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Numerical modeling of APR1400 reactor using Monte Carlo Continuous Energy Burnup Code

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1. Introduction

In this article, we present the development of an advanced numerical model of the Korean APR1400 nuclear reactor for burnup calculations using the Monte Carlo Continuous Energy Burnup Code (MCB)[1]. The parameters that have been obtained in Monte Carlo simulations of radiation transport coupled with burnup calculations using the linear chain method are the concentrations of selected actinides and fission products for the first reactor cycle. The numerical model was developed mainly with the use of the rector's design data presented in publicly available documents.

2. The MCB code

The MCB code is a numerical tool for the modeling of radiation transport as well as isotopic changes in matter in any three-dimensional geometry of the nuclear system. The code allows the calculation of many parameters of reactor physics, including effective neutron multiplication factor, reaction rates, distribution of neutron flux and power as well as nuclide concentrations for any arbitrary time steps during the nuclear system's operation. The code is capable of modeling the operation of the nuclear system with varying power over the time steps and also in the pure radioactive decay mode. The user receives a set of parameters for each defined time step. In addition, the code allows modeling of the geometric changes in the nuclear system, for example, to determine the influence of control rods movement on the reactor's parameters.

Technically speaking, the MCB code is a combination of the Monte Carlo N-Particle Transport Code (MCNP) code and the Transmutation Trajectory Analysis (TTA) code at the FORTRAN source code level. The MCNP code is used for the radiation transport calculations and the TTA code for the calculations of isotopic changes in matter. The MCB can use any nuclear data libraries in the Evaluated Nuclear Data File (ENDF) format. The code also has functionalities that allow its execution in parallel mode with the help of the Message Passing Interface (MPI).

The code was used to model many IIIth and IVth generation nuclear systems, e.g. Pressurized Water Reactors (PWR), Height Temperature Gas-cooled Reactors (HTGR), Lead-cooled Fast Reactors (LFR), and also Accelerator-Driven subcritical Systems (ADS). In addition, the comprehensive validation of the code using spent fuel assay data from PWR reactor was performed[2].

3. Numerical model

The APR1400 numerical model was developed at the level of the reactor pressurized vessel. It contains the core as well as most of the elements constituting the top, bottom, and side reflector, in accordance with the available technical specifications. Above the core, there is a grid of control rods, which gives the possibility of modeling the control rod movements during reactor operation. The rector's core has been divided into 11 axial and 22 radial zones in order to accurately reproduce the distribution of individual nuclides in the reactor core during burnup. The burnup zones were established according to the fuel assemblies specifications for the first reactor cycle – assemblies A0 to C3 [3]. Differently enriched fuel rods and burnable poison rods are separate zones within each type of assemblies. The calculations were performed for the variable content of boric acid in the cooling water for twenty time steps representing burnup from 0 to 17.581 GWd/tHM$_{int}$.

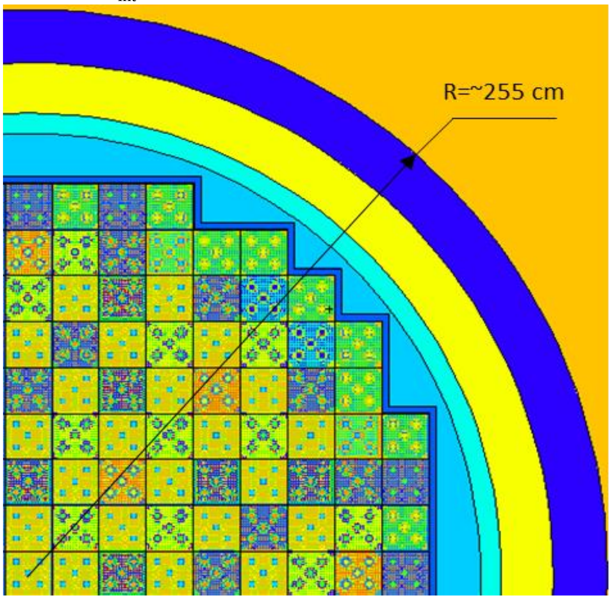

Fig. 1. Visualization of the 1/4 initial APR1400 reactor core model for numerical simulations.

4. Results

Table 1 shows the final masses of fission products for the entire reactor core at a burnup of 17.581 GWd/tHM$_{int}$. The isotopes were selected taking into account the Nuclear Energy Agency (NEA) recommendation of key isotopes for the four research areas, i.e. waste management, burnup credit, radiological safety, and burnup indication [4]. As shown in Table 1, the highest masses were obtained for ^{139}La and ^{137}Cs and the smallest for ^{151}Eu and ^{155}Gd.

