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Research Article

Carrying out calculations of radiation safety during unloading and disassembly of cores of spent removable parts of reactors with liquid metal coolant of submarines*

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Abstract

The results of calculations conducted to substantiate radiation safety while handling spent removable sections (SRS) of reactors with a liquid metal coolant (LMC) for nuclear submarines (NS) are presented in the article. The spent removable sections of reactors with liquid metal coolant for nuclear submarines are the sources of intense neutron and gamma radiation. Shielding should ensure the dose rate level for neutrons and gamma radiation which is not exceeding the values allowed for transportation of nuclear materials established by the NP-053-04 therefore it will attenuate emission of neutrons and gamma quanta by several orders of magnitude.

A homogeneous model of the reactor core was used for calculations. Sources of neutrons and photons in the spent nuclear fuel (SNF) of the SRS, sources of photons in the reactor control devices and in construction materials (the reactor vessel and grids of fuel rods) have been taken into consideration while conducting the calculations. The computer code MCNP-4B was employed to calculate dose rates for neutrons and photons.

In most cases direct calculations of dose rates for neutrons and secondary gamma-quanta using the MCNP-4B code provided acceptable results with admissible methodical errors. For the tasks with sources of gamma quanta direct calculation using the MCNP-4B brought unsatisfactory results due to strong attenuation.

Various methods were applied to reduce dispersion: the first one is to assign importance to the cells and the second one is the method of weight windows iteration.

Values of dose rates were obtained with acceptable errors. The results of the calculations provided necessary information to conduct operations to unload spent nuclear fuel from the SRS. The results of performed calculations were also used in the design and manufacturing of the shielding.

Keywords

Spent removable section of reactor (SRS), equivalent dose rate (EDR), method to assign various importance in the cells, method of weight windows

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Introduction

In 1962–1990 several nuclear submarines (NS) with liquid metal (lead-bismuth) coolant (LMC) in the primary coolant circuit were registered in the Russian naval forces (Zabud'ko et al. 2008). At present all these submarines are decommissioned and spent removable sections (SRS) are kept in storage facilities. Activities related to decommissioning of NSs, their utilization, storage and SNF processing must be considered in the indissoluble connection to the NPI life cycle (Zrodnikov et al. 2007).

Operation of NS reactor installation at power levels is accompanied with generation and accumulation of long-lived radioactivity in the reactor core, control and protection rods, in adjoining structures and in the coolant. Its amount in each NS depends on the overall power generation of reactors (Zabud'ko et al. 2008).

The SRS of the reactors with LMC are the sources of intensive neutron and gamma radiation (Zrodnikov et al. 2007). For gamma-quanta sources from various SRS differ by an order of magnitude and for neutrons by two orders of magnitude (Zrodnikov et al. 2004).

Handling of the SRS includes a series of technological operations when dismantling, reloading and transportation of the SRS and assemblies with spent fuel are carried out. One needs to ensure nuclear and radiation safety while performing these operations.

Calculations assuming individual approach to each SRS are necessary for:

- Substantiation of nuclear safety at each phase of dismantling;
- Assessment of radiation conditions at working places of the crew;
- Calculation of dose rate loads on the crew;
- Calculation and upgrade of the shielding.

Calculation procedure

The program code MCNP-4B was used to calculate dose rates for neutron and photon radiation. The Program MCNP-4B is attested for simulation of neutron and photon transportation while conducting calculations of radiation shielding and substantiation of radiation safety of facilities employing nuclear energy including calculations of neutron flux density and absorbed dose of photons for containers with spent nuclear fuel (The Program Complex MCNP4B 2007).

Neutron sources in fuel were taken into account while conducting the calculations. Sources of photon radiation from fission products in the fuel, from europium in the control system, from construction materials (reactor vessels and grids of fuel rods) were considered.

