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RADIOACTIVE GASEOUS-AEROSOL RELEASES FROM THE WWR-M RESEARCH REACTOR

The radiation impact of the WWR-M research reactor is possible due to the release of radioactive substances into the environment, which are generated during both operation and decommissioning. One of the operator's responsibilities is to take measures to avoid or optimize the generation and management of radioactive waste to minimize the overall impact on the environment. Gaseous and airborne wastes are released into the environment through ventilation and air cleaning systems, which are essentials of the overall reactor design. The main method for preventing radioactive contamination spread to the environment is the combination of a well-designed ventilation system having thorough cleaning of the exhaust air. An analysis of the source terms of the gaseous-aerosol emissions, a description of the existing radiation control system and special ventilation system, and an analysis of the actual gaseous-aerosol emissions are presented. The total amount of gaseous-aerosol emissions during the planned reactor dismantling activity is estimated. The sufficiency and effectiveness of the existing system to ensure the required level of reactor safety are shown.

Keywords: WWR-type research reactor, emissions, radioactive gas, aerosols, radioactive isotopes, decommissioning, dismantling.

1. Introduction

During the normal operation of nuclear facilities, radioactive waste is generated, which can lead to an impact on the population and the environment. In many cases, total prevention of such emissions is technically impractical or extremely expensive. Facilities and types of activities that create radiation risks must be designed, constructed, and operated with an appropriate level of protection for the population and the environment. Doses of exposure to the population from such radioactive emissions should be below the established limits [1].

Operations involving radioactive materials might cause radioactive air pollution. The main difference between airborne effluents and radioactive waste in the condensed (i.e. liquid or solid) state is that the airborne material does not have a definite volume and its dispersion in the environment is very rapid. Regarding gaseous radioactive waste, the term “waste” is not applicable in Ukraine, but the term “emissions” is used, namely, the gaseous-aerosol emissions.

Gaseous emissions include radioactive gases and aerosols. The effect of radioactive aerosols on the human body is mainly due to internal exposure. External radiation from radioactive aerosols is small compared to internal radiation. Although radioactive gaseous emissions are not the main waste stream from the nuclear facility, they are a major source of the potential direct environmental impact. Thus, effective control and management of gaseous emissions in both normal and emergency conditions are one of the main issues in the design and operation of a nuclear installation [2, 3].

Gaseous and aerosol emissions enter the environment through the ventilation and air cleaning systems, which are an important part of the overall design of the nuclear installation. The main method for preventing the radioactive pollution of air is the combination of a well-designed ventilation system with a thorough cleaning of the exhaust air. Ventilation and air cleaning systems should provide effective treatment of off-gas flows under normal conditions of operation, maintenance, and accidents [4 - 6].

One of the operator's responsibilities is to take measures to prevent or optimize the generation and management of radioactive waste to minimize the overall impact on the environment. This includes ensuring that gaseous and liquid radioactive releases to the environment are within authorized limits and that doses to public and environmental impacts are reduced to levels that are as low as reasonably achievable (ALARA principle). The responsibilities of the regulatory body include the exclusion of the radioactive materials within the scope of permitted activities from any further regulatory control (known as a permit) and the control over emissions of the gaseous radioactive materials originating from nuclear facilities [7, 8]. Accordingly, any facility or activity that generates radioactive waste needs to establish a dose limit and obtain the corresponding permission from a regulatory body.

The WWR-M research reactor is in operation for over 60 years and has excellent safety records. Today, the status of the reactor is an operational installation.

The operator of the reactor is the Institute for Nuclear Research (Kyiv), which possesses all the necessary licenses and permits for the reactor operation. The reactor operation is carried out following the current norms, standards, and rules on nuclear and radiation safety established by Ukrainian legislation and under the international practice and recommendations of the IAEA, ICRP, and other international organizations [9, 10]. The radiation impact of the reactor on the environment during the entire period of operation is at a low level, the emissions of radionuclides into the environment have never exceeded the established norms for their content in atmospheric air, soil, and vegetation at the control points of their measurements, which indicates the absence of man-made influence on them. The technical conditions of the reactor as a whole, alongside separate systems, allow further operation. At the same time, according to the current national legislation, the future decommissioning of the reactor should be considered by the operator as early as possible, regardless of the possible extension of the operational period [11]. The given article presents the technical and organizational aspects of using the special ventilation system (SVS) both during reactor operation and during planned decommissioning activities.

