Influence of Cross-section on the Estimated Source Terms of Side-Structural Components in CANDU Reactor

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INTRODUCTION

Radwaste inventory from a nuclear reactor to be decommissioned comprises of neutron-activated materials and contaminated materials. The amount produced by activation is quite large as compared to the contamination products [1]. Computer codes have been developed to estimate the radionuclide inventory attributed to neutron activation with reasonable accuracy. These computer codes use their own build-in cross-section and decay data libraries for the calculation of buildup and decay of a radionuclide in irradiated material. The built-in cross-section libraries generally contain spectrum-averaged one-group cross-sections for various nuclear reactions computed at certain reactor condition and therefore seem inappropriate for inventory calculations in far-core regions where the neutron spectrum is significantly different from that of in-core region.

In this paper, we have analyzed the influence of spectrum-averaged capture cross-sections on estimated source terms for side structural components of a CANDU reactor, using MCNP/ORIGEN2 system which is a candidate computer code for radionuclide inventory calculations.

METHODOLOGY

The computation of source terms attributed to neutron activation is a two step process [1]: (1) calculation of spatial and energy distribution of neutrons in the core and (2) calculation of reaction rates. A nuclide buildup under various nuclear reactions, considering its production and destruction can be simply represented as,

$$\frac{dX_i}{dt} = \sum_{j=1}^{N} l_{ij}\lambda_j X_j + \phi \sum_{k=1}^{N} \sigma_k X_k - (\lambda_i + \phi\sigma_i) X_i \qquad (1)$$

where,

X_i = atom density for nuclide i, l_{ij} = decay fraction of nuclide j to nuclide i, σ_i = spectrum-averaged neutron capture cross-section for nuclide i, λ_i = decay constant for nuclide i and ϕ =spectrum-averaged one-group neutron flux

ORIGEN2 [2] utilizes the matrix exponential method to compute radionuclide inventory using Eq. (1). The neutron flux for Eq. (1) can be supplied from any reactor physics code. We utilized the MCNP [3] for this purpose due to its greater ability to accurately predict neutron flux in the far-core regions such as side structural components [4]. Also, Eq. (2) was used for the generation of spectrum-averaged capture cross-sections in MCNP calculations.

$$\bar{\sigma} = \frac{\int\int \sigma(E)\phi(E)dEdV}{\int\int \phi(E)dEdV} \qquad (2)$$

where,

$\bar{\sigma}$ = one-group collapsed cross-section, σ (E)= energy-dependent capture cross section and ϕ (E)= energy-dependent neutron flux.

One channel of the side structural component including liner tube, shield plugs, end fittings, etc of Wolsong unit 1 is 245 cm long and arranged in a complex structure with the coolant. The explicit side structural component was modeled with implicit twelve cylindrical, homogenized regions, as shown in Fig. 1. In the implicit model, material composition of each compartment was averaged over each region with the dimension of 28.575 cm (width, lattice size of channel) × 28.575 cm(width, lattice size of channel) × 20 cm(length). Only two-fuel bundles were described in the model. Neutron leakage was considered at the right-end surface of the Fig. 1. And, white boundary condition was applied for four surfaces surrounding the channel.

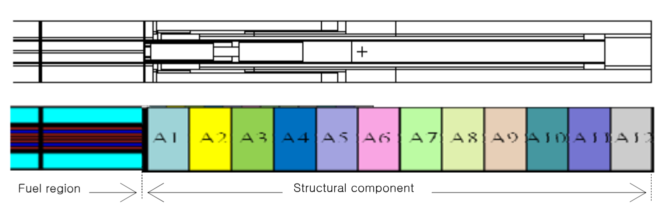

Figure 1: Model for side-structural components in single channel problem.

