CALCULATION OF INDUCED ACTIVITY IN THE V-230 REACTOR

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1. Introduction

To date, there are 437 operational reactors in the world. Most of them are in the second half of their planned lifetime. One of the most important issues, in order to ensure safe and effective decommissioning of a nuclear power plant, is to collect all the necessary operational data and to create proper plans before the start of the decommissioning process. This includes also determination of the radioactivity inventory of a nuclear power plant.

2. Radioactive inventory

Three basic categories of radionuclides, originating from different physical processes, are created during the operation of a nuclear power facility:

a) Fission products resulting from the fission of fuel nuclei by neutrons. These are highly active nuclides and their half-life ranges from a few seconds to thousands of years.
b) Transuranium elements, especially plutonium and higher actinides (Am, Cm), resulting from successive captures of neutrons by nuclear fuel and their subsequent transformations, starting mostly with the ^{238}U.
c) Activation products, i.e. radionuclides resulting from the interaction of neutrons with inactive coolant, moderator and structural materials nuclides of the reactor. Most occurring nuclear reactions types are (n, γ), (n, 2n), (n, p) and (n, α). Activation products are then converted to stabile nuclides by beta transformation which is often accompanied by a gamma radiation [1].

Following reactor shutdown and the removal of irradiated fuel, two categories of radionuclides contribute the most to the radioactive inventory: the activation products bound in the reactor structural materials designated as induced activity and radionuclides from materials contamination, which are activated corrosion and erosion products conveyed by coolant, dispersed fuel particles and fission products escaping through cladding breaches and settling on the cover primary circuit components. This means that a radiological characterization of a nuclear facility is a result from contamination and activation of materials. For reactors, where during operation no accident occurred, the principal component of radioactive inventory is, in terms of reactor decommissioning, the activation of its structural materials [2].

Major activation products can be categorized by material into 2 groups:

- in steels: Fe-55, Ni-59, Co-60, Ni-63, Mo-93, Nb-94, Ag-108m,
- in reinforced concrete: Cl-36, Ca-41, Fe-55, Ni-59, Co-60, Ni-63, Eu-152, Eu-154, Eu-155 [2].

In terms of the decommissioning processes, it is necessary to know the accurate inventory of induced activity for the individual reactor structural parts, in order to determine the quantity and type of radioactivity that effects: the decommissioning scenario selection process, the application of certain dismantling techniques, the requisite for decontamination, waste treatment method and of course the possibility of reducing the workers dose load too.

3. Radiological characterization

The essential objective of a characterization program is to obtain representative calculations, in situ measurements and samples/analyses which provide an understanding of the radiological conditions that will be encountered during decommissioning. Characterization will not provide all information desired, but should yield sufficient data to permit logical assumptions to be made. It must be recognized that the more calculations and measurements are made and the more samples taken and analyzed, the more the characterization program will cost [2].

The induced activity, and the radionuclide concentrations present in neutron irradiated components (e.g. reactor pressure vessel, reactor vessel internal components, biological shield, some accumulated operational wastes stored on-site) and the associated γ dose rates are usually estimated by using neutron activation calculations for the components of interest. Remotely deployed detectors are used to measure high radiation parts of the reactor vessel and internals. Remote sampling of the activated materials is generally necessary to support validation of the computer codes used to perform the calculations. However, this process can be expensive and difficult for highly activated components and structures where trace amounts of sample material can produce radiation dose rates in the range of Gy/h [2]. In the case of an operating reactor, such measurements are difficult, if not even infeasible [3]. Due to this fact, validated computer codes and methodologies are important to calculate the induced activity in the reactor and its immediate surroundings.

3.1. Neutron induced activity calculations

The calculation of neutron induced activity requires, as a first step, knowledge of the spatial and energy distributions of the neutron flux throughout the system. The neutron flux is then used to determine the individual reaction rates of the parent radionuclides whose daughters give rise to the ionizing radiations. These reaction rates are then used to obtain the level of activity per unit weight of parent element according to the reactor irradiation history and the subsequent decay time. The final stage is the calculation of the component activity from the 'known' concentration of the parent elements in the material from which the component is manufactured, together with the mass of the components. 'Known' means averaged value obtained in the course of the composition sampling program or inferred from other relevant information (e.g. from the reactor builders) [2].

