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# Study for Benchmark Calculation using TORT code in BWR Reactor Vessel

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Sensitivity analyses have been made to improve accuracy of TORT calculations. Reasonable calculation conditions such as number of energy groups, flux convergence criterion, number of angular directions, order of Legendre expansions and the effective densities were investigated. No significant differences are identified from the calculation conditions and neutron energy group structure. It is possible to confirm that the water density of the outer bypass region and the annulus area from the shroud to the RPV is an important parameter for accurate estimation of fast neutrons.

***KEYWORDS: BWR type reactor, neutron flux, Sn code, cross section, dpa, fast neutron***

## I. Introduction

An accurate evaluation of the radiation field inside the BWR reactor vessel is essential for accurate lifetime prediction of structural materials such as vessel steel and core internals. Recent improvements in the computational environment have made it possible to get three-dimensional (3-D) flux distributions in the BWR reactor vessel using the 3-D Sn code TORT[1]. To validate applicability of TORT to radiation analysis in BWR, benchmark analysis has been done using dosimetry measurements in the reactor vessel. The reliable measured data at the various locations in the annular region between the shroud and the RPV at KKM (a BWR-4 plant in Switzerland operated by BKW) were obtained as reaction rates; these are referred to as the KKM measurement data. From the benchmark tests, the spatial distribution of computed to measured results(C/M ratios) for thermal neutrons agreed well with the measured data [2-4].

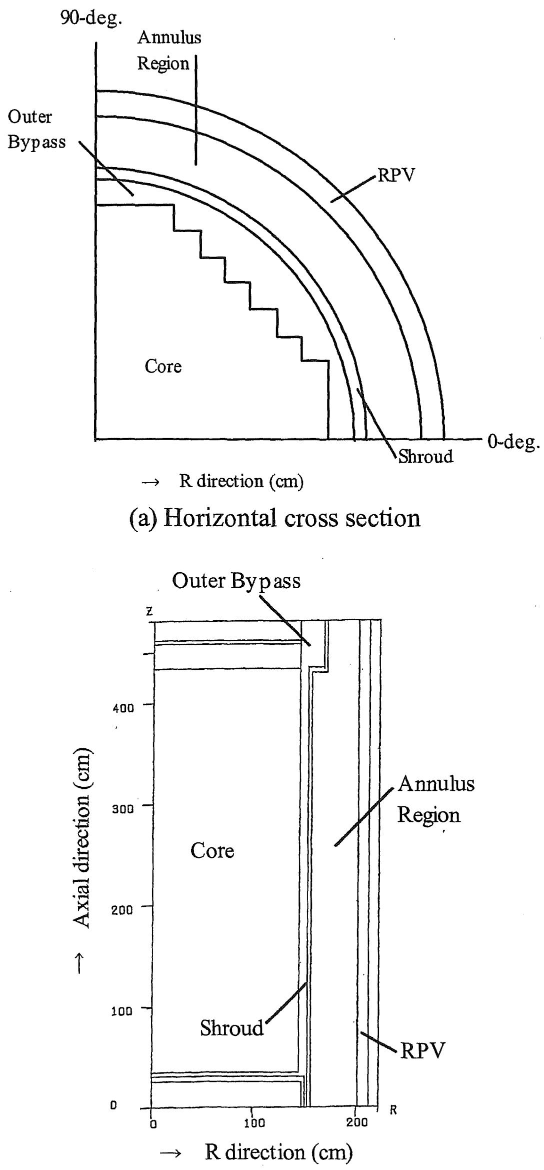
In this paper, further sensitivity analyses were made to improve the TORT calculation accuracy. A methodology to estimate neutron flux distributions in the reactor pressure vessel (RPV) by TORT was validated.

## II. Further benchmark analysis using measurements in the reactor vessel

Previously, it was found that TORT gives reliable results from the benchmark tests using KKM measurement data[5-6]. In comparison with two-dimensional calculations, TORT calculations improved the accuracy at least 20%. However, the discrepancy for fast neutron flux was still 30%. In order to improve the C/M for fast neutrons, suitable calculation conditions for TORT were surveyed.

### 1. Calculation procedures

TORT calculations for KKM plant were performed using a quarter-cylindrical R-θ-Z model that covered an area from the core to the RPV. The model geometry is shown in **Fig.1**.