Calculations and Results

A single channel problem was set up like Fig. 1, and simulated using MCNP, to calculate reference flux profile and capture-cross sections for the forty nuclides. These nuclides were chosen by considering initial composition of structural materials and their activation products, and decay products of the activation products by considering their half-lives, absolute value of capture cross-section, decay and production modes, and abundances from nuclear data base at KAERI [5].

To obtain absolute neutron flux along the channel of the side structural components, the flux decreasing ratio to the innermost region of each channel was calculated, and then the flux profile was scaled up reflecting a neutron flux value of the inner-most region from a full-core calculation of MCNP.

The important characteristics [6] such as specific activities, decay heat and radioactive ingestion hazard were computed twice using ORIGEN2, one with built-in cross- section library and second with the updated cross section library. The results for the specific activity, decay heat, and radioactive ingestion hazard calculated at different times, were in general overestimated when computed using built-in cross section library and the magnitude of errors became larger with the increase in the distance from the core. Table 1 summarizes source term results computed with built-in and updated cross-section library for cells A3, A7 and A11 of the model. The errors of 45% - 97% were found in both the activities of ^{55}Fe and ^{63}Ni reflecting significant effect of cross-section variation. However, the results for ^{60}Co were quite close in both cases reflecting no significant change in the cross-sections of nuclides contributing to its buildup. The errors of ranges 30-50%, 60-90% and more than 90% were observed in cells A3, A7, and A11 respectively. The errors in the calculation of radioactive ingestion hazard were found to be in ranges 45-50%, 88-93%, and 92-96 % for cells A3, A7 and A11, respectively.

Table 1: Comparison of source term results with built-in and updated cross-sections

Cell ID	Decay heat (Watts)		Ingestion Hazard (m^3 of water)		Total Activity (Bq/g)	
	built-in XS lib.	updated XS lib.	built-in XS lib.	updated XS lib.	built-in XS lib.	updated XS lib.
A3	7.51E-04	4.06E-04	5.70E+02	3.07E+02	3.94E+06	2.08E+06
A7	2.13E-07	2.26E-08	1.68E-01	1.73E-02	1.63E+03	1.23E+02
A11	4.58E-08	3.65E-09	3.65E-02	2.83E-03	3.63E+02	2.00E+01

Finally, the source term characteristics of the side structural components at the central channel were determined using the updated cross-section library to deduce absolute values of decay heat, radioactivity and hazard index. Decay heat values of 1.54×10^{-9} to 0.935 Watts were found in side structural components at the time of reactor shutdown. The maximum value was found to decrease to 10^{-4} Watts in the second year following shutdown. The maximum and minimum ingestion hazard indexes were found to be 7.76×10^5 and 1.22×10^{-3} m^3 of water at the time of reactor shutdown. The representative specific activities at different time steps are given in Table 2.

Table 2: The specific activity variation in each cell as a function of time with updated cross-section library

Cell ID	Total specific activity (Bq/g)				
	Time years after shut down				
	0	3	5	10	15
A1	4.61E+09	6.07E+08	3.71E+08	1.24E+08	5.78E+07
A2	6.98E+07	1.12E+07	6.72E+06	1.99E+06	7.27E+05
A3	2.08E+06	3.60E+05	2.14E+05	6.05E+04	1.96E+04
A7	1.23E+02	1.81E+01	1.06E+01	2.86E+00	7.99E-01
A8	7.04E+01	1.04E+01	6.06E+00	1.63E+00	4.58E-01
A11	2.00E+01	3.01E+00	1.74E+00	4.63E-01	1.25E-01
A12	8.85E+00	1.39E+00	8.03E-01	2.13E-01	5.70E-02

CONCLUSIONS

The influence of cross-sections on source term estimation of side structural components in CANDU reactor was analyzed. The specific activity, decay heat, and radioactive hazard index estimated by using the updated cross-sections and were also compared with the results computed with built-in cross sections. It was found that the source terms were overestimated up to 95% for major nuclides defining the waste class when computed using built-in cross section library, and the magnitude of difference became larger with the increase of distance from the core.

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