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RADIOLOGICAL IMPACT ASSESSMENT FOR POSSIBLE ACCIDENTS DURING THE DECOMMISSIONING OF THE WWR-M REACTOR

The WWR-M research reactor is planned for decommissioning. An essential component of the safety analysis report is the analysis and assessment of potential accident scenarios during decommissioning. This article presents an approach to identifying hazards and possible radiological consequences of the most dangerous initial event. The assessment results show that the radiological consequences will be significantly lower than the established standards.

Keywords: WWR-type research reactor, decommissioning, accident, exposure, release.

1. Introduction

Decommissioning of nuclear facilities requires proper planning and confirmation of the possibility of safe dismantling. Current safety standards require a proper safety assessment to substantiate the decommissioning plan for such a facility [1, 2]. During decommissioning, it is necessary to perform a variety of complex tasks related to the occurrence of radiological and non-radiological hazards, which can significantly affect not only staff, but also the population and the environment. The safety case for a nuclear facility is a strategic document specifically designed to analyze these hazards and the specific measures required to reduce the risk. Providing a safety case is one of the three main pillars on which the safety of a nuclear facility rests. The other two are a preliminary hazard assessment and the application of appropriate technical and management approaches to reduce the hazards and limit their consequences. The key issue in the decommissioning of nuclear facilities is the gradual elimination of hazards through phased decontamination and dismantling, which must be carried out safely and within the approved safety case.

The greatest difficulty in developing a safety analysis report (SAR) lies in analyzing potential accident scenarios that may occur during decommissioning. It is generally accepted that the radiological hazard associated with a nuclear facility undergoing decommissioning is significantly lower than during its regular operation [3]. It is due to the fact that after the removal of the spent fuel and operational radioactive waste, the amount, composition, and distribution of residual radionuclides are significantly reduced. Operations related to dismantling and decontamination often lead to new or unforeseen situations. Such events can occur during decommissioning, although usually on a much smaller scale than during operation (mainly due to the absence of high temperatures and pressures).

In any case, abnormal events and errors should not cause negative impacts on staff, population, and environment. IAEA recommends using a systematic approach to identify these hazards, which includes: hazard identification and initiating events; hazard screening; identification of the possible scenarios; evaluation of results; and identification of the safety controls [4]. It is critical for the safety assessment that all foreseeable initiating events and accident scenarios are identified. The safety analysis should identify all relevant scenarios arising from emergency situations in which the selected hazards could be realized [5].

Documentation package is being developed for the planned decommissioning of the WWR-M reactor, including the corresponding SAR [6]. The proposed structure includes the following: decommissioning activities, hazard identification, hazard analysis, hazard assessment, and hazard or risk control [7].

The given article presents the results of an analysis and assessment of anticipated decommissioning events and assesses the potential for adverse impacts. The analysis considers events that differ from those previously assessed for reactor operation or maintenance. The results of the analysis indicate that decommissioning activities can be performed in a manner that will not significantly impact safety.

2. Classification of the emergency situations

2.1. Terminology of events, incidents, accidents

The basic nuclear law of Ukraine defines a radiation accident as an event resulting in the loss of control over a nuclear facility or a source of ionizing radiation. Such an event results in or may result in exposure of people and the environment that exceeds the permissible limits established by safety norms, rules, and standards [8]. A similar definition is provided in the state sanitary norms, where a radiation accident is described as an unplanned hazardous event at any facility with radiation or radiation-nuclear technology [9]. This definition is compiled when two necessary and sufficient conditions are met: a) loss of the regulatory control over the source; and b) actual (or potential) exposure of people associated with the loss of the regulatory control.

The given definition of the radiation accident encompasses a wide range of events, such as depressurization, theft, or loss of sealed sources. Thus, any unplanned event that meets these conditions and occurs at a power-generating facility, transport-energy, research, or industrial nuclear reactor qualifies as a radiation accident, regardless of its causes and scale.

In international practice, a different set of terms is used [10]: *event*, *incident*, and *accident*. An event is defined as any occurrence – including human errors during operation, equipment failures, or other malfunctions, as well as intentional actions by others – whose actual or potential consequences cannot be ignored from a protection or safety standpoint. The word "incident" is often used to refer to events that are essentially represent minor accidents, differing from them only in severity of consequences. While this distinction is rarely observed in common usage, an incident can be minor or major, just like an accident, but unlike an accident, it can be created intentionally.

Events related to decommissioning can be classified in different ways. The main international classification is the International Nuclear Event Scale (INES) [11], which focuses on the radiation and nuclear safety implications of events at nuclear facilities. Within this scale, events are classified into seven levels. Events at levels 4–7 are called "accidents", and events at levels 1–3 are called "incidents". Events with no safety significance are classified as "below the scale event/level 0." Events unrelated to radiation or nuclear safety are not classified according to this scale.