2. Source terms of radioactive releases at the normal operation

A characteristic feature of all technological processes of nuclear installations is the presence/avaiability of the sources of radiation risk caused by emissions and discharges of radioactivity. Under certain conditions, it might lead to negative effects on humans and the environment. Radioactive releases into the atmosphere are divided into two types - gases and aerosols. An aerosol is a gaseous suspension of ultramicroscopic particles of liquid or soil. Particles could have different shapes and sizes in the range from 0.01 to 100 μm.

Liquid radioactive discharges containing harmful impurities might be found in the form of solutions or finely dispersed mixtures [12]. Releases and discharges could be both regular (permanent or periodic), which are under the control of operating personnel, and emergency (usually spikes). Being included in the various movements of the atmosphere, surface, and underground waters, radioactive and toxic substances spread into the environment, get into plants, animals, and humans.

The radiation impact of the reactor is possible due to the release into the environment of the radioactive substances generated during operation. Therefore, the main sources of the radiation effect on the environment are the gases and aerosols released through the chimney due to the activation of the coolant and the air of the ventilation systems, which spreads in the atmosphere by scattering as a result of turbulent diffusion and wind transport.

During the operation of the reactor at power, many fission products are formed. Radioactive gaseous fission products enter the reactor cooling system through defects in the fuel element cladding. The main radioactive noble gases (NG) include Kr and Xe isotopes, as well as radioiodine. A distinctive feature of NG is low reactivity due to the filling of electron shells. Radioactive iodine can exist in various chemical forms such as I_2, IO_3, HOI, and CH_3I. Radioactive iodine compounds can be in the air simultaneously in a dispersed and gaseous state. They require special attention due to the difficulty of capturing and determining their content.

Among the NG, ^{41}Ar ($T_{1/2}$ = 1.83 h) is predominant (almost 99.5 %) formed as a result of the $^{40}Ar(n, \gamma)^{41}Ar$ reaction due to the activation of ^{40}Ar contained in the ventilated air through horizontal channels and a niche of the thermal column, as well as a result of the activation of ^{40}Ar dissolved in the coolant, when the reactor is operating at power. Other NG, such as Xe and Kr isotopes, are fission products that can escape from the fuel matrix. NG cannot be contained by the HEPA filters located in the chimney, and therefore the entire amount of produced gases is released into the environment.

The cladding of fuel element is the initial barrier that prevents the release of radioactive fission products generated in the fuel into the environment. The release of gaseous fission products from the fuel element occurs mainly through the cladding defects. The loss of tightness of the cladding during the operation is possible due to a manufacturing defect, due to violation of the operational conditions, due to mechanical damage of the fuel element by external objects. In the case of the cladding defects within the range of 10 ÷ 50 μm, gaseous fission products will enter the coolant (gas leakage).

In case of gas leakage defects in the fuel element, the output mechanism of radionuclides from the irradiated fuel is determined by the following processes: diffusion of gaseous, volatile, and fusible fission products from the fuel into the space under the fuel element cladding, diffusion of these nuclides from under the cladding through the leak. As a result of the organized and unorganized leakage flows for coolant, its evaporation or degassing, those gases enter the ventilation system and then released into the atmosphere. Thus, during normal reactor operation, there is an outtake of NG into the environment.

The concentration of fragmented fission products in the coolant depends on the type of fuel element and the size of defects in their claddings. Sealed (non-defected) fuel elements with the diffusion of fission products through their cladding lead to an equilibrium concentration of radionuclides in coolant, which depends on the reactor power and the level of purification of the ion exchange filter.

Thus, the main sources of contamination of coolant and ventilation systems are:

– fuel elements, claddings that have lost their tightness during operation;

– residual surface contamination of the fuel element claddings occurred during their manufacture. According to the manufacturer's certificate, the surface contamination of the fuel element of WWR-M2 type by ^{235}U does not exceed $5\cdot10^{-9}$ g/cm^2;

– fuel elements with manufacturer's defects.