3.2. Computer codes for neutron flux spatial and energy distribution

For flux calculations in general, the system is divided into several volume elements, called zones, and the average neutron fluxes in each zone are calculated for each energy group. Typical codes used to calculate these spatial and energy distributions of neutrons are:

- 1-D: ANISN [4], [5], [6], XSDRNPM [7] (discrete ordinate method), or SN1D [8] (deterministic method), the APOLLO code [9] solves the transport equation in integral form as a multi-group approximation by the collision probability method, in either one or two dimensional geometry.
- 2-D: DOT/DORT [10], [11], COROUT [12] or TWODANT [13], [14].
- 3-D: TORT [15], [16].

- For very complex geometries, codes based on the Monte Carlo method may be employed, e.g. MCBEND [17], MORSE [18], [19], KENO 5 [20], MCNP [21], [22] and TRIPOLI [23], [24].

3.3. Computer codes for spatial distribution of neutron induced radioactivity

Codes are available to calculate the activity induced in system materials by neutrons. In general, these codes utilize the average neutron fluxes in all of the zones representing the fixed structure of the reactor, the material compositions of the zones and the time–power histograms for the reactor lifetime operations. The outputs of these codes are the radionuclide specific activities present in the zones, and the integral over the zones can be used to estimate the total activity of the fixed component [2]. For this purpose, several codes are employed: ORIGEN-S [25], ORIGEN2 [26], FISPIN [27], DARWIN/PEPIN [28], CINDER'90 [29] etc.

4. MCNPX simulation

In my calculations I used the MCNPX version 2.7.0. MCNPX is a general-purpose Monte Carlo radiation transport code for modeling the interaction of radiation with everything. MCNPX stands for Monte Carlo N-Particle eXtended. It extends the capabilities of MCNP4C3 to nearly all particles, nearly all energies, and to nearly all applications without an additional computational time penalty. MCNPX is fully three-dimensional and time dependent [30]. It utilizes the latest nuclear cross section libraries and uses physics models for particle types and energies where tabular data are not available. This code enables precise three-dimensional modeling of the active zone and of the reactor internals. The main drawback of this calculational system was the high demand on computing resources. However, in the last few years, significant improvement in the field of computer technology opened the door towards greater use of this method. On the other hand, one of the main advantages is the use of the continuous-energy spectra in neutron transport calculations, which allows achieving more precise results.

5. Results of the simulation

Based on technical documentation from EBO Unit 1&2, I was able to create a simplified model of the VVER-440/V-230 reactor, which can be used for further calculations. In this work, I utilized the ability of MCNPX to calculate the reaction rates. Potentially occurring radionuclides are limited by the chemical composition of structural materials. Some of them can be neglected due to their neutron-physical properties, e.g. low activation rate or short half-life. I was monitoring following nuclides which contribute the most to the activation of the 08Ch18N10T structural steel: Cr (50, 52, 53, 54), Mn (55), Fe (54, 56, 57, 58), Co (59), Ni (58, 60, 61 62, 64) and Mo (92, 94, 95, 96 ,97, 98, 100) and the most probable reactions: (n,γ), (n,p), (n,d), (n,t) and (n,α).

In Tab. 1 you can see the specific activities of selected radionuclides, which were produced in the dummy elements during their first reactor campaign in EBO Unit 1 (after reducing the zone and change of the loading pattern). It was the 13th campaign (268,1 effective days).