A homogeneous model of the reactor core was employed for calculations of radiation safety. The reactor core was substituted with a cylinder which volume is equal to that of the reactor core. In total 27 control elements were inserted into the cylinder. Five homogeneous zones were identified along the height: bottom grid of fuel rods, reflector, homogeneous fuel, compensation volume and top grid of fuel rods. The following sources were taken into consideration in the performed calculations: sources of neutrons and gamma-quanta from the spent fuel, sources of gamma-quanta from elements of the control system, sources of gamma-quanta from grids of fuel rods and the reactor vessel.

Homogenous source of neutrons and gamma-quanta from spent fuel was set in the fuel zone without profiling along its height and radius. The source from the control system was represented by 27 cylinders coinciding in size and location with the real control elements. It was assumed that each of the 27 sources of radiation contributes equally to the formation of EDR. The homogeneous volume source from fuel rod grids was assigned for the top and bottom grids. The source for the reactor vessel was defined in form of a cylindrical ring without height profiling. Separate calculation was performed for each type of the source (neutron radiation, residual radioactivity etc.). Calculation model of the SRS is presented in Fig. 1.

Dose rates were calculated in reference points. Precalculated reference points were chosen for assemblies in containers in order to estimate the maximum dose rate. For the SRS dose rates were calculated in places where the personnel are present. Location of reference points is shown in Figs 2 and 3.

Local estimate we used to calculate the dose rate. The results with errors below 5% are considered to be reliable, results with errors from 5 to 10% are estimated as doubtful and results with errors in excess of 10% are not taken into consideration (The Program Complex MCNP4B 2007). These are the values of errors of the Monte-Carlo method provided by the program. Errors related to the transfer from the real physical to the calculation model (geometry, sources, material composition, constants etc.) are not taken into consideration.

In most cases calculations of dose rates performed for neutrons and secondary gamma-quanta, direct calculation using the MCNP-4B code provided acceptable results with admissible methodical errors below 5%. For the tasks with sources of gamma quanta direct calculation using the MCNP-4B brought unsatisfactory results due to strong attenuation of radiation.

Various methods were applied to reduce dispersion: the first one is to assign importance to the cells and the second one is the method of weight windows iteration.

The method to assign various importance values consists in the approach when different importance values are assigned to the cells. In case of direct calculations using the MCNP-4B each geometrical zone (cell) is assigned with the value of importance which is equal to 1. Values of importance are increasing in the direction of particle motion (in the given case it is from the center to the periphery). The values of importance must differ between neighbouring cells by a factor of two to three (Gusev et al. 1989). While using this method one needs to split geometrical zones into smaller fragments, which complicates the geometry of the task. Importance of the cells is assigned "manually" and for complicated geometry it is not always possible to make choice of the importance value correctly

