RADIONUCLIDE INVENTORY CALCULATION IN VVER AND BWR REACTOR

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Received 09 May 2014; accepted 25 May 2014

1. Introduction

Increasing number of reactors reaching their designed lifetime poses challenges on decommissioning scenario planning. One of the tasks preceding the radioactive waste treatment is determination of its radionuclide inventory. Experimental determination of radionuclide inventory is difficult and expensive. Therefore, it is convenient to perform proper calculations by validated computer codes. Realistic estimate of the radionuclide inventory of a reactor is supposed to lead to cost reduction of the decommissioning process as well as to increased radiation safety of the workers. In this work, MCNPX calculation will be performed for a Slovak VVER-440/V-230 and a German BWR construction line 69 reactor, which represent a significant share in the world reactor fleet.

2. Calculation code

In the calculation, the MCNPX (Monte Carlo N-Particle eXtended) version 2.7.0 was used. MCNPX is a general-purpose Monte Carlo radiation transport code for modelling the interaction of particles (neutrons) with matter. Its greatest advantage is the ability to model three-dimensionally the core and the reactor structures. Further it uses continuous neutron energy spectra in the neutron transport calculation [1]. For reaction calculations, established evaluated data files (continuous energy neutron cross-section libraries ENDF/B–VII.0) are used. This code enables calculations of neutron flux per surface or volume element in criticality or fix source mode, but also neutron activation calculations utilizing the burn-up/activation feature accessed via BURN card (in case of the criticality mode).

3. BWR calculation

The core is created by rectangular fuel elements. Several types were developed during the reactor operation (7*7, 8*8 fuel pins, different ^{235}U enrichment, half-length etc.). The fuel elements are placed inside the reactor shroud with thickness of 40 mm. The water steam mixture used in the primary loop to cool the core must be very pure in order to avoid any activation. The reactor pressure vessel (RPV) has an 8 mm weld pad on the inner surface, the thickness of the RPV itself is 138 mm.

Geometrical model was created based on the technical documentation from a German boiling water reactor of the construction line 69. The model includes all the structures inside the RPV like steam drier, reactor shroud and outside the RPV like thermal shield, condensation chamber and building structure until the containment border. For the neutron transport and activation calculations inside the reactor containment, the MCNPX model shown in Fig. 1 was developed.

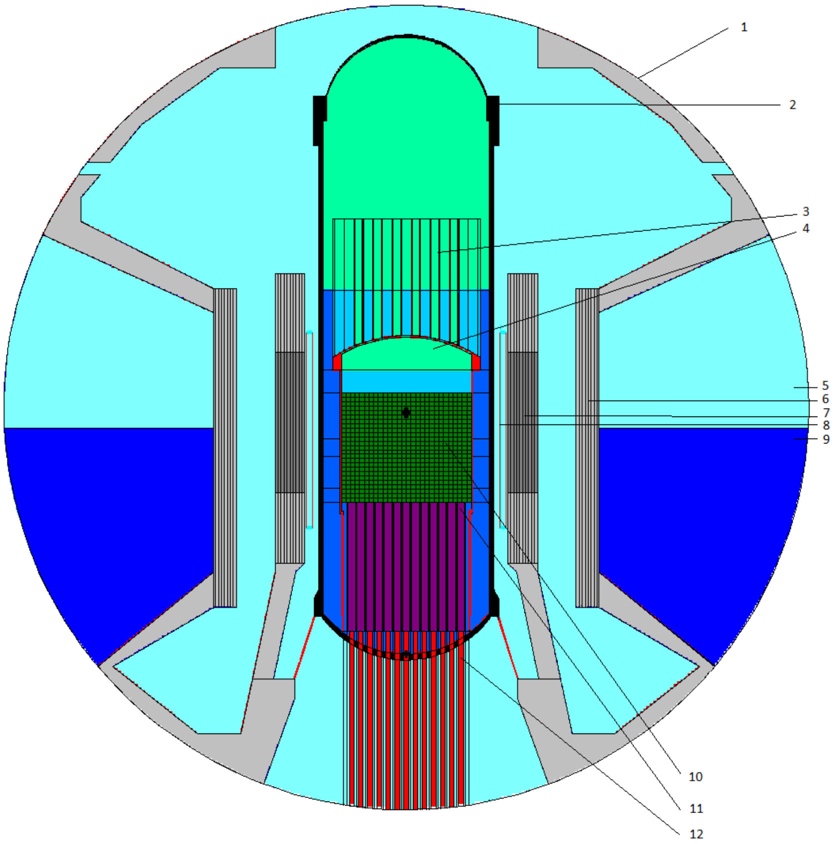

Fig. 1: *Vertical cross section of the BWR model: 1 – containment, 2 – RPV, 3 – steam separator, 4 – steam dome, 5 – condensation chamber, 6 - thermal shielding, 7 – radiation shielding, 8 – thermal isolation, 9 – water in the condensation chamber, 10 – core, 11 – control assemblies, 12 – control rods propulsion system.*