Tab. 1. *Specific activity of selected radionuclides.*

Radionuclide	$T_{1/2}$, radiation type	A_i* [GBq/t]	σ [GBq/t]
^{51}Cr	27.7025 d, EC	1.29E+08	1.39E+05
^{55}Fe	2.73 y, EC	1.63E+07	2.63E+04
^{58}Co	70.86 d, EC, β^+	4.10E+06	4.05E+05

^{59}Fe	44.503 d, β$^-$	2.69E+06	1.81E+04
^{54}Mn	312.3 d, EC, β$^-$	1.73E+06	2.51E+02
^{60}Co	5.2714 y, β$^-$	1.36E+06	1.94E+05
^{63}Ni	100.1 y, β-	2.97E+05	1.78E+02
^{59}Ni	7.6E04 y, EC	2.24E+03	1.34E+00
^{95}Nb	34.975 d, β$^-$	9.73E+02	7.08E+01
^{57}Co	271.79 d, EC	6.48E+02	1.50E+01
^{93}Mo	4.0E03 y	1.56E+02	9.39E-02
Total specific activity:		***3.90E+08***	*2.272E+07*

Specific activity of a single radionuclide (A_i) as well as the total specific activity can be calculated based on the reaction rate (RR) as follows:

$$A_i = (1 - e^{-\lambda\tau}).RR, \qquad (1)$$

where λ[s^{-1}] is the decay constant and τ[s] is the irradiation time.

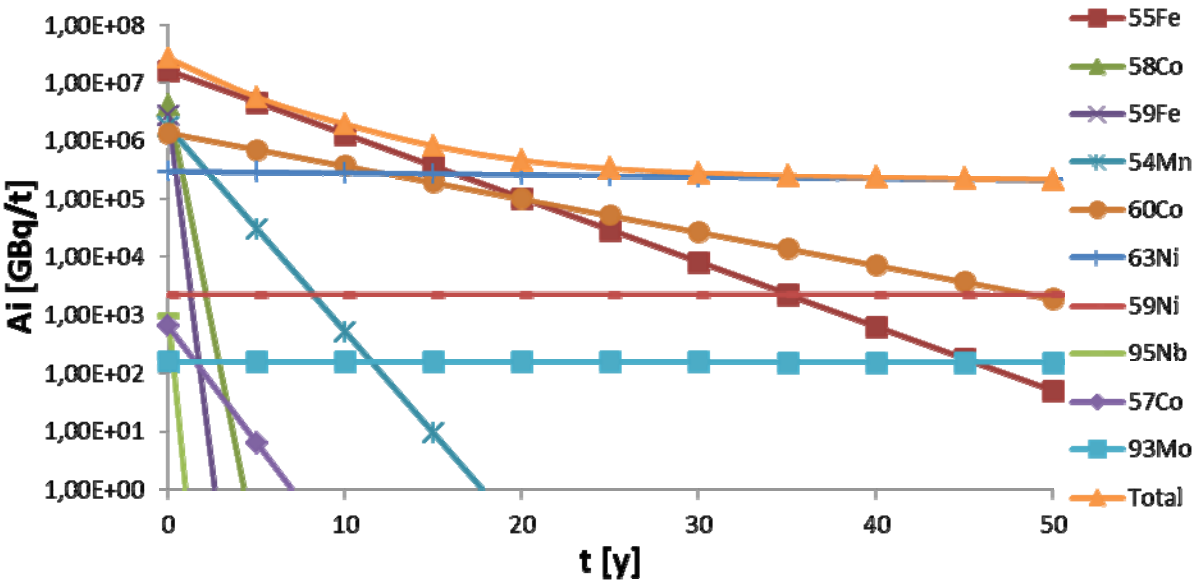

Fig. 1 *The course of specific activity after reactor shutdown.*

6. Discussion and conclusion

In this paper, I focused on the calculation of the neutron induced activity of nuclear reactor components for decommissioning purposes. The results confirm, that the most important radionuclides in the reactor components dismantling process are 55Fe (1st decade), 60Co (10 - 50y) and 63Ni (during the whole process). Another aim of this paper was to refer to the possibility to improve the accuracy of the calculations using continuous energy Monte Carlo methods. In future work, I would like to build on previously acquired knowledge and study the activation of biological shield of the reactor using the MCNPX code in combination with the CINDER'90 activation code.

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