Table I: Fission product masses at the end of the initial reactor cycle.

No	Isotope	Mass [g]	Importance
1	^{79}Se	3.14E+02	W
2	^{95}Mo	2.88E+04	B
3	^{90}Sr	3.64E+04	R,W
4	^{99}Tc	4.72E+04	B
5	^{101}Ru	4.65E+04	B
6	^{106}Ru	7.83E+03	R,W
7	^{103}Rh	2.31E+04	B
8	^{109}Ag	1.70E+03	B
9	^{125}Sb	7.75E+02	R
10	^{129}I	1.69E+04	W
11	^{133}Cs	5.88E+04	B
12	^{134}Cs	4.19E+03	R
13	^{135}Cs	1.52E+04	W
14	^{137}Cs	6.38E+04	R, W
15	^{139}La	6.72E+04	I
16	^{143}Nd	5.21E+04	B
17	^{145}Nd	4.69E+04	B
18	^{148}Nd	2.24E+04	I
19	^{144}Ce	3.52E+04	R
20	^{147}Pm	1.74E+04	B
21	^{147}Sm	2.93E+03	B
22	^{149}Sm	2.06E+02	B
23	^{150}Sm	1.75E+04	B
24	^{151}Sm	8.37E+02	B
25	^{152}Sm	6.19E+03	B
26	^{151}Eu	8.24E-01	B
27	^{153}Eu	4.51E+03	B
28	^{154}Eu	6.79E+02	R
29	^{155}Eu	2.54E+02	B
30	^{155}Gd	2.12E+01	B
W-Waste Management, B-Burnup Credit, R-Radiological Safety, I-Burnup Indicator			

Figure 2 shows the time evolution of the key actinides, also selected in line with the NEA recommendation. Uranium contains ^{234}U, ^{235}U and ^{236}U, neptunium contains ^{237}Np, plutonium contains ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, americium contains ^{241}Am, ^{242m}Am, ^{243}Am, and curium contains ^{242}Cm, ^{243}Cm, ^{244}Cm, ^{245}Cm, ^{246}Cm, ^{247}Cm. The depletion of ^{238}U equals about 1.47E+7 grams from the initial 1.01E+08 grams to the final 9.93E+07 grams. The time evolutions of the presented elements are characteristic of the PWR-type reactor.

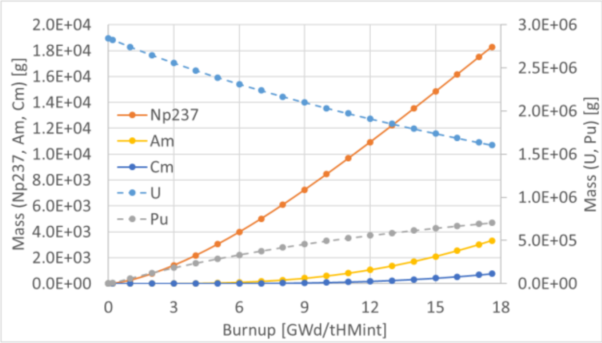

Fig. 2. Time evolutions of actinides during the initial reactor cycle.

5. Conclusions

As part of the research work, a new numerical model of the APR1400 reactor was developed at the reactor pressurized vessel level. This is a new approach to the modeling of PWR-type nuclear systems as they are usually modeled only at the level of the reactor core surrounded by reflector layers. The obtained results are consistent with the data presented by other research groups. It is worth noting, however, that the results of numerical simulations largely depend on the parameters of the adopted numerical model, such as model geometry, the division into burnup zones, calculation method, nuclear data libraries, number of time steps, modeling of reactivity control systems etc. Unfortunately, the data presented in the open sources are limited, which also limits the scope of the comparative analysis and validation of the obtained results, mainly considering final nuclide isotopic concentrations.

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REFERENCES

[1] M. Oettingen, J. Cetnar, T. Mirowski, The MCB code for numerical modelling of fourth generation nuclear reactors, Computer Science, 16, 329–350, 2015.

[2] M. Oettingen, Validation of fuel burnup modelling with MCB Monte Carlo system using destructive assay data from Ohi-2 PWR, Kraków, Wydawnictwa AGH, 174, 2016.

[3] M. Salam, Ch. J. Hah, Comparative study on nuclear characteristics of APR1400 between 100% MOX core and UO_2 core, Annals of Nuclear Energy, 119, 374-381, 2018.
[4] Nuclear Energy Agency, Organization for Economic Co-operation and Development, Spent Nuclear Fuel Assay Data for Isotopic Validation, State-of-the-art Report, Nuclear Science, NEA/NSC/WPNCS/DOC(2011)5, June 2011.