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Application of SCALE 6.1 MAVRIC Sequence for Activation Calculation in Reactor Primary Shield Concrete

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1. Introduction

For the planning of decommission, activation radioactivity by neutron interactions in various regions of nuclear reactor should be estimated. The production rates of activation in concrete primary shield may be small compared to those in reactor internals due to the flux level. But, considering the concrete volume, the quantity of radioactive waste expected to be produced from decommissioning cannot be ignored. Activation calculation requires flux information at desired location and reaction cross sections for the constituent elements to obtain production rate of activation products. Generally it is not an easy task to obtain fluxes or reaction rates with low uncertainties in a reasonable time for deep penetration problems by using standard Monte Carlo methods. The MAVRIC (Monaco with Automated Variance Reduction using Importance Calculations) sequence in SCALE 6.1 code package [1] is intended to perform radiation transport on problems that are too challenging for standard, unbiased Monte Carlo methods. And the SCALE code system provides plenty of ENDF reaction types enough to consider almost all activation reactions in the nuclear reactor materials.

To evaluate the activation of the important isotopes in primary shield, SCALE 6.1 MAVRIC sequence has been utilized for the KSNP reactor model and the calculated results are compared to the isotopic activity concentration of related standard [2].

2. Methods and Results

2.1 MAVRIC Transport Calculation

MAVRIC is based on the CADIS (Consistent Adjoint Driven Importance Sampling) methodology, which uses an importance map and biased source that are derived to work together. MAVRIC automatically perform a coarse mesh, three-dimensional, discrete ordinates calculation using *Denovo* to determine the adjoint flux as a function of position and energy. This adjoint flux information is then used by MAVRIC to construct a space and energy-dependent importance map (i.e., weight windows) and a mesh-based biased source distribution. MAVRIC then passes the importance map and biased source distribution to the functional module *Monaco* to complete the particle transport problem. Functional module *Monaco* is a three-dimensional, fixed-source, multi-group shielding code using the Monte Carlo method.

Figure 1 shows KSNP reactor model for transport calculation. Pin power data of equilibrium core are used as horizontal source distribution. For axial power distribution and fission source spectrum, special built-in distributions provided by *Monaco* module are utilized.

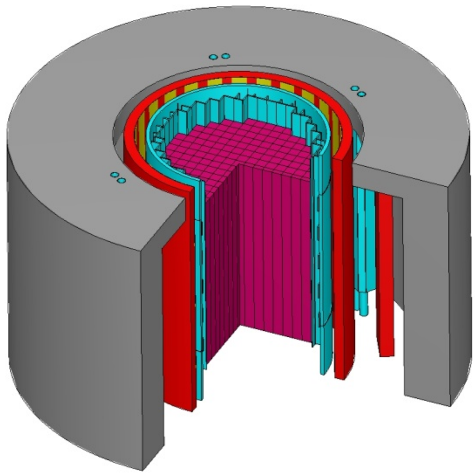

Fig. 1. Cutaway view of reactor model for transport calculation

Around 90° off the centerline of reactor vessel outlet nozzle, where maximum flux occurs at inner surface of the primary shield, and the range of 0 to 100 cm depth into the concrete wall have been chosen for tallying of activation reaction rates. Maximum thermal neutron flux occurs near 13 cm into the concrete.

2.2 Activation Calculation

To setup the activation calculation model for primary shield, it is essential to compose the constituent nuclides of concrete. Normally, concrete composition is hard to identify unlike other well controlled construction materials of nuclear power plant. Composition data obtained from US nuclear power plants by performing comprehensive analysis [3] on samples of concrete are reviewed and the quantities provided for trace elements, important for concrete activation analysis, are determined to be used in this study. Among many trace elements listed in NUREG/CR-3474 [3], noteworthy isotopes are selected based on the half-life and the expected inventory of activation products for the activation calculation of reactor primary shield. The quantities of selected trace elements in this study are shown in Table 1. The major chemical composition of concrete primary shield is taken from ANSI/ANS-6.4-2006 [4].

Table 1: Quantity of trace elements

Element	Quantity
	Average ± 1σ [ppm]
Li	20
Cl	45 ± 18
Co	9.8 ± 10.3
Ni	38 ± 25
Nb	4.3 ± 3.0
Ba	950 ± 1950
Sm	2.0 ± 1.3
Eu	0.55 ± 0.38

Table 2 shows considered elements and activation products with principle production mode in concrete. Among target elements in Table 2, only iron element belongs to the major components and others belong to trace elements.

