THE USE OF CODES VISIPLAN 3D ALARA AND MATLAB FOR ASSESSMENT OF CONTRIBUTIONS FROM RADIATION SOURCES WITHIN THE DECOMMISSIONING OF NUCLEAR POWER PLANTS

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Abstract

The knowledge of radiation situation is one of the crucial requirements for the realisation of dismantling activities in a safe and economic way. The calculation of external exposure is (among the material composition and geometric dimensions) strongly dependent on the source term – activity and nuclide composition. However, the exact data regarding this source term are often not available. Thus only approximate results can be obtained. From this reason the methodology was developed which reflects this phenomenon. The calculated values (using VISIPLAN 3D ALARA code) are related to ^{60}Co only. These values are then used for calculation of the dose rates with considered activity and nuclide composition (using MATLAB code). The obtained results are compared with the "standard" results obtained from direct entering all the data to the VISIPLAN 3D ALARA code. The comparison of the results from these two approaches is presented and analysed. The results of the comparisonenable the use of this methodology in the calculation of external exposure.

1. Introduction

The decommissioning of nuclear power plants (NPP) is the end part of their life-cycle. In Slovakia, V1 NPP in Jaslovské Bohunice is currently in the second decommissioning stage with the planned duration between 2015 and 2025 [1] . According this document, the activated and contaminated parts will be cut in situ and the fragmented parts will be either stored or disposed in National radioactive waste repository in Mochovce. The components to be cut can be divided to *activated components* (the activity content is mostly due to neutron activation) – reactor pressure vessel, reactor cavity, reactor cavity bottom, core basket and to *contaminated components* (the activity content is a result of contamination of activated corrosion products and/or fission products) – pressuriser, steam generator.

In this paper, steam generator is studied from the perspective of calculation of external exposure during various tasks connected with its dismantling and final disposal of resulting radioactive waste. This component is a part of primary circuit and consists of the following parts:

- Heat exchange tubes
- Collectors
- Steam generator casing and steam collector (non-contaminated – part of secondary circuit).

As was mentioned, the source of the activity is the contamination of the inner walls of heat exchange tubes and collectors due to the flow of primary circuit coolant.

In the following chapter, the issues connected with the source term are identified and discussed.

2. The issues connected with the source term

The estimation of the activity content and nuclide inventory of activated parts can be carried out using for instance MCNP software. The exact data regarding material composition, neutron flux (spatial distribution, power and history of operation – frequency and duration of outages) must be known. Significantly different situation is in case of contamination. The level of contamination is strongly dependent (among the parameters mentioned in case of activated parts) on the creation of oxidation layer of the metallic surfaces, chemical conditions during the operation and outages, temperature and of the process of coolant's flow, the geometry and dimensions of technological equipment (e.g. pipes, elbows, etc.) and on the chemical properties of radionuclides (e.g. solubility and adhesion). From this reason, the calculation of contamination is either impossible or with high uncertainties [2] Thus the estimation of the contamination is often based on in-situ measurements. The measurements (carried out during radiological characterisation before decommissioning), however, could not reflect the real state in each part of the technological equipment. From these measurements and samples analysis average values are obtained which can vary in local places. Direct entering of these data to calculation software can lead to approximate results. During partial tasks, however, the calculations should be repeated to achieve higher accuracy. Moreover, in case of application of decontamination methods secondary waste is generated (e.g. spent ion exchange resins) and the amount, nuclide inventory and activity content can vary case-by case and the exact data will be known just in the time of realisation.These factors make the calculation of external exposure difficult and require development of such calculation methodology which is more flexible than the direct entering of approximate data into calculation software.

3. Computer code VISIPLAN 3D ALARA

For calculation of external exposure the computer code VISIPLAN 3D ALARA was used. The calculation principle is as follows [3] The photon fluency rate at the dose point near the volume source can be determined by considering the volume source as consisting of a number of point sources. By adding the contribution of every point source to the dose at the dose point the photon fluency rate at the dose point is expressed as:

$$\phi = \int_V \frac{S.B.e^{-b}}{4\pi\rho^2} dV \ (\mathrm{m^{-2}.s^{-1}}) \qquad (1)$$

where: S – source strength per unit volume ($\mathrm{n.s^{-1}}$),
ρ – distance from a point source (m),
B – buildup coefficient (-),
b – attenuation effectiveness coefficient (-),
V – volume ($\mathrm{m^3}$).

Each small source is called a kernel and the process of integration, where the contribution to the dose of each point is added up, is called "point kernel" integration.

Based on the photon fluency rate at a point it is possible to calculate the effective dose rate depending on the dose conversion factors selected in the calculations:

$$E = \sum_i h_i.\phi_i \quad (\mathrm{Sv/s}) \qquad (2)$$

where: h_i – dose conversion coefficient for photons of energy E_i (Sv per photon/$\mathrm{m^2}$),
ϕ_i – fluency rate of the photons at energy E_i ($\mathrm{m^{-2}.s^{-1}}$).

From the Eq. (1) and Eq. (2) it is obvious that the relation of the dose rate from the activity is linear, i.e. if the activity increases (decreases) n-times, the dose rate will increase (decrease) n-times (when the other parameters remain unchanged).

One of the utilities of the VISIPLAN 3D ALARA code is that the contribution from each source to the dose rate in the point can be shown which is crucial for the creation of calculation methodology described in the following chapter.