There are no regularities connecting the release intensity with the reactor power. Obviously, this is related to the quality and type of cleaning equipment installed at the reactor facility. Since the release intensity is determined by nuclides mostly having the origin of the fragments, it depends on the quality of the fuel elements, the number of defective fuel elements operating in the core, and, as a result, the activity of the coolant and its leaks. Naturally, this has nothing to do with the reactor power and duration of its operation.

Gaseous radioactive releases are waste in the most mobile form and cannot be stored as generated waste. Any storage of the gaseous releases fills up quickly and therefore their storage is not economically justified. Gaseous releases cannot be stored as waste and must be processed as they are generated, they are subject to exposure and (or) purification on filters to reduce their activity to the levels regulated by the permissible release, after which they can be released into the atmosphere.

3. Authorized gaseous-aerosol releases during the normal operation

The authorized gaseous-aerosol release is the regulated maximal cumulative release when the total annual effective dose of a representative of a critical public group (outside the reactor site) due to all radionuclides in the release does not exceed the dose limit quota. The dose limit quota (dose limit) is the main radiation hygiene standard having the purpose to limit the exposure of persons of categories A, B, and C from all sources of radioactive exposure during the reactor's normal operation. Whilst the quota itself is part of the effective dose limit for category B [13, 14].

There are several methods used to determine the release limit from a nuclear installation. For example, the method based on IAEA-TECDOC-1638 [15] defines as the first step the source term and characteristics of release, which are usually based on the parameters of the nuclear installation. A dose estimate is then performed by looking at the radionuclide dispersion pattern and exposure pathways and is compared to the dose limit value authorized by the regulatory body.

For the WWR-M reactor, regulation and control of exposure to the public (category B) in its surveillance zone is carried out based on calculations of the annual equivalent doses of exposure to critical groups of the public, obtained due to the gaseous-aerosol release. The calculations made it possible to determine the critical group of the population that has the greatest value of the annual effective dose from the gaseous-aerosol release, namely: the population group of the reference age "1 year", which can be located at a distance of 300 m (border of the sanitary zone).

The results of the calculations are reflected in the document [16] and approved by the regulatory body. That document sets limits on the total amount of radioactive material, which is authorized to be released into the environment to ensure that any resulting human exposure is small and well below the legal limits. Numerical values of authorized emissions are presented in Table.

Authorized gaseous-aerosol releases

Nuclide	Authorized gaseous-aerosol releases, Bq/year	Nuclide	Authorized gaseous-aerosol releases, Bq/year
NG	$5.50\cdot10^{14}$	^{134}Cs	$1.50\cdot10^{9}$
^{131}I	$3.70\cdot10^{10}$	^{137}Cs	$2.70\cdot10^{9}$
Mixture of iodine radionuclides	$1.43\cdot10^{12}$	^{59}Fe	$6.00\cdot10^{7}$
Long-lived nuclides:	$7.10\cdot10^{9}$	^{54}Mn	$6.20\cdot10^{8}$
^{110m}Ag	$3.60\cdot10^{8}$	^{95}Nb	$1.00\cdot10^{8}$
^{58}Co	$1.40\cdot10^{9}$	^{90}Sr	$4.00\cdot10^{7}$
^{60}Co	$2.80\cdot10^{9}$	^{95}Zr	$4.00\cdot10^{7}$
^{51}Cr	$2.70\cdot10^{8}$	^{3}H	$1.00\cdot10^{12}$

4. Layout and operation of the ventilation system

The safe operation of the reactor is based on compliance with the principle of deeply echeloned protection, which ensures the maximum possible interception and fixation of radioactive products on the way to their possible release into the environment achieved by the system of organizational and technical measures for radiation safety and rapid response (control and monitoring systems, special protective barriers, systems and technologies of pollution localization).

At nuclear installations, as a rule, airflows from heavily contaminated areas are called off-gas flows. They can contain higher concentrations of radionuclides in the air than airflows of room ventilation, contaminated only by equipment or leakages from the tight-sealed area [17]. Therefore, off-gas streams must be treated before mixing with the ventilation air for occupational safety and environmental reasons. The general purpose of the ventilation and air purification systems are following:

– control of the air contamination below a safe working level;
– single filtration and air intake control;
– maintaining a directed flow from the point of the least contamination potential to the point of the greatest contamination potential;
– purification of exhaust air before releasing it into the atmosphere;
– monitoring of contaminants in the working areas and releases into the environment.

