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Verification of the radiation source terms generated by BESNA through shielding calculation problem

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1. Introduction

The accurate prediction of the nuclide compositions through depletion calculation is important in many nuclear engineering problems such as the source term estimation for criticality and shielding calculations. Typically, shielding calculations with spent nuclear fuel cask are performed using the radiation source terms generated from depletion calculation codes, such as SCALE/ORIGEN [1] for fuel assembly model.

Recently, the depletion code BESNA [2] has been developed by Hanyang University through the supports from NRF and KINS with the capabilities to estimate the nuclide compositions during burnup as well as to evaluate the radiation source terms, such as neutron and photon spectra. In this work, the radiation source term generated by BESNA is verified through comparisons with SCALE/ORIGEN for a shielding calculation problem with the KN-12 spent fuel cask model [3]. The rest of this paper shortly introduces the depletion calculation method with the modified predictor-corrector depletion scheme implemented in BESNA as well as describes a shielding problem and comparison results to verify the radiation source terms generated by the BESNA code.

2. Calculation Methods

2.1 Depletion calculation module in BESNA

This section presents the method for solving depletion equation implemented in the BESNA code. For a mixture of nuclides, the system of Bateman depletion equations is written in matrix-vector form where the diagonal terms of the depletion matrix represent the loss rate of nuclide by radioactive decays and neutron induced reactions, while the off-diagonal terms represent the production rate of a nuclide contributed by decays and reactions from other nuclides.

$$\frac{d\vec{N}(t)}{dt} = (A^d + A^r\phi)\vec{N}(t), \qquad (1)$$

where $\vec{N}(t)$ is the nuclide concentration vector, A^d and A^r are the decay and reaction matrices, respectively.

In practical problem, the nuclide effective one-group (1G) cross sections and neutron flux are continuously changed due to the change of mixture compositions. For that reason, BESNA employs predictor-corrector scheme in depletion calculations, where the predictor calculation performed by applying CRAM to the formal solution of Eq. (1) as follows:

$$\vec{N}_{EOS}^{pred} = e^{t(A^d + A_{BOS}^r \phi_{BOS})}\vec{N}_0. \qquad (2)$$

In Eq. (2), the subscripts BOS and EOS are used to represent the beginning-of-step and the end-of-step, respectively.

On the other hand, the corrector calculation in BESNA is performed by solving:

$$\vec{N}_t = e^{t(A_{MOS}^r \phi_{MOS} - A_{BOS}^r \phi_{BOS})}\vec{N}_t^{pred}, \qquad (3)$$

where the subscripts MOS is used to represent the middle of a depletion step. Then, Eq. (3) is solved using TEM (Talyor Expansion Method) rather than CRAM (Chebyshev Rational Approximation Method). This approach reduces computing time in corrector step without meaningful loss of accuracy.

In the BESNA code, the effective cross section used in each depletion step are obtained by linear interpolating from two nearest neighboring burnups available in the burnup dependent cross section library, which is generated by MCNP6 [4].

2.2. Radiation source term generation in BESNA

The neutron emission spectra calculation module in BESNA considers the contributions from (α,n) reactions and spontaneous fissions, where the method and data are adopted from SOURCES 4B code [5]. The neutron spectra contributed by spontaneous fissions are obtained by using the Watt fission spectrum, while the source spectra contributed by (α,n) reactions can be calculated with an assumption that neutron emitted from (α, n) reactions are isotropically distributed in the center-of-mass (COM) system. On the other hand, the method and data used in photon emission spectra calculation in BESNA are adopted from those in SCALE code system. The photon data includes the line-energy and intensity data for the X-rays, gamma from decay, spontaneous fission gamma, (α, n) reaction gamma, and bremsstrahlung from beta and position particles slowing down in the UO_2 mixture or in water. These data are provided in discrete line data for the source from gamma decay and in continuum data for the remaining sources. In the calculation with the photon source from decay, since the data are given in the discrete form, the photon spectra calculation module in BESNA adjusts the data provided in the library into the user-defined energy grid. For the continuum photon data, the calculation for each nuclide is started by creating a union energy grid that covers the user specify energy grid and the energies specified in the data library for that nuclide. After that, the emission rates are weighted for each group in the union energy grid and then re-assigned into the user energy grid.