In order to consider the axial change of coolant temperature (density) and creation of steam bubbles in the core, 25 fuel pins lattices with different coolant density and 25 FAs lattices also with different coolant density were designed. Each FAs lattice is then filled into another axial level of the core lattice. In the end, the core lattice has 25 axial fuel levels plus one coolant level above and below (total of 27 levels) in order to avoid miniature geometry errors (undefined spaces). The fuel model also consists of 25 axial layers to take density and steam effects into account. The overall model consists of 28 Lattices, 121 Universes, 361 Cells and internal cells. The majority of the cells define the core itself, cells outside the RPV create only less than one third of the total cells amount. Neutron source is inserted as pin-by-pin data coming from deterministic calculations. The pin-wise neutron source definition is more demanding than the criticality source definition. However, this is well compensated in form of the achieved calculation time savings.

The radiation shielding (heavy concrete) structure was divided into 10 sections with 10 cm step in diameter. It is due to the change of neutron flux which is in 4 orders of magnitude (see Tab. 1).

Tab.1. *Course of the neutron flux in the radiation shielding layers[neutron/cm^2].*

NPS	1st	2nd	3rd	4th	5th	6th	7th	8th	9th	10th
1.00E+09	1.34E-11	6.24E-12	2.36E-12	7.59E-13	2.26E-13	7.75E-14	2.73E-14	8.46E-15	2.79E-15	2.53E-15
1.00E+08	1.29E-11	6.38E-12	2.55E-12	7.52E-13	2.35E-13	8.32E-14	2.82E-14	2.26E-15	1.25E-15	7.54E-16
1.00E+07	1.41E-11	5.69E-12	3.33E-12	6.45E-13	1.91E-13	3.63E-14	1.40E-14	1.19E-14	1.56E-15	-

**NPS number of source neutrons to be started*

The neutron flux calculation shows a significant decrease of neutron flux in 4 orders of magnitude in the 1 meter thick heavy concrete layer. Uncertainties in the most distant layers are more than 50%. In order to reach rational uncertainties (under 10%), more starting neutrons have to be applied. However, this poses high demand on calculation resources. On a cluster of 12 processors, a 1.00E+09 starting neutrons calculation takes around 30 hours computational time. For this reason, implementation of variance reduction methods is required.

4. VVER calculation

The core is made of hexagonal fuel elements with different ^{235}U enrichment. It is contained in the core basket (35 mm thick) inside the reactor shroud (60 mm). The core is cooled and moderated by water. The coolant includes boric acid added in order to regulate the reactivity reserve in the reactor and hence the chain reaction. The RPV is designed to sustain higher pressure than in the BWR which reflects to the weld pad thickness of 9 mm and wall thickness of 140 mm.

Material data was determined for the mean coolant temperature 280°C. The chemical composition of the fuel was taken from the calculations for the 3.82% 235U-enriched fuel assemblies. Mean position of the 6th control assembly group was based on the technical documentation from a Slovak NPP determined at 175 cm. Last part of preparation of the input file for the criticality calculation was determination of the boric acid critical concentration. The task was calculated by the MCNP5 code.Based on the technical documentation, it was possible to design a VVER-440/V-230 reactor model, which can be used for further calculations (see Fig. 2).

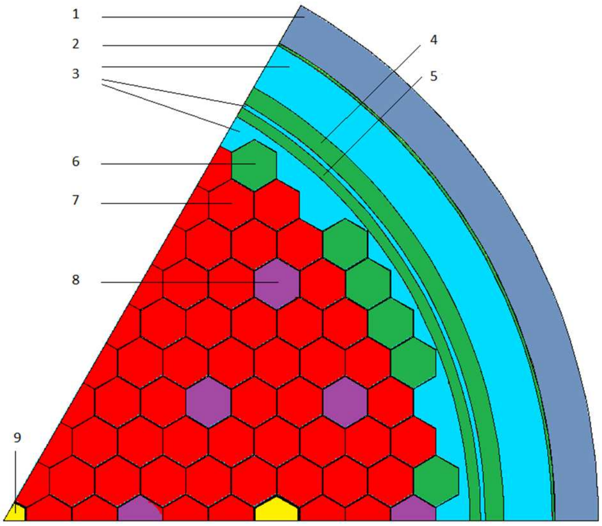

Fig. 2: *Horizontal cross-section of the VVER core model: 1 – RPV, 2 – austenitic and reinforcing RPV weld pad, 3 – coolant, 4 – reactor shroud, 5 – core basket, 6 – dummy element, 7 – fuel assembly, 8 – control assembly, 9 – 6th control assembly group.*