The SVS is an integral part of the radiation control system (RCS) [18]. The SVS was assembled and put into operation during the construction of the entire reactor facility. In the process of operation, SVS undergoes technological maintenance according to the regulations on preventive inspections and repairs. The system also includes an air purification system consisting of a gasholder, an expanded clay filter, and an air purification system during emergency situations.

The SVS is designed to remove air from places where radioactive gases and aerosols may appear and eject them through the chimney. The SVS functional scheme is shown in Fig. 1. The SVS runs ventilation of volumes where radioactive gases and aerosols can accumulate, and therefore, for reliability, the main elements are duplicated for all four systems. All four pairs of fans are mounted in a special room (vent center), which is located next to the reactor hall. Four pipes of different diameters run underground from the air intake points to the fans. After each pair of fans (fans of each pair work one at a time, the other of the pair starts automatically when the working one stops) air enters the chimney. The total air consumption through the chimney is 20000 - 30000 m^3/h (depending on the combination of simultaneous operation of 4 ventilation systems).

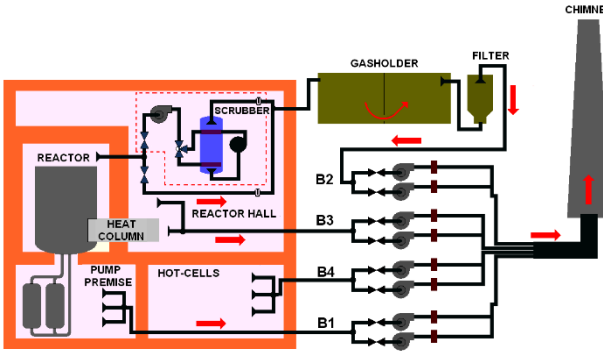

Fig. 1. SVS functional scheme.
(See color Figure on the journal website.)

SVS consists of 4 separate systems, air removal is carried out from the following places:

– space above the reactor and spent nuclear fuel storage (B2 system);
– reactor hall, horizontal experimental channels, and a heat column niche (B3 system);
– pump-premise of the primary cooling loop and space under the reactor (system B1);
– hot cells (B4 system).

All premises inside the reactor building are slightly depressurized compared to the pressure outside the building to avoid so-called diffusion leakage. Fans maintain reduced pressure, which prevents any diffuse leakage of dust, aerosols, and gases from the building. Therefore, it can be assumed that all ventilation air reaches the chimney. The ventilation volumes (inflow, exhaust) are set by the reactor design in accordance with the requirements of sanitary rules and regulations in force in Ukraine in the field of nuclear energy. Air exchange calculations for separate premises are given in the design documentation and are periodically confirmed. Accordingly, the reduced air pressure in the technological premises under the permitted modes of normal operation must be maintained within the following limits (de-pressure, mm Hg): in the niche under the reactor – 25 ± 2; in the space under the reactor cover – 10 ± 1; in the pump-premises of the primary cooling loop – 25 ± 2; in the deaerator (scrubber) – 10 ± 1; in the hot-cells – 20 ± 2; in the space above the water of the spent nuclear fuel storage – 14 ± 1.

To clean the air released into the atmosphere from radionuclides, a filtration system is installed consisting of a gasholder and an expanded clay filter. The gasholder and expanded clay filter are designed for air retention and purification during both normal operations of the reactor and emergency situations.

Due to reduce of the airflow rate inside the volume of the gasholder (500 m^3), part of the short-lived isotopes decays. The gasholder is a labyrinth room in which air is held before entering the chimney for up to 10 hours. In 1989, an expanded clay filter with a capacity of 25 m^3 was introduced into the system. The filtering element is expanded clay balls. The efficiency of the gas cleaning system is at least 90 % for gaseous iodine and 99 % for aerosols.

In the case of emergency situations with the release of a large amount of uranium fission products from the core into the space above the reactor, an additional element for the air purification is turned on – a scrubber, which is filled with a solution having pH 8 - 10 where aerosols are intensively "absorbed".

Detection units are used for continuous radiation monitoring of the volumetric activity in releases through the chimney. Signals from these sensors are transmitted to the emission control radiometer as part of AKRB-06. The airflow rate in the chimney is measured continuously using a flow meter.