(a) Horizontal cross section



(b) Vertical cross section

**Fig.1** Geometry in R-θ-Z model

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The axial calculation area was from the bottom of core plate to the top of the fuel bundle. In the active fuel region, pellets, claddings, water rods, and other materials are considered and modeled as a homogeneous material in each cell area. The axial void distribution that is averaged in the radial direction is also considered. The actual burn-up distribution during the irradiation period was given to TORT as an input condition of the neutron source distribution. The water densities of these areas were obtained from plant design data. The effective densities of components in the reactor vessel were used. The number of mesh intervals of this calculation model was 84 x 60 x 67 (Radial x Axial x Azimuthal). The calculations were done as a fixed neutron source calculation with P5 expansion of the angular dependence of the scattering cross sections, and a symmetric S12 directional quadrature set.

## 2. Sensitivity analysis for TORT input parameters

To validate the optimum calculation conditions for TORT, several calculations with different conditions, such as number of energy groups, flux convergence criterion, number of angular directions ($S_n$) and order of Legendre expansions ($P_L$), were done.

### 2.1 Impact of neutron energy group structure

Twenty-six energy group constants are used for the cross section data set for the three - dimensional transport calculation. These cross section data are obtained from the JSSTDL library[7] (100 groups for neutrons, 40 groups for gamma rays) which based on JENDL-3.2[8]. Two kinds of calculations by ANISN[9], a one-dimensional Sn code, are performed with the two geometry models focusing on the radial direction of the core center plane and the axial direction of the core center axis to collapse 100 energy groups to 26 energy group cross section sets. The threshold energy of activation reaction was influenced to the reaction rates, so the impact of energy group structure on the reaction rate was examined. The neutron flux distributions at the shroud, midplace between shroud and RPV and near the RPV were calculated with the two cross section, 26 energy group constants and a 100 energy group constants. The calculated neutron spectra at 4 degree are shown in **Fig.2**. The activation foils were put in the capsules and these capsule Ids indicate their location:"L", "M", and "H" denote the low-, mid-, and high-axial positions and "S", "C", and "R" represent the shroud, mid-annulus, and RPV, respectively.

The reaction cross sections of $^{109}$Ag (n, γ) and $^{54}$Fe(n, p), which were taken from the JENDL Dosimetry File 99 (JENDL/D-99)[10] library are shown in **Fig.3** and **Fig.4**. The summary of the impact of energy group structure of cross section to the reaction rates is listed in **Table 1**. Comparison of the reaction rates of the 26 energy group structure to that of 100 energy group structure shows good agreement between them. The difference between both calculations is within 2 %.

### 2.2 Impact of calculation conditions

The calculation conditions, such as order of Legendre expansion and number of angular directions and flux convergence, were surveyed. By combining the following parameters, two $P_L$ parameters, 3 $S_n$ parameters and 4 flux convergence parameters, 7 different calculations, as shown in **Table 2**, were done. The impact of calculation conditions in the 3-D Sn calculation on the results is summarized in **Table 3**. As shown in **Table 3**, the calculated results have no discrepancy within 3% deviation. The difference between JENDL3.1 and JENDL3.2 cross section data sets was also investigated. Any difference in calculated reaction rates with the two libraries is negligible.