The calculation of radionuclide inventorywas performed for 15 campaigns after installation of the dummy elements into the active zone. The mean burn-up value per campaign for the fuel was considered 16 MWd/kg$_U$. In Tab. 2.you can see the total activities of selected (based on half/life) radionuclides, which were produced in the dummy elements during the reactor campaigns in VVER reactor (after reducing the zone and change of the loading pattern). The uncertainty of the MCNPX calculation is estimated not to exceed 5%. These results are calculated for the time directly at the outage of the reactor.

Tab. 2.*Total activity of selected radionuclides in one dummy element.*

Radionuclide	$T_{1/2}$	Radiation type	A_i [Bq]
^{55}Fe	2.73 y	EC (X)	4.69E+15
^{60}Co	5.2714 y	β- (γ)	8.41E+14
^{54}Mn	312.3 d	EC (γ)	3.57E+14
^{63}Ni	100.1 y	β-	1.95E+14
^{57}Co	271.79 d	EC	3.92E+12
^{59}Ni	7.6E04 y	EC (X)	1.18E+12
^{93}Mo	4.0E03 y	EC (X)	2.54E+10
^{94}Nb	2.03E04 y	β- (γ)	6.15E+06
Total activity of selected radionuclides:			**<u>6.09E+15</u>**

**EC – electron capture, X – X-ray emission*

Fig. 3 represents the course of the summary activity of selected radionuclides (see Tab. 2.) in all 36 dummy elements after installation into the NPP's active zone before the 13th campaign.

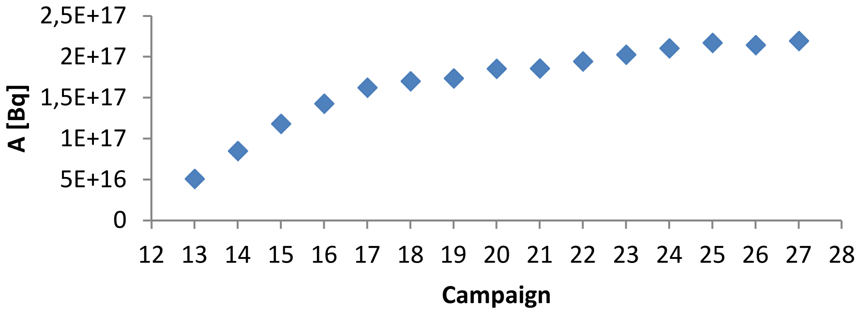

Fig. 3:*Course of the summary activity of selected radionuclides in dummy elements during first 15 campaigns after installation.*

After irradiation in the active zone during the reactor operation, the radioactive decay will take place. The activity of the short-lived radionuclides (e.g. ^{58}Co $T_{1/2}$=70.86 days, ^{59}Fe $T_{1/2}$=44.503 days) falls sharp, which is why they were not taken into account. The long-lived radionuclides will dominate the total activity. According to the results, ^{55}Fe (half-life 2.73 years) will be the most important activation product in the first 14 years after reactor operation. However, with considering of longer irradiation time (30+ years), this period should fall towards 10 years. In the next years ^{60}Co and ^{63}Ni will be mostly responsible for the total activity of the dummy elements.

5. Conclusions

The paper shows different aspects in the radionuclide inventory determination. Precise determination of the neutron flux distribution, presented for a BRW reactor, is vital for the activation calculations. The precision can be improved utilizing variance reduction methods as importance treatment, weight windows etc. Direct calculation of the radionuclide inventory via Monte Carlo code is presented for a VVER reactor. Burn-up option utilized in this calculation appears to be proper for reactor internal components. However, it will not be probably effective outside the reactor core. Further calculations in this area are required to support the forth-set findings.

Acknowledgement
This work was financially supported by grant of Science and Technology Assistance Agency no. APVV DO7RP-005-12and Scientific Grant Agency of the Ministry of Education of Slovak Republic and theSlovak Academy of Sciences No. VEGA 1/0366/12 and VEGA 1/0204/13.

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