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Evaluation of Bias for SCALE 6.1 MAVRIC Sequence Calculation Using Measurement Data of Neutron Flux on Ex-Core Detector

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1. Introduction

For the preliminary study of activation calculation in reactor concrete primary shield, the systematic uncertainty (bias) of transport calculation using MAVRIC sequence in SCALE 6.1 code package [1] is evaluated by comparisons with measurement data. The measured data are sub-channel powers detected from safety channel ex-core detector obtained during Power Ascension Test of initial core of KSNP reactors. MAVRIC sequence is a coupled radiation transport code intended to deal with problems that are too challenging for standard, unbiased Monte Carlo methods with an aid of deterministic discrete ordinates method. The calculation results from MAVRIC sequence are compared to the measured neutron fluxes obtained from operating KSNP reactors.

2. Methods and Results

2.1 MAVRIC Transport Calculation

Monte Carlo particle transport calculations for deep penetration problems can require very long run times in order to achieve an acceptable level of statistical uncertainty in the final answers. Monte Carlo calculations can be modified to produce results with the same variance in less time if an approximate answer is already known about the problem. Discrete ordinates can be used to quickly compute the approximate answer as a form of adjoint flux. Monte Carlo and discrete ordinates can be used together to find solutions to thick shielding problems in reasonable times. Based on this idea, the MAVRIC (Monaco with Automated Variance Reduction using Importance Calculations) sequence has been developed by ORNL. MAVRIC automatically performs a quick three-dimensional, discrete ordinates calculation using *Denovo* to find the adjoint flux as a function of position and energy. This adjoint flux information is then used to construct an importance map (i.e., target weights for weight windows) and a biased source distribution that work together. The multi-group shielding code *Monaco* then uses the importance map for biasing during particle transport and the biased source distribution as its source.

Figure 1 shows KSNP (Korea Standard Nuclear Power Plant) reactor model for transport calculation. Cycle average pin power data and core average axial power distribution at BOC (Beginning of Cycle) of Cycle 1 [2] are used as spatial source distributions. For source energy distribution, Watt fission spectrum from U^{235} thermal fission, a built-in distribution function provided by *Monaco* module, is utilized. For the generation of importance map and biased source distribution in MAVRIC sequence, a three-dimensional, rectangular grid that covers from source to detector regions must be

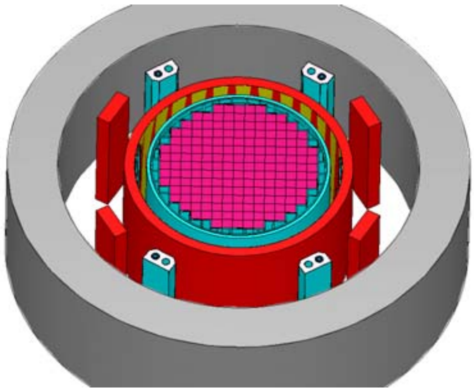

Fig. 1. Cutaway view of reactor model for transport calculation

defined. In this study, upper half of the reactor model from core center to primary shield is chosen as calculation bounds of importance map. Figure 2 shows calculated mesh importance map to be used for optimizing Monte Carlo transport calculation.