The terminology used in the INES – particularly the terms "incident" and "accident" – differs from that used in safety regulations. Therefore, particular care should be taken to avoid confusion. In practice, incidents are defined as events of a smaller magnitude than accidents. Events of no safety significance are classified as "below the scale/level 0." Many events that are not significant to radiation or nuclear safety are not classified, though they are often considered indicators of unsafe practices and precursors to more serious events. It is also worth noting that industrial (non-radiological) events (e.g., falls from height or exposure to toxic fumes) are not specifically defined in INES or are classified at level 0 regardless of their severity.

2.2. Probability of initial events

An emergency situation is a stochastic event, the exact occurrence of which cannot be precisely determined. While the exact time of an event cannot be predicted, its probability (or frequency) can be determined. The classification of events by qualitative probability is presented below in Table 1 [12].

Table 1. **Qualitative probability classification**

Event	Probability range, per year	Definition
Anticipated	$> 10^{-2}$	Events that may occur several times during the lifetime of the facility (incidents that commonly occur).
Unlikely	$10^{-2} > \ldots > 10^{-4}$	Events that are not anticipated to occur during the lifetime of the facility. Natural phenomena of this class include: earthquake, major flood, maximum wind gust, etc.
Extremely unlikely	$10^{-4} > \ldots > 10^{-6}$	Events that will probably not occur during the lifetime of the facility.
Beyond extremely unlikely	$< 10^{-6}$	All other accidents.

One of the methods define the probability of an initial accident event is the use of historical data. Historical accident data can be utilized if they represent the frequency of the initial events for a specific scenario rather than the frequency of the entire scenario. While information on specific anomalous events during decommissioning projects is partially available from various sources [13], there is no systematic international collection, classification, or comparison of these disparate fragments. Furthermore, there is a notable lack of comprehensive descriptive material internationally on identification, selection, development, and root causes of the typical accidents. It should also be noted that most cases are available only within national or corporate databases and have limited access.

Given the limitations of historical data, expert assessment represents another method determine such probability. Experts must use their professional judgment to extrapolate from their experience and determine the significance of a particular problem from an emergency perspective. However, there is no absolute guarantee that all potential incident situations, causes, and consequences are considered. Different aspects of hazard assessment are sensitive to expert assumptions, which means that various experts, using the same information, may generate different results when analyzing the same problem.

The probability of emergency situations is conditionally assumed as follows:

– for initial events of an external nature – 10^{-7} per year;

– for initial events associated with equipment failure – 10^{-5} per year;

– for initial events caused by staff errors – 10^{-4} per year.

2.3. Safety criteria for emergencies

Safety analysis consistently and systematically assesses the consequences of the decommissioning activities in accordance with established safety criteria. The choice of methods for execution of decommissioning works is based primarily on the information about the sources of contamination, the estimated potential airborne releases and the possible levels of exposure of the personnel involved. If the safety assessment confirms that limit values will not be exceeded during planned decommissioning activities or in the event of an accident, safety is considered justified. Otherwise, measures must be implemented to ensure safety is brought to an acceptable level.

Safety assessment during decommissioning must consider three key categories of safety to ensure that exposure does not exceed established limits:

– *Radiation safety:* the aspect related to worker exposure to radiation.

– *Occupational safety:* the aspect concerning worker interaction with equipment and the working environment.

– *Technological safety:* the aspects related to the operational processes of the system or equipment.

Safety is defined by the quantitative criteria, such as the maximum amount of radionuclides released into the environment, the dose limits of ionizing radiation for staff and population, and the probability of an accident. An approximate estimate of the levels of radiation exposure to staff and the population is provided in Table 2 [14].

Table 2. **The level of radiological consequences**

Radiological consequences	Level of exposure, mSv/yr	
	Staff	Population
Insignificant	< 0.1	< 0.01
Minor exposure	0. ÷ 1	0.01 ÷ 0.1
Moderate exposure (under dose limit)	1 ÷ 20	0.1 ÷ 1
Major exposure (above dose limit)	20 ÷ 50	1 ÷ 5
Critical exposure	> 50	> 5

The Ukrainian national legislation establishes specific limits for the emergency exposure of staff [9]. To prevent the development of an accident that could lead to catastrophic consequences, planned increased exposure is permitted for the emergency staff, provided that all measures are taken to ensure their total exposure does not exceed 100 mSv. Furthermore, in cases where work is performed to save lives, all possible measures must be taken to ensure emergency staff do not receive an equivalent dose to any organ exceeding 500 mSv.

3. Composition of the reactor decommissioning activities

The current planning for the decommissioning of the WWR-M reactor is outlined in a concept that adopts an immediate dismantling strategy [15]. This concept defines and justifies the essential technical and organizational measures for the preparation and execution of decommissioning, detailing the sequence of works and the necessary conditions for their completion.