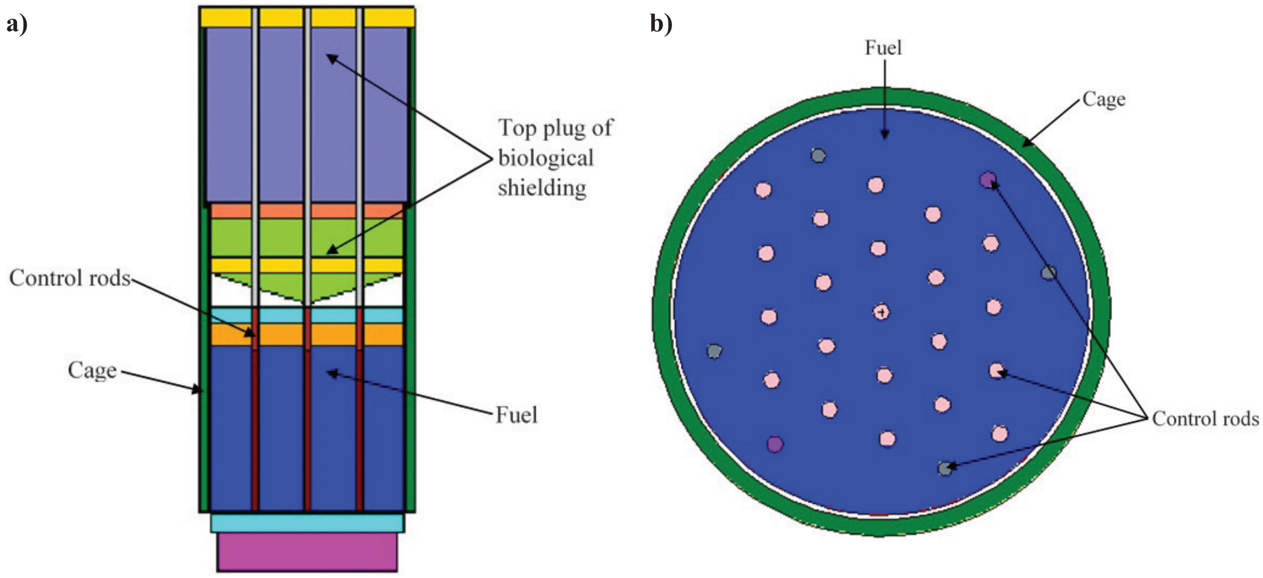

Figure 1. Calculation model of the SRS: **a)** vertical section; **b)** transverse section.

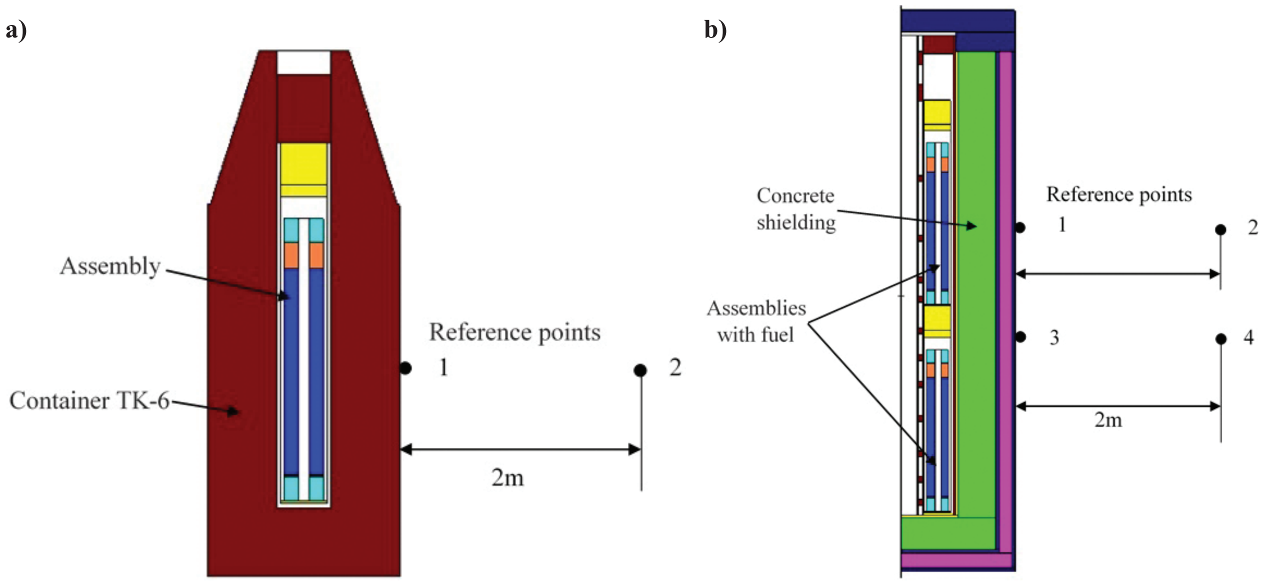

Figure 2. Location of reference points: **a)** for an assembly in the container 1; **b)** for an assembly in the container 2.

(Devkina 2018). This method was used for calculations of dose rates with errors about 20%.