Table 2: Activation products in Concrete

Target elements	Reaction type	Activation products
Li-6	(n,α)	H-3
Cl-35	(n,γ)	Cl-36
Fe-54	(n,p)	Mn-54
Fe-54	(n,γ),	Fe-55
Co-59	(n,γ)	Co-60
Ni-62	(n,γ)	Ni-63
Ba-132	(n,γ)	Ba-133
Sm-150	(n,γ)	Sm-151
Eu-151	(n,γ)	Eu-152
Eu-152	(n,γ)	Eu-154

The specific activity of activation product is calculated as

$$A(t,T) = \int \Sigma(E)\,\varphi(E)dE \cdot \left[1 - e^{-\lambda t}\right] \cdot e^{-\lambda T}, \quad (1)$$

where Σ, λ, φ, t, and T represent the activation reaction cross-section of target isotope, decay constant of activation product, neutron flux, reactor operation period, and decay time after shutdown respectively. The production rates expressed as integral term in Eq. (1) can be obtained directly from the MAVRIC calculation.

The calculated specific activities at maximum position in concrete primary shield for KSNP reactor after 40 effective full power years of operation are listed in Table 3. They are compared with the minimum specific activity for exemption standard related to the radiation protection [2]. At shutdown, it is evaluated that about half of the considered activation products exceed the minimum activity for exemption. After ten years of decay period, 3 isotopes (Co-60, Eu-152, 154) remain to exceed the exemption standard. Table 4 shows the activity inventories of activation products by considering effective volume of primary shield zone where most activation reaction occurs. They are also compared with the minimum quantity for exemption [2].

Table 3: Specific activities of activation products

Nuclides	Activity concentration [Bq/g]		
	Specific activity for exemption	at shutdown	10 years after shutdown
H-3	1.00E+06	3.52E+05	2.01E+05
Cl-36	1.00E+04	7.79E+00	7.79E+00
Mn-54	1.00E+01	**3.40E+02**	1.03E-01
Fe-55	1.00E+04	**5.45E+04**	4.30E+03
Co-60	1.00E+01	**1.33E+04**	**3.56E+03**
Ni-63	1.00E+05	1.52E+02	1.42E+02
Ba-133	1.00E+02	**1.53E+02**	7.90E+01
Sm-151	1.00E+04	6.17E+01	5.71E+01
Eu-152	1.00E+01	**2.50E+04**	**1.50E+04**
Eu-154	1.00E+01	**1.45E+03**	**6.46E+02**

Table 4: Quantities of activation products

Nuclides	Activity inventory [Bq]		
	Quantity for exemption	at shutdown	10 years after shutdown
H-3	1.00E+09	**1.62E+09**	9.22E+08
Cl-36	1.00E+06	8.05E+04	8.05E+04
Mn-54	1.00E+06	**9.64E+08**	2.92E+05
Fe-55	1.00E+06	**1.54E+11**	**1.22E+10**
Co-60	1.00E+05	**2.98E+07**	**8.02E+06**
Ni-63	1.00E+08	1.33E+06	1.24E+06
Ba-133	1.00E+06	**3.33E+07**	**1.72E+07**
Sm-151	1.00E+08	2.83E+04	2.62E+04
Eu-152	1.00E+06	**3.16E+06**	**1.89E+06**
Eu-154	1.00E+06	1.83E+05	8.16E+04

The isotopes that exceed the quantity for exemption at shutdown are the same as those of activity concentration case with the exception of H-3 and Eu-154 isotopes. After 10 years of decay period, 4 isotopes (Fe-55, Co-60, Ba-133, Eu-152) remain to exceed the exemption standard.

3. Conclusions

Related to the planning for decommission, the activation products in concrete primary shield such as Fe-55, Co-60, Ba-133, Eu-152, and Eu-154 are identified as important elements according to the comparisons with related standard for exemption. In this study, reference data [3, 4] are used for the concrete compositions in the activation calculation to see the applicability of MAVRIC code to the evaluation of activation inventory in the concrete primary shield. The composition data of trace elements as shown in Table 1 are obtained from various US power plant sites and accordingly they have large variations in quantity due to the characteristics of concrete composition. In practical estimation of activation radioactivity for a specific plant related to decommissioning, rigorous chemical analysis of concrete samples of the plant would first have to be performed to get exact information for compositions of concrete.

Considering the capability of solving deep penetration transport problems and richness of reaction cross-section data in SCALE code system, the MAVRIC sequence in SCALE 6.1 can be used as an effective tool for the evaluation of activation radioactivity especially when it is related to the deep penetration such as activation problem in the reactor primary shield.

REFERENCES

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