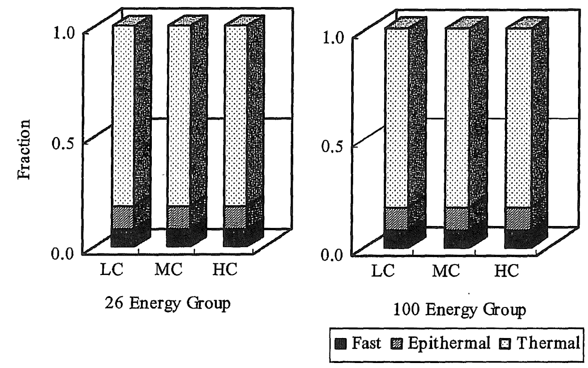


**Fig.2** The calculated neutron spectra at 4 degree

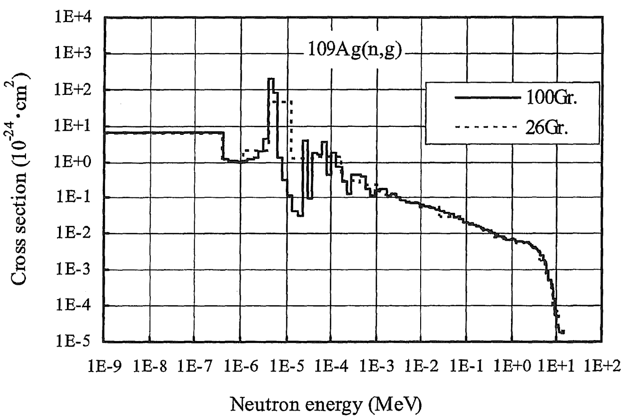


**Fig.3** Cross section of $^{109}$Ag(n,γ) taken from JENDL/D99

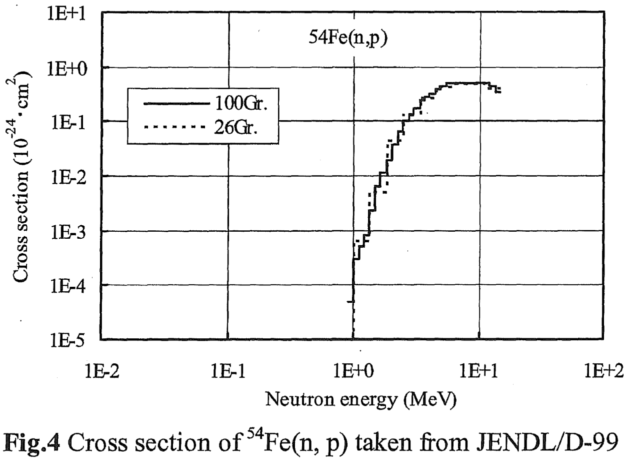


**Fig.4** Cross section of $^{54}$Fe(n, p) taken from JENDL/D-99

**Table 1** Summary of the effect of energy group structure of cross section in the 3-D Sn calculation at 4 degree

| Capsule location | C(26 energy groups) / C(100 energy groups) ratio (Reaction rate) | | | | | |
|---|---|---|---|---|---|---|
| | $^{109}Ag$ | $^{59}Co$ | $^{54}Fe$ | $^{58}Fe$ | $^{93m}Nb$ | $^{45}Sc$ |
| LS | 0.96 | 0.99 | 1.00 | 0.99 | 0.99 | 0.99 |
| MS | 0.96 | 0.99 | 1.00 | 0.99 | 0.99 | 0.99 |
| HS | 0.96 | 0.99 | 1.00 | 0.99 | 0.99 | 0.99 |
| LC | 0.98 | 0.98 | 0.99 | 0.98 | 0.98 | 0.98 |
| MC | 0.98 | 0.98 | 0.99 | 0.98 | 0.98 | 0.98 |
| HC | 0.98 | 0.98 | 0.99 | 0.98 | 0.98 | 0.98 |
| LR | 0.97 | 0.98 | 0.97 | 0.98 | 0.97 | 0.98 |
| MR | 0.97 | 0.98 | 0.97 | 0.98 | 0.97 | 0.98 |
| HR | 0.97 | 0.98 | 0.97 | 0.98 | 0.97 | 0.98 |

Significant differences are identified for the attenuation ratios of the calculated neutron flux, compared with measured data at the shroud to the RPV. The C/M ratios for fast neutrons ranged from 0.6 to 0.9[4]. The modeled water density in this annulus area seems to be the cause of this discrepancy. The water density distribution in the annulus area that cannot be measured is modeled with the same data as for the plant design. This may possibly cause some difference. The neutron flux calculations were done with two data sets, case1 was a modeled water density that was considered with the density distribution and case2 did not consider the density distribution. The differences between the attenuation neutron flux that is considered with and without the water density distribution in the annulus area are almost 10 % at the RPV surface. It was found from the above validation that a modeled water density in annulus area would influence the neutron flux distribution in the RPV. The C/M ratios at the shroud are also different. The water density of the outer bypass region that is from out of core to the shroud has a possibility to cause the discrepancy of neutron flux at the outer surface of shroud. The calculated neutron flux distribution with two different water densities at the outer bypass region is shown in **Fig.5**. As shown there, the fast neutrons at the outer shroud are different, but the attenuations of neutron flux between the shroud to the RPV show almost the same tendency. It is possible to confirm that the water density of the outer bypass region and the annulus area from the shroud to the RPV is an important parameter for accurate estimation of fast neutrons.