The method of weight windows iteration is the automated method of weight windows generation where in difference to hybrid methods only the results of Monte-Carlo (MCNP-4B) calculations are used to obtain the weight windows.

In practice it is easier to use the iteration method than the hybrid one because it is not necessary to create the calculation model for the grid code and to intercorrelate the calculation models for MCNP-4B and the grid code. With the help of the iteration method we succeeded to obtain the EDR values with acceptable accuracy for all phases of SRS dismantling and SNF handling.

The method of weight window iteration consists of several steps. At first the calculation is carried out using the MCNP-4B code where distribution of particle fluxes is determined within the whole phase space with the help of the *fmesh* estimation. The calculated results are used subsequently in the program MESHMOD, which generates the weight windows. The simplest procedure of data reconstruction is employed in the program MESHMOD. Zero values in the grid cells are substituted with arithmetical mean value ($\Sigma\Phi i$)/n) or with geometrical mean values ($(\Pi\Phi i)1/\pi$ over the neighbouring cells. As a result one obtains sufficiently smooth distribution which however is wide from the real one. This distribution is suitable for subsequent calculations. Calculation with weight windows obtained at the first step is conducted at the second step and the results of these calculations are then used by the MESHMOD program to generate weight windows which are used at the following step. The cycle is continued till necessary accuracy is achieved. Dose estimates for the points are assigned during the last iteration. Most of the tasks are converging after the third iteration (Chernov et al. 2015). The method of weight windows iteration is more effective in comparison to the method of assignment of different importance values to the cells; however it requires higher time consumption. Relation of contribution in the EDR formation caused by different radiation components varies depending on the stage of the STS dismantling and location of calculation points. For the phase of dismantling which is shown in Fig. 5 the interrelation between the components of radiation appears as follows:

- For calculation points 5 and 6 neutron radiation is the decisive one ~ 85% from the total EDR, gamma-quanta from fission products ~ 7%, contribution from the control system ~ 4%, and ~ 3% from construction materials (2% from the reactor vessel and 1% from fuel rod grids);
- For calculation points 1–4 and 7, 8 main contribution is coming from the secondary gamma radiation ~ 73%, gamma-quanta from fission products ~ 12%, from the control system ~ 6%, from construction materials ~ 3% (1% from the reactor vessel and 2% от from fuel rod grids), and from neutrons ~ 6%.

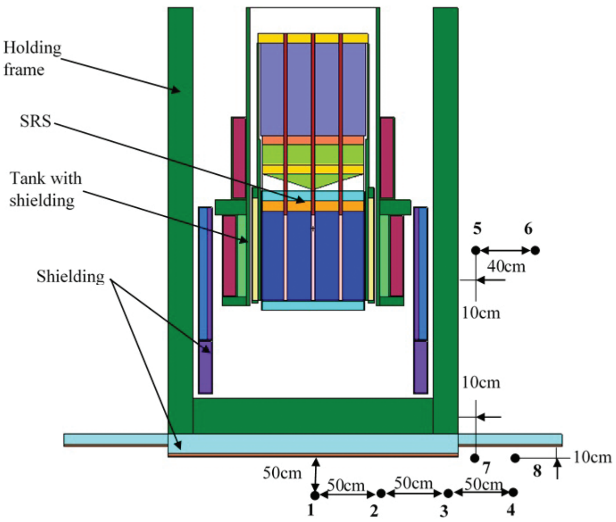

Figure 3. Location of calculation points for the SRS at one of the dismantling phases.

Maximum total calculated dose rates (from gamma-quanta and neutrons) (in µSv/h) on the surface and at a distance of 2 meters for containers 1 and 2 as well as for the SRS at a distance of 50 cm from the surface are presented in Table 1.