4. Calculation methodology combining VISIPLAN 3D ALARA and MATLAB codes

From the perspective of external exposure, only those nuclides are important, during whose decay γ or rtg photons are emitted. The number of different energy groups of photons emitted during decay of specific radionuclide can be characterised by *gamma ray dose constant* Γ $[C.m^2.kg^{-1}]$ [4] In practical applications the unit of *(mSv/h)/MBq*can be used. In this case, Γ represents the unshielded gamma ray dose-equivalent rate at 1 meter from a point source with given activity [5] . Using this interpretation of Γ, the contributions of the nuclides to the dose rate can be related to one reference nuclide. In the analysis, ^{60}Co as the reference nuclide was selected. This was done from two main reasons: ^{60}Co is so-called easy-to-measure nuclide which can be measured by γ-spectrometry[6] . The other reason is that this radionuclide is one of the most abundant radionuclides within the V1 NPP inventory.

Based on these assumptions, the calculation methodology can be as follows:

- Creation of calculation model in VISIPLAN 3D ALARA code with the activity of each source 1 Bq of ^{60}Co and creation of set of points where the dose rate will be calculated (so-called trajectory)
- Processing of obtained data (dose rates from each source) – input data for further calculations – in Eq. (3)depicted as *f*(scalar)
- Determination of so-called conversion vector – set as ratio Γ_i / Γ_{Co} (Γ_i represents gamma ray dose constant of nuclide i) – in Eq. (3)depicted as $\overline{converse}$ (vector)
- Vector of share of radionuclides (obtained by in-situ measurements) – in Eq. (3) depicted as $\overline{share}$ (vector)
- The total activity of the source (obtained by in-situ measurements) – in Eq. (3) depicted as *activity* (scalar).

Thus the dose rate can be calculated in matrix processing software (e.g. MATLAB):

$$dose\_rate = sum(\overline{share} * activity .* \overline{converse}) * f \qquad (3)$$

The symbol '.*' means the multiplication of an element in the n-th row of the first vector with the element in n-th row of the second vector.

The main advantage of this approach is that the variable parameters are vector of share of radionuclides and the total activity of the relevant source. In case of change of these two parameters only calculation in MATLAB using the Eq. (3) is required. The repeated entering the data to VISIPLAN 3D ALARA (which can be in case of complex models time-consuming) and repeated calculation is then avoided.

4.1 Comparison of the calculation methodologies

To compare the results obtained by calculation after direct entering the same source data to the VISIPLAN 3D ALARA code with the methodology already described, the disposal of radioactive waste (placed in fibre-concrete containers – FCC) into National Radioactive Waste Repository in Mochovce is studied. This situation was analysed in detail in[7] and in 17th Regional Seminar on Radioactive Waste Disposal (20-22 October 2014, Budapest).

The comparison of obtained results by direct entering source data into VISIPLAN code and obtained using Eq. (3) is depicted in Tab. 1, Tab. 2 and Tab. 3. The deviation was calculated as follows:

$$deviation = \frac{(D_{V+M} - D_V)}{D_V} * 100 \qquad (4)$$

where: D_{V+M} – dose rate obtained using Eq. (3) (mSv/h),

D_V – dose rate obtained by direct entering the source data into VISIPLAN(mSv/h).

Note: all the values depicted in the tables are rounded at two decimal places.

Tab. 1.*Calculated dose rates for truck driver.*

Task	**Dose rate [mSv/h]**		**Deviation [%]**
	VISIPLAN	**VISIPLAN and MATLAB**	
Transport to the repository	2.30E-03	2.35E-03	2.09E+00
Arrival to the repository	2.30E-03	2.37E-03	3.05E+00
Unloading the truck	2.30E-03	2.35E-03	1.64E+00

Tab. 2.*Calculated dose rates for assistant.*

Task	**Dose rate [mSv/h]**		**Deviation [%]**
	VISIPLAN	**VISIPLAN andMATLAB**	
Visual control of FCCs	5.15E-03	5.24E-03	1.69E+00
Unloading the truck	2.25E-02	2.28E-02	1.49E+00
Dose rate measurements	2.60E-02	2.64E-02	1.42E+00
Lifting one FCC	2.00E-02	2.02E-02	8.90E-01
Transport of FCC to the box	3.20E-02	3.22E-02	5.30E-01
Disposal of the FCC	5.30E-02	5.48E-02	3.40E+00
Covering the box	1.75E-02	1.75E-02	4.94E-04

Tab. 3.*Calculated dose rates for crane operator.*

Task	**Dose rate [mSv/h]**		**Deviation [%]**
	VISIPLAN	**VISIPLAN andMATLAB**	
Visual control of FCCs	5.50E-04	5.64E-04	2.51E+00
Unloading the truck	1.25E-03	1.27E-03	2.71E+00
Dose rate measurements	1.00E-03	1.06E-03	6.37E+00
Lifting one FCC	4.10E-03	4.22E-03	2.81E+00
Transport of FCC to the box	9.40E-04	9.49E-04	9.11E-01
Disposal of the FCC	3.00E-04	3.09E-04	2.84E+00
Covering the box	2.80E-05	2.80E-05	6.93E-07

From the depicted data it is obvious that the dose rates obtained by calculation methodology using Eq. (3) are slightly higher than the results of direct entrance to VISIPLAN code. The differences can be explained by the fact that output from VISIPLAN code (the dose rates) is rounded at two decimal places. The other fact is that during sampling the sources (chapter 3 – point kernel integration) Monte Carlo method is used. That means that the dose rates can slightly vary if the calculation would be repeated.

5. Conclusion

The calculation of external exposure during dismantling of contaminated components is connected with many uncertainties of the source term (nuclide composition and activity) due to the non-homogeneous distribution of contamination. From this reason the presented calculation methodology was developed which reflects this situation and enables flexibility when the source term is changed (as a result of in-situ measurements).

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