## III. Conclusion

From the sensitivity analysis for TORT input parameter, it was found that the calculation conditions for TORT such as number of energy groups, flux convergence criterion, number of angular directions ($S_n$) and order of Legendre expansions ($P_L$) did not affect so much neutron flux distributions in the BWR reactor vessel because there is around 2 to 4 % difference in **Table 1**.

The water density of the outer bypass region and the annulus region in the reactor vessel was sensitive for the fast neutron distribution, and further investigation on the optimum water density set up is needed.

**Table 2** Calculation cases for neutron flux distribution

| Case No. | $P_L$ | Sn | Flux convergence criteria |
|---|---|---|---|
| 1 | P5 | S12 | 1.00E-02 |
| 2 | P5 | S12 | 1.00E-03 |
| 3 | P5 | S12 | 1.00E-04 |
| 4 | P5 | S12 | 1.00E-05 |
| 5 | P3 | S12 | 1.00E-02 |
| 6 | P5 | S16 | 1.00E-02 |
| 7 | P5 | S6 | 1.00E-02 |

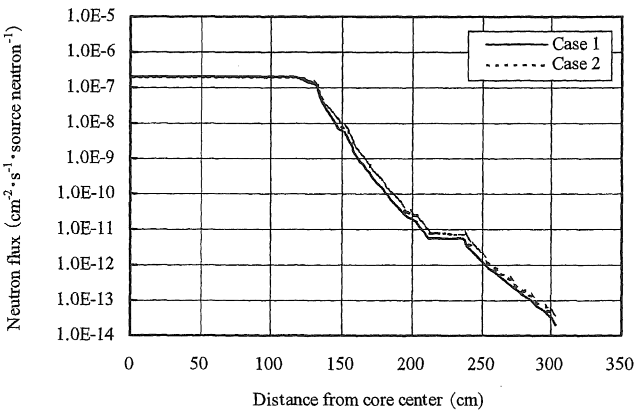


**Fig.5** Fast neutron flux distribution with different water densities in the radial direction

**Table 3** Summary of the effect of calculation parameters in the 3-D Sn calculation at 4 degree

| Parameters | Flux convergence criteria | | | | | | P3/P5 | | S16/S12 | | S6/S12 | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Capsule location | Case 2/Case 1 | | Case 3/Case 1 | | Case 4/Case 1 | | Case 5/Case 1 | | Case 6/Case 1 | | Case 7/Case 1 | |
| | Fast | Thermal | Fast | Thermal | Fast | Thermal | Fast | Thermal | Fast | Thermal | Fast | Thermal |
| LS | 1.00 | 1.00 | 1.00 | 1.00 | 1.00 | 1.00 | 1.00 | 1.00 | 1.00 | 1.01 | 1.00 | 0.99 |
| MS | 1.00 | 1.00 | 1.00 | 1.00 | 1.00 | 1.00 | 1.00 | 1.00 | 1.00 | 1.01 | 1.00 | 0.99 |
| HS | 1.00 | 1.00 | 1.00 | 1.00 | 1.00 | 1.00 | 1.00 | 1.00 | 1.00 | 1.01 | 1.01 | 0.99 |
| LC | 1.00 | 0.99 | 1.00 | 0.99 | 1.00 | 0.99 | 1.00 | 0.99 | 1.00 | 1.02 | 0.99 | 0.99 |
| MC | 1.00 | 0.99 | 1.00 | 0.99 | 1.00 | 0.99 | 1.00 | 0.99 | 1.01 | 1.02 | 0.98 | 0.99 |
| HC | 1.00 | 0.99 | 1.00 | 0.99 | 1.00 | 0.99 | 1.00 | 0.99 | 1.00 | 1.02 | 1.00 | 0.99 |
| LR | 1.00 | 0.99 | 1.00 | 0.99 | 1.00 | 0.99 | 1.00 | 0.99 | 1.01 | 1.02 | 0.97 | 0.98 |
| MR | 1.00 | 0.99 | 1.00 | 0.99 | 1.00 | 0.99 | 1.00 | 0.99 | 1.01 | 1.03 | 0.96 | 0.98 |
| HR | 1.00 | 0.99 | 1.00 | 0.99 | 1.00 | 0.99 | 1.00 | 0.99 | 1.00 | 1.02 | 0.97 | 0.97 |

## Acknowledgements

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