Taking into account the fact that critical elements were replaced with new ones (electric motors, electrical panels, and electrical equipment, electrical cables, etc.), there are reasons to believe that the service life of the SVS is at least 30 years, subject to the timely replacement of system elements.

5. Actual gaseous-aerosol releases at the normal operation

The dynamics of gaseous-aerosol releases from the reactor for the period 1998 - 2020 are presented in Fig. 2. Operating period of 20 years was chosen because it provides sufficient data to identify characteristics and trends in releases relevant to the operational practices, allowing analysis of any trends. According to the data of systematic measurements, the activity of NG radionuclides in the releases (only when the reactor is operating at power) is caused by radionuclides, on average: ^{41}Ar – 95 %; ^{85}Kr – 0.8 %; ^{88}Kr – 2.5 %; ^{135}Xe – 1.7 %. The analysis of the release values for NG and iodine radionuclides into the atmosphere for 2012 - 2014 was impossible due to the lack of operational time of the reactor at power during this period due to repair work performed on the pipelines of the primary loop.

During the entire operation of the reactor, there were no violations that would lead to exceeding the limits and conditions of normal operation. Most of the violations were related to the automatic unscheduled shutdowns. Such situations are representing approximately 85 % of all registered cases. About 8 % of violations were due to equipment malfunctions and 7 % were due to erroneous actions of personnel. There was no contamination by radionuclides of the premises of the free access zone above the established control levels. Unforeseen contamination of the premises of the control zone occurred mainly due to erroneous actions of the personnel (≈ 95 %). Doses of external exposure for personnel did not exceed the established control levels.

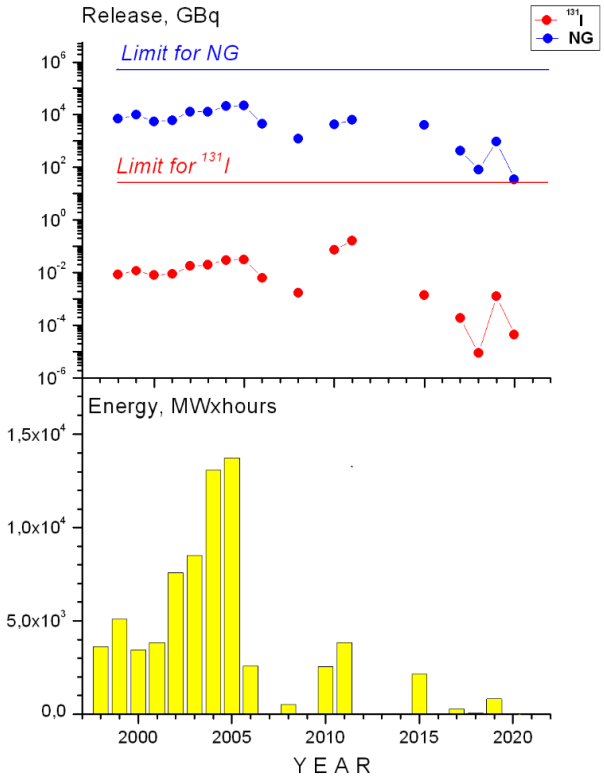

Fig. 2. Dynamics of the gaseous-aerosol releases. (See color Figure on the journal website.)

6. Source terms of radioactive releases at the decommissioning

Decommissioning of a research reactor, like any other nuclear facility, is an integral and inevitable stage of its life cycle. The current normative document [11] defines decommissioning as "the stage of the life cycle of a nuclear installation starting after the end of the termination of operation phase and when the complete or limited withdrawal of the nuclear installation site from the regulatory control is achieved". The essence of decommissioning consists of the consistent implementation of a whole set of administrative and technical measures aimed at stopping any activity related to the functional purpose of the reactor and transferring it to the ecologically safe state that does not require control by the regulatory authorities. For any reasons that lead to decommissioning, this is a mandatory stage of the life cycle and requires careful planning of both the decommissioning process as a whole and its individual components using a large volume of design and technical documentation.

The main difference between the operation commissioning and decommissioning tasks is that the purpose of the operation is the routine execution of well-

known repetitive actions while decommissioning requires many unique and non-standard activities with possible unknowns.