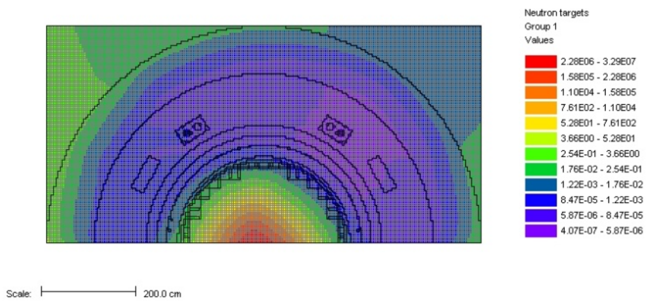

Fig. 2. Mesh importance map at reactor mid-plane for neutron group 1

2.2 Measured Flux Data

The plant-measured data from the safety channel ex-core detector are obtained and converted to thermal neutron flux for the comparison with the calculated neutron flux. The ex-core detectors are located in reactor cavity to monitor neutron flux levels by detecting neutrons leaked from the reactor vessel for normal and accident conditions. The four safety channel ex-core detectors are located 35° off the centerline of reactor vessel outlet nozzle and each channel consists of vertically aligned three fission chambers (sub-channels). The plant data are obtained from "Linear Power Sub-channel Calibration Test" for YGN 4 [3] and UCN 3&4 [4, 5]. Table 1 shows one of 5 sets of the raw data obtained from UCN 3 for sub-channel calibration. Each set of recorded data as shown in Table 1 consists of data for 4 azimuthal safety channels (A, B, C, D), each of which has 3 sub-channels (bottom, middle, top). The sub-channel powers are values read from the ex-core detector signal before current calibration and the plant power represents the primary calorimetric power (real power). The sensitivity of safety channel fission chamber is known as 1.7×10^{-10} mA/nv [6].

Table 1: Measured sub-channel power and as-found current

Sub-channel	Ex-core linear sub-channel power [%]			
	Ch A	Ch B	Ch C	Ch D
bottom	32.250	31.525	32.700	29.700
middle	44.275	39.700	41.950	42.125
top	30.875	32.225	31.975	31.650
Plant power [%]	20.06			
Sub-channel	As-found current for 200% power [mA]			
	Ch A	Ch B	Ch C	Ch D
bottom	0.1972	0.1906	0.1892	0.1901
middle	0.1963	0.1916	0.1890	0.1915
top	0.1963	0.1912	0.1869	0.1910

From these data, the measured fluxes at 20% power on the ex-core detector sub-channel are calculated as follows:

$$\text{sub-channel flux at 20\% power level} = \frac{1}{10}\times\frac{\text{sub-channel power [\%]}}{\text{plant power [\%]}}\times\frac{\text{as-found current [mA]}}{\text{sensitivity [mA/nv]}}$$

The average fluxes on each sub-channel in the safety channel ex-core detector obtained from three KSNP units data are listed in 3rd column of Table 2.

2.3 Comparison of Calculated and Measured Data

The three sub-channels vertically aligned in the safety channel ex-core detector at the 1st quarter of the reactor model are chosen for the tally regions in MAVRIC calculation and the calculated results are compared with the measured neutron fluxes as shown in Table 2. The measured data for each sub-channel detector is the averaged value of a total of 60 sub-channel flux data from three KSNP reactors (20 data for each plant unit).

Table 2: Calculated and Measured Thermal Neutron Flux

Sub-channel	Thermal Neutron Flux		Ratio
	Calculated (1σ)	Measured (1σ)	
bottom	2.42E+09 (1.1%)	1.88E+08 (11.1%)	1.28
middle	3.33E+09 (0.9%)	2.54E+09 (10.4%)	1.31
top	2.25E+09 (1.2%)	1.71E+08 (9.00%)	1.32

Calculated values on the ex-core detector sub-channels using MAVRIC sequence are approximately 1.3 higher compared to the measured data obtained from KSNP units. The differences between calculated results and measured data are attributed to modeling strategy, input data selection, and uncertainty in detector sensitivity. The calculation model in this study contains the following simplifications: complicated but not critical structures such as brace-rings and flanges in shroud assembly are omitted and axial variation of coolant density is neglected by applying average core coolant density in the core and bypass water region. As for source input data, cycle average pin powers from design data for typical KSNP unit are utilized for radial core neutron source distribution. The utilized detector sensitivity for the conversion of measured data is nominal value and it might have variations due to factors such as integrity of detector fill gas and coated uranium thickness of detector, etc. Other factors such as uncertainties due to the mechanical tolerances, uncertainties in nuclear cross-section data, and bias inherent in Monte Carlo methods can also contribute to the deviation of calculation results from the measured data.

3. Conclusions

As a preliminary study for activation calculation in reactor primary shield, the calculation bias of MAVRIC sequence of SCALE 6.1 code in transport analysis for a typical reactor model especially in ex-vessel region has been determined by the comparisons with measured data from operating KSNP plants. Evaluated systematic uncertainty of +30% can be considered as calculation margin when applying MAVRIC sequence to activation analysis of concrete primary shield for decommission.

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