3. Verification Problem and Results

3.1 Problem description

In this verification problem, the effects of the radiation source terms estimated by BESNA on the radiation shielding are analyzed for the KN-12 spent fuel cask. The KN-12 spent fuel cask was designed to transport 12 PWR spent fuel assemblies with the type of 14x14, 15x15, or 17x17, under wet or dry conditions, with maximum allowable initial enrichment of 5.0 wt%. The cask overview and the cross-sectional view at the axial mid-plane of the cask are given in **Figures 1a** and **1b**, respectively. The KN-12 spent fuel cask has a height of 480.9 cm and a cask wall thickness of 37.5 cm. The cylindrical cavity has a diameter and height of 119.2 cm and 419 cm, respectively, and contains twelve fuel baskets where each basket has an open dimension of 22 cm x 22 cm. The spent fuel cask is covered by two impact limiters at the top and the bottom of the cask, with a total thickness of 70 cm and a diameter of 245 cm. The fuel assemblies loaded in the cask are assumed to be identical, and the FA regions with UO_2 fuel rod and cladding are homogenized, which is the typical approximation in shielding analysis. The homogenized FA regions are also assumed not to contain fission products for neutron and gamma transport calculations. The dose rate calculations in this problem were performed by the MAVRIC computing sequence in SCALE 6.2 using the neutron and photon sources from depletion calculation for seven cases with various burnups and cooling times, as presented in **Table 1**.

The dose rate calculations were performed with four tally regions. The tally regions have the size of 20 cm x 20 cm x 20 cm each and they are located at the axial mid-plane on the outer surfaces of the cask, which are denoted by the grey color in **Figures 1a** and **1b**. For simplicity, the radiation sources are assumed to be identical over 12 FAs in the cask, and the axial distributions are also assumed to be flat. The shielding calculations are also performed with the neutron and photon source spectra calculated by SCALE/ORIGEN for comparisons.

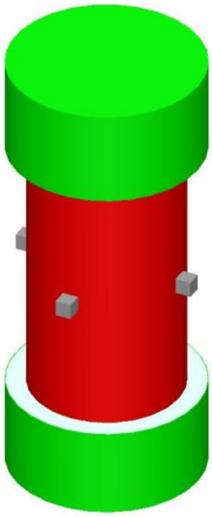

Figure 1a. Overview of KN-12 spent fuel cask

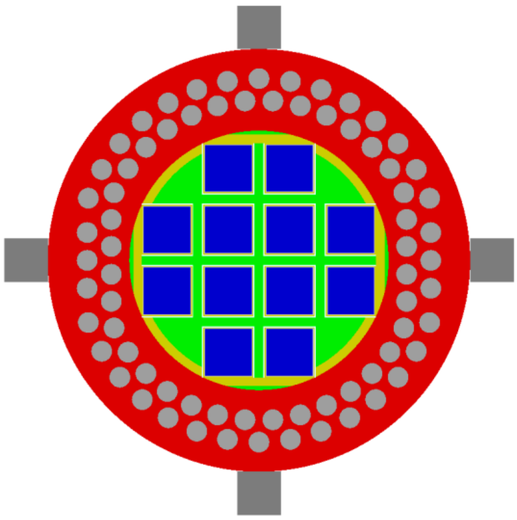

Figure 1b. Cross-sectional view at axial mid-plane of KN-12 spent fuel cask

Table 1. Radiation source term verification data

Case number	Discharged burnup (GWd/tHM)	Cooling time (years)	Neutron source (1/s)		Photon source (1/s)	
			BESNA	ORIGEN	BESNA	ORIGEN
1	20	10	2.13E+04	2.13E+04	5.94E+12	5.97E+12
2	30	10	1.39E+05	1.37E+05	8.82E+12	8.87E+12
3	40	10	4.07E+05	4.01E+05	1.11E+13	1.12E+13
4	50	10	1.06E+06	1.03E+06	1.39E+13	1.39E+13
5	55	10	1.56E+06	1.51E+06	1.52E+13	1.52E+13
6	55	20	1.07E+06	1.04E+06	1.05E+13	1.06E+13
7	55	50	3.59E+05	3.51E+05	5.17E+12	5.20E+12