After the final shutdown of the reactor, and especially after the completion of the termination of the operational phase (after the removal of the spent nuclear fuel), the radiation situation in the reactor premises stabilizes. A number of factors of radiation hazard such as the presence of NG and iodine, cease to exist. The amount of radioactive substances in the reactor is decreasing; the number of areas and operations where changes in the radiation situation are possible is sharply reduced.

Taking into account the fact that the decommissioning process itself is one of the stages of the reactor's life cycle and, in fact, one of the operating modes, all rules and fundamental principles of safety should apply to this stage. On the other hand, at the decommissioning stage, the reactor will be a fundamentally different facility compared to the operational reactor. For the reactor in operation, the main source of hazard, and therefore the main object of control, is the reactor core, while for a reactor in the decommissioning stage, when there is no reactor core, the main source of hazard is radioactive process media and radioactive equipment. In practice, this will mean that the entire RCS is subject to review, starting with the principles of its operation and organization and ending with the value of control levels.

The organization of the radiation protection system during the decommissioning will be a completely logical continuation of the existing system during the reactor operation. The functioning of the radiation protection system during the decommissioning will be an integral part of ensuring the standard operational mode. From the beginning of the decommissioning, the existing system will be adapted to meet the requirements determined by the nature and composition of such work [19, 20].

The dismantling technology includes the technological processes of directly dismantling the equipment from the installation place, moving it to the area of processing, fragmentation, packaging of fragments, and moving containers (packages) to radioactive waste processing sites or to storage facilities organized on the reactor site (or outside). It is also possible partially or completely to segment the equipment at the installation place and store it without segmentation.

Such activity leads to an increase in the emission of radioactive and non-radioactive gases and aerosols. During regular execution of dismantling works, there will be no uncontrolled emission by air due to the presence of barriers, such as pressure barriers, closed tents, ventilation system filters, etc. Dismantling works will be carried out in a such way that the air pollution does not exceed threshold values in the atmospheric air, which will be the result of the controlled emissions. All air pollutants from the dismantling and decontamination activities will be vented through the existing ventilation system adequate for most activities. The negative impact on atmospheric air will not exceed the threshold values.

From all planned decommissioning operations [21, 22], only those activities that represent a significant radiation hazard were considered, namely:

– dismantling/disconnection of pipelines of the cooling loop;

– segmentation of the extracted pipeline components;

– demolishing the reactor's biological concrete shield and the protective walls of the hot cells.

Assessments for each of these operations are discussed in the next two sections of this article. They were developed based on accepted engineering practice, taking into account the experience of similar works. Conservative assumptions have been made, so the estimate can then be refined and the release quantification can be improved.

7. Estimation of release at the cutting of metal components

Over the past decades, a large number of pilot and demonstration projects have been implemented abroad for the decommissioning of various nuclear and radiation-hazardous facilities, including research reactors. Their tasks included the development and application of traditional and commercially available industrial technological means and equipment, as well as the development and implementation of innovative technologies and technical means that facilitate the performance of work and contribute to increasing their efficiency and safety. Based on the implementation of these projects and the experience gained, it was concluded that the decommissioning of research reactors of any type and power is adequately equipped with proven and commercially available technological means, specialized equipment, and tools, which in most cases require only minor adaptation or modification taking into account specific requirements and conditions of their application [23].

All the variety of cutting methods can be reduced to two categories: fire and non-fire methods. Non-fire methods include: mechanical cutting (saws and guillotine shears, abrasive tools, circular cutting machines, rope cutting). Fire methods are electric arc cutting, plasma-arc cutting, oxygen-acetylene cutting, and explosion cutting. However, the indicated cutting methods are not universal, each of them has its advantages and disadvantages. Thus, mechanical cutting methods, accompanied by a small output of

the sludge waste, require precision in tool handling, control, and management of the cutting process. This limits the possibilities of their use for cutting large-sized equipment and materials of great thickness due to the need to install specialized technological equipment.

The choice of the cutting method for the contaminated material is determined by the generation rate of radioactive aerosols that are formed, which are determined by the values of the output coefficients per unit of cutting length of the contaminated surface release value (RV) or per unit of the cutting area.

The assessment of activity emitted during cutting is based on an engineering justification of the effects of different cutting methods. There are two methods of defining the quantity of activity released during cutting contaminated materials:

1. Release per unit length cut per unit surface contamination with units of $Bq \cdot m^{-1} \cdot [Bq \cdot cm^{-2}]^{-1}$. The RV, for each method, is independent of the cut width but different values may be appropriate for different materials. The value of the release parameter takes into account activity released not only at the cut but also from adjacent material.

