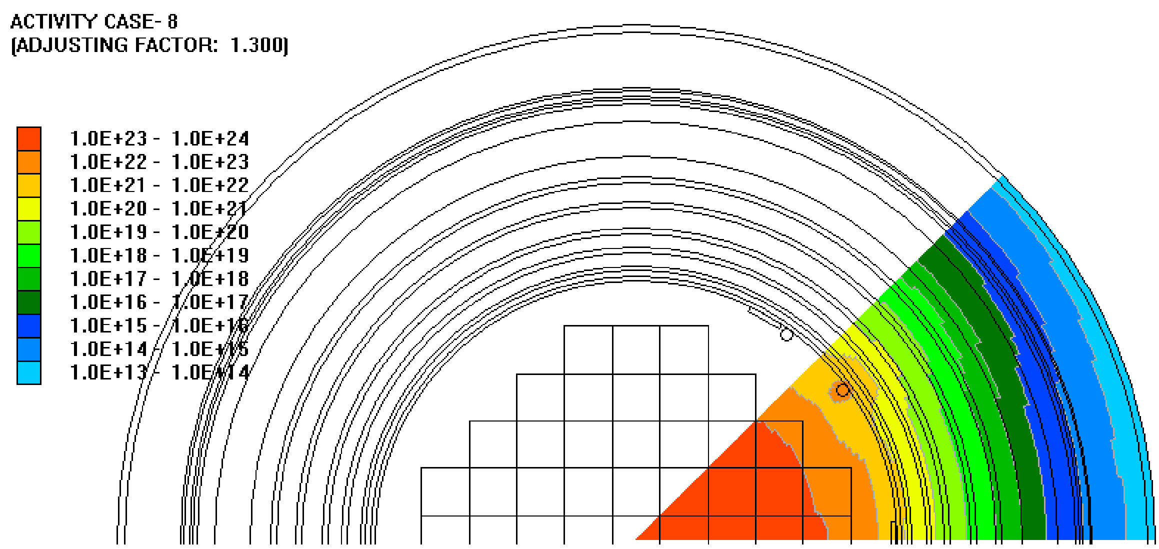


Figure 10 Fast neutron fluence distribution using GEOSHIELD code system [neutrons/cm^2]

The maximum fluences were taken from the largest value among the values for the segments of the inner surface and outer surface, which were 5.0×10^{15} neutrons/cm^2 and 7.0×10^{13} neutrons/cm^2 at the first segment of the inner surface and outer surface respectively. These values are far less than 1.0×10^{20} neutrons/cm^2, the requirement specified by the Korean Atomic Energy Law[9] and U.S. SRP[10]. Therefore it is concluded that the integrity of the SMART vessel is preserved throughout the lifetime of SMART.

The GEOSHIELD code was successfully applied to a vessel fluence evaluation for SMART. Because the GEOSHIELD calculation has the advantage of a consistency and time saving for a modelling, it is suggested that the code could be applied to a reactor shielding design.

7. Conclusion

In order to evaluate the fast neutron fluence distribution in the SMART reactor assembly, the DORT input file was generated automatically and the output file was processed by using the GEOSHIELD code, which has been developed recently. Graphical visualization of the GEOSHIELD code helps users to prepare easily for the DORT calculations and to summarize the DORT outputs.

The independent calculation of the reactor vessel fluence of SMART performed by the GEOSHIELD code showed that the design value by the conventional method met the requirement.
Fast neutron fluence distributions at the inner surface and the outer surface of the reactor vessel generated by the GEOSHIELD code were compared with those previously generated by a manual method. The fast neutron fluence distributions by the GEOSHIELD code were very consistent with those by the manual method.

8. Acknowledgement

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