Table 1. Results of dose rate calculations, µSv/h

	Tk-6	TUK 108/1	SRS in the holding frame
On the surface	1171.80	1368.46	8.52
At a distance of 2 meters for containers, 50 cm for SRS	36.21	158.22	15.70

The obtained calculated dose rates for different types of radiation satisfy the requirements of the NP-053-04 (NP-053-04 2004) for transportation of nuclear materials (they do not exceed 2 µSv/h on the container surface and 0.1 µSv/h at a distance of 2 meters from the container surface).

Results of calculations for gamma-quanta mostly demonstrate good agreement with the readings of dose-meters. The results of neutron dose rate calculations show decrease in comparison with the readings of dose meters. As a whole results of calculations provided information to carry out unloading of spent nuclear fuel from the SRS. The calculated results were also used in design and manufacturing of the shielding.

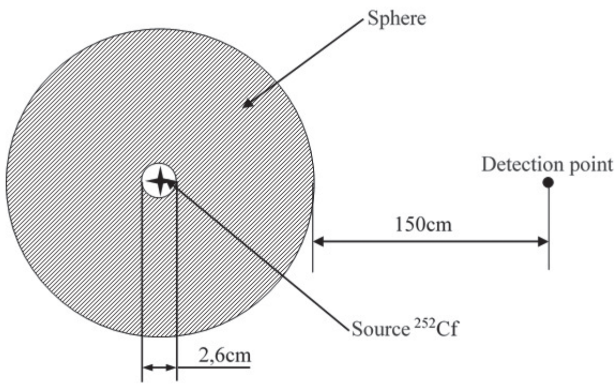

Figure 4. Calculation model for the experiment.

Calculation for the benchmark experiment used in verification of computer programs was conducted to explain the discrepancy between the experimental and calculated data for neutron dose rates. This benchmark (Trykov et al. 1985) was chosen because there are thick layers of iron in the construction of shielding. The experimental installation consisted of the iron spherical model, radionuclide neutron source 252Cf, and neutron and gamma radiation spectrometers placed on top of a special support. The source was installed in the iron sphere with radius 50 cm (See Fig. 5). Neutron spectrum was estimated. Investigation of neutron and gamma-radiation spectra (hereinafter in the test - neutron spectra and gamma radiation spectra) from the 252Cf radionuclide neutron source was carried out by a set of spectrometers which allows one to measure neutrons with energy varying from thermal energy to approximately 15 MeV and gamma-quanta with energy from 0.2 to 12 MeV. Group neutron spectra were estimated.

Comparison of experimental and calculated data is presented in Fig. 5.

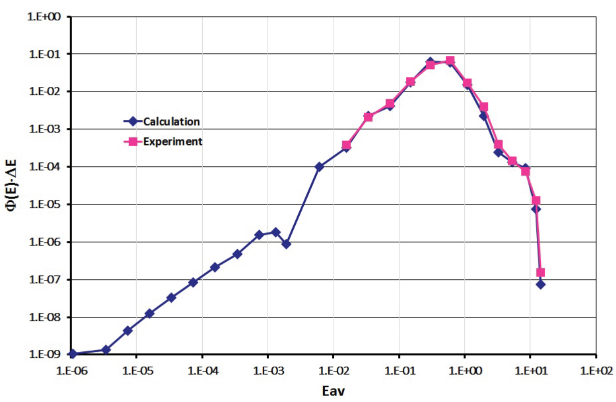

Figure 5. Experimental and calculation distributions of the neutron fluence at the detection point.

Difference between the experimental and calculated data on the integral fluence is within the error limits of the spectrometer (10%).

Conclusions

The chosen calculation model describes real construction of containers and radiation shielding quite well. The simulated sources take main types of radiation into consideration.

Sources which are used in the calculation model consider all types of radiation.

Direct calculation using the MCNP-4B code is not always providing the results with admissible methodical accuracy. For simple geometry it is better to use the method where different values of importance are assigned to cells, for more complex geometry it is better to use the method of weight windows iteration.

As a whole the calculated results provided necessary information to carry out operations on the unloading of spent nuclear fuel from the SRS.

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