2. Release Fraction (RF) is a dimensionless factor, which is then multiplied by the area of contaminated material cut and the surface contamination level. The way the activity arising from adjacent areas is taken into account in the RF needs to be carefully defined.

The complexity for contaminated objects is that the cutting operation can affect material away from the actual cutting area. For example, by vibration or deformation from mechanical methods and heating of adjacent metal when hot cutting. RV represents this effect better than the RF because, conventionally, an RF cannot have a value greater than 1 and an extra multiplier would be needed for the area other than that of the cut which is affected.

It is possible to relate the two methods by:

$$A = \mathrm{RV} \cdot L \cdot \mathrm{SC} = \mathrm{RF} \cdot L \cdot 100 \cdot \frac{W}{10} \cdot \mathrm{SC},$$

where A – activity released, Bq; RV – release value, $Bq \cdot m^{-1} \cdot [Bq \cdot cm^{-2}]^{-1}$; L – length of cut, m; SC – surface contamination, $Bq \cdot m^{-2}$; RF – dimensionless release fraction; W – width of cut, mm.

Therefore:

$$\mathrm{RF} = \frac{\mathrm{RV}}{10\,W}.$$

For releases from the cutting of contaminated material, data is available from a number of different references and there are some conflicting conclusions on release values/fractions. The quantity released depends strongly on a number of parameters such as: radionuclides and their compounds which make up the contamination, how strongly adhered the contamination is to the substrate material and the experimental procedure for measuring aerosols as well as the shape of the component being cut and other parameters.

The total amount of activity released into the surrounding space will depend on the number of cuts of each type. The pipeline system of the primary cooling loop consists of pipes with a diameter of 219, 325, and 370 mm (wall thicknesses of 10 and 12 mm) made of SAV aluminum alloy. It is planned to remove large fragments of pipes from the pump premise with subsequent segmentation at the cutting area in the reactor hall [24]. It is estimated that 10 - 15 cuts will be required onsite and 30 - 35 cuts at the cutting area The representative value of the internal contamination of the pipe surface was estimated as 600 Bq/cm^2 according to the data on activity measurements during the replacement of pipeline segments [25]. Therefore, the total emission of activity into the air of the room is estimated as $1.7 \cdot 10^6$ Bk, and taking into account the efficiency of the ventilation system (more than 90 %), the emission of activity to the environment will not exceed $1.7 \cdot 10^5$. This value is much lower than the established permissible levels (see Table). The data show the dominant fission product is ^{137}Cs and the dominant activation product is ^{60}Co, other radionuclides are present in much smaller quantities. The measured ratio of ^{60}Co and ^{137}Cs activities ranges from 60 : 40 to 90 : 10 %. The results of alpha-spectrometry of smear samples showed that contamination with uranium isotopes is practically absent ($\sim 10^{-4}$ Bq/cm^2).

8. Estimation of release at the demolishing of concrete components

The most proven and traditional methods for the destruction of concrete and reinforced concrete structures are punch methods of destruction [26]. In the conditions of the reactor premises, which are characterized by cramped working space and difficult accessibility, these methods and developed means are most suitable for destruction during dismantling. Destruction of the biological protection is proposed to be done by hydraulic hammer, for example, installed in "Brokk" [21]. Such a decision is acceptable in relation to both radiological and economic aspects. Dismantling will begin from the inside of the top of the biological protection down so that the activated material is removed first. As soon as the concrete extraction reaches the cast iron rings around the reactor, the rings will be transferred to the dismantling site to be cut with a saw or other available device. Then they

are characterized, and the material is loaded into containers for transport outside the site.