3.2 Verification results

The calculations in this problem are performed with 1000 cycles of 1,000,000 particles per cycle for photon dose rate calculations and performed with 500 cycles of 300,000 particles per cycle for neutron dose rate calculations. The calculation conditions lead to the statistical uncertainties of about 1% in tallied dose rates. **Figures 2** and **3** present the relative discrepancies in neutron and photon dose rates, respectively, of the cases using source terms estimated in BESNA in comparisons with those using source terms estimated by SCALE/ORIGEN. The relative discrepancies in neutron dose rates in **Figure 2** show the same trends and same levels as the relative discrepancy in neutron emission rate presented in **Table 1**. The first two cases with low burnup have relative discrepancies of less than 1%, which are within the statistical uncertainties of SCALE/MAVRIC. Those discrepancies then increase as the discharged burnup increase, up to a value of less than 4% for case 5, which has the discharged burnup of 55 GWd/tHM and a cooling time of 10 years. As the cooling time increases, the neutron dose rate discrepancies are slowly decreased, with a value of around 2.5% at the cooling time of 50 years. On the other hand, the photon dose rate relative discrepancies are less than 1% for all cases, except for the region tally 2 of the Case 6, which has the value of 1.2%. From these results, it can be concluded that the estimation of dose rates using the radiation emission spectra calculated by BESNA are comparable with those of the source terms from SCALE/ORIGEN, with the discrepancies of about 1% for photon and 4% for neutron dose rate with high discharged burnup and low cooling time. From that results, it can be concluded that the dose rates estimated by using source terms calculated by BESNA are comparable with the cases with the source terms from well-known depletion code SCALE/ORIGEN.

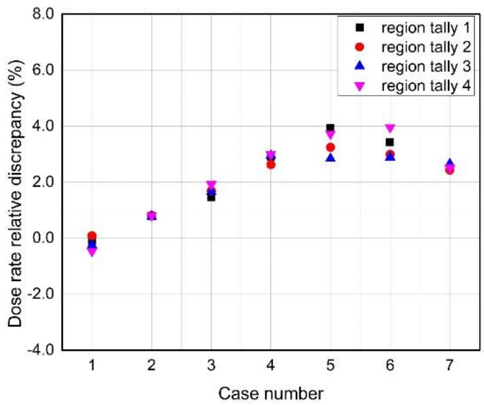

Figure 2. Neutron dose rate relative discrepancy

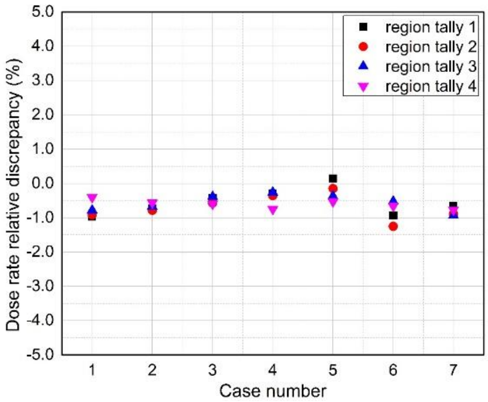

Figure 3. Photon dose rate relative discrepancy

4. Conclusions

In this paper, the radiation source terms estimated by the BESNA code were verified with those calculated by SCALE/ORIGEN through a shielding problem with the KN-12 spent fuel cask. Seven cases with various burnup ranged from 20 GWd/tHM to 55 GWd/tHM and cooling times from 10 to 50 years were considered in the verification problem. The results of the estimated dose rates show that the relative discrepancies between the cases using source term from BESNA and the cases using source term calculated by SCALE/ORIGEN are about 1%, which are about within the statistical uncertainties. On the other hand, the neutron dose rate relative discrepancies are varied with the discharged burnup and cooling time, where the maximum discrepancy of 4% is found at the high burnup and low cooling time. From these verification results, it can be concluded that the radiation dose rates calculated by using source terms estimated by BESNA are comparable with those using source terms from SCALE/ORIGEN. For that reason, it can be concluded that the radiation emission spectra calculated by BESNA are reliable in practical shielding calculations.

Acknowledgments

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