Analysis of the modern practices estimates the emission parameters during the demolition (dismantling) of buildings and structures contaminated with radioactive substances showed that, apart from the use of a few empirical data on the aerosol release, there are currently no approaches fully equipped with a constant basis. This might be explained by the small number of research in this area and, accordingly, the lack of measurement data at the sites of nuclear installations during practical work, the great variability of the conditions of their implementation, and the structural and technological features of each object being dismantled (demolished). In view of this, an empirical approach using a five-factor formula was chosen to estimate the emission source [27]. The range of used values of each component is taken from [28] and given in parentheses. All components of this formula depend on the features of the technological process of destruction, including the order of operations:

$$ST = MAR \cdot DR \cdot ARF \cdot RF \cdot LPF,$$

where ST (source term) – release value; MAR (material at risk) – maximum or expected amount of radioactive substances that can be affected by a given physical impact; DR (damage ratio) – MAR fraction being actually affected by the demolition (dismantlement) of structures. It is estimated based on the reaction of structural materials to the type and level of impact (0.1 ÷ 0.9); ARF (airborne release fraction) – DR fraction released into the air in the form of aerosols available for subsequent transport in the atmosphere ($6 \cdot 10^{-6} \div 3 \cdot 10^{-3}$); LPF (leak path factor) – ARF fraction (airborne release fraction) that has passed through the deposition system (air ducts, industrial premises), emission reduction system (misting, contaminant retainers), trapping and filtering system (HEPA filters, sand filters) and escaped from technological premises, tanks and protective shielding (0.1); RF (respirable fraction) – the fraction of radioactive material being present in the form of air particles carried by air and inhaled by humans (0.05 ÷ 0.8).

The last four factors are sometimes combined into an emission factor (EF) by which the MAR must be multiplied, where:

$$EF = DR \cdot ARF \cdot RF \cdot LPF.$$

If we use the specified range of values, we could have $EF = 3 \cdot 10^{-9} \div 2 \cdot 10^{-4}$. It is estimated that the total amount of the activated concrete should be about 42 t [21], and the representative value of the activity is $\sim 10^3$ Bq/g. Therefore, the total emission of the activity to the environment will not exceed $\sim 8 \cdot 10^6$ Bq, which is also much lower than the established permissible emission levels.

9. Conclusions

The RCS has been implemented and is working effectively at the reactor facility. It was confirmed that the radiation impact on the environment during the reactor operation is at a low level. Emissions of radionuclides into the environment never exceeded the established standards, which indicates the absence of man-made influence on the environment. Provided that the current technological process of operation is preserved, and technological and organizational barriers remain unchanged, further operation of the reactor is possible.

An assessment of the potential radiation impact on the environment during the full-scale execution of the planned decommissioning works was carried out. The assessment predicts a total aerosol release into the environment at the level of $\sim 8 \cdot 10^6$ Bq throughout the whole decommissioning project. This value of emission is much lower than a similar one during the reactor operation. The technical condition of the SVS allows its further operation; therefore, it is quite appropriate to use it for decommissioning without any reconstructions or modifications. In general, by analyzing the location of the reactor site, the chosen decommissioning option, the planned procedures, and measures for the dismantling of the reactor, it was established that there is a low probability of radiation hazard to the population and the environment during the reactor decommissioning.

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РАДІОАКТИВНІ ГАЗО-АЕРОЗОЛЬНІ ВИКИДИ ДОСЛІДНИЦЬКОГО РЕАКТОРА ВВР-М

Радіаційний вплив дослідницького реактора ВВР-М можливий за рахунок виходу до довкілля радіоактивних речовин, що утворюються при експлуатації і при знятті з експлуатації. Одним із обов'язків оператора є вжиття заходів для уникнення або оптимізації генерації та поводження з радіоактивними відходами з метою мінімізації загального впливу на навколишнє середовище. Газоподібні та повітряні відходи викидаються в навколишнє середовище через системи вентиляції та очищення повітря, які є важливою частиною загального проекту реактора. Основним методом запобігання радіоактивному забрудненню довкілля є поєднання добре спроектованої системи вентиляції з ретельним очищенням відпрацьованого повітря. Подано аналіз джерел утворення газо-аерозольних викидів, опис існуючої системи радіаційного контролю і системи спеціальної вентиляції, аналіз фактичного газо-аерозольного викиду. Оцінено загальну кількість газо-аерозольних викидів під час запланованої діяльності з демонтажу реактора. Показано достатність та ефективність існуючої системи для забезпечення необхідного рівня безпеки реактора.

Ключові слова: реактор типу ВВР, викиди, інертний радіоактивний газ, аерозолі, радіоактивні ізотопи, зняття з експлуатації, демонтаж.

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