Planned decommissioning activities include the dismantling, segmentation, and decontamination of equipment, followed by the temporary on-site storage and off-site transport of radioactive waste for processing or disposal. The majority of these dismantling activities involve mechanical and thermal cutting. The specific challenges of this work are determined by several factors [16]: high radiation fields from contaminated systems and equipment; the confined space of the premises, which are filled with various contaminated components; and the necessity to dismantle and fragment large systems on-site before their removal and transportation.

For the analysis of potential emergency situations, the following activities with significant radiation hazards were considered:

– dismantling/disconnection of the reactor vessel and its internals (heat column and channels) [17, 18];

– dismantling/disconnection of the large-scale equipment and components of contaminated systems (pumps, filters, heat exchangers, valves, secondary cooling circuits, water purification systems, etc.) [19];

– dismantling/disassembling/cutting of the large-scale components available after dismantling the reactor and all pipes, supports, and equipment installed on those pipelines;

– dismantling of all equipment (mechanical tools, stands, lifting devices and local vehicles, power supply system);

– demolishing of the concrete shielding, concrete reactor basement, and shielding walls of "hot-cells";

– dismantling of the cooling pond (dismantling of the internal structure, aluminum tank, etc.);

– removal of the spent filters from the ventilation systems, pump-compressor pipe, and faulty ventilation equipment and air conditioners (in order to restore the ventilation system for use);

– dismantling/cutting of the ventilation stack and disassembling of all components of the ventilation system where the ventilation is no longer required;

– dismantling/disassembling/cutting of all auxiliary components and systems (including the drainage system).

Preliminary assessments have indicated that accidents with the auxiliary equipment (e.g., ventilation, power, or heat supply systems) will not lead to significant consequences for staff, the public, or the environment. Given the slow pace of the decommissioning itself, there is sufficient time for implementing the corrective measures in any situation. Additionally, since such accidents might lead to significantly greater negative consequences during the operation phase, most of this equipment has standard systems for duplicating functions or compensating measures. Consequently, these types of accidents were excluded from consideration.

Certain events were also excluded from this analysis. Fuel-related accidents are not considered, as decommissioning activities begin only after the nuclear fuel has been removed from the reactor. These activities are also not specific to decommissioning, as they are mostly the same as those carried out during the reactor's operational phase. Similarly, off-site radioactive waste management (transportation and disposal) is not included because it largely consists of activities that also occur during operation. The loss, theft, or unauthorized use of radiation sources is also excluded, as these events can occur at any time throughout the reactor's life cycle, and they are addressed under physical protection and radiation source security regulations.

4. Analysis of risks caused by events and incidents

4.1. Events during the reactor operation

During the operation, no violations occurred at the reactor that would have resulted in exceedance of the normal operating limits and conditions. Most violations were associated with automatic unscheduled reactor shutdowns. Such situations accounted for approximately 85 % of all registered events. About 8 % of violations were due to equipment malfunctions, and 7 % were caused by personnel errors. Unscheduled shutdowns (automatic reactor shutdowns) occurred for the following reasons: short-term power outage, equipment failure, errors of operational staff, or a change in a parameter value (higher/lower) from the set value. There was no contamination of the free-access area premises with radionuclides above the established control levels (20–50 Bq/cm²), nor were aerosol concentrations above the established control levels (50 Bq/m³). Unscheduled contamination of the limited-access area premises occurred primarily due to staff errors. Nevertheless, external and internal radiation doses to staff did not exceed the established control levels for radiation workers.

4.2. List of potentially dangerous cases

The initial step in developing a safety assessment for decommissioning activities is to identify existing and potential hazards (both radiological and non-radiological). This includes hazards with potential radiological or non-radiological impacts on workers, the public, and the environment under emergency conditions during decommissioning.

Subsequently, engineered and administrative controls should be identified to prevent, eliminate, or mitigate these hazards and their consequences. It is critical that all foreseeable initiating events and accident scenarios are identified.

An external events review was conducted to assess the impact of natural and anthropogenic events on the radiological consequences of the decommissioning activities. The analysis of the reactor site location, the selected decommissioning option, planned activities, and the identified hazards resulted in the identification of the following types of possible accidents/incidents during the work implementation: events due to extreme external natural impact; events due to external man-made impact; and events due to internal impact. A list of potential events that could occur during the work was also identified. The identified hazards are briefly analyzed below.

4.3. Cases due to extreme external natural impact

An analysis of cases due to extreme external events is presented in the SAR [20] for reactor operation. It is noted that the reactor buildings and structures meet the requirements for ensuring the strength of the bearing capacity and the operability of building structures in terms of seismic resistance. The reactor is built outside the zone of possible floods, above the level to which groundwater can rise, and is equipped with a reliable drainage system for sewage, so flooding of the reactor premises cannot occur. Lightning discharges will not lead to emergency situations due to a dedicated lightning protection system compliant with applicable standards. Strong winds, which can generate additional loads on some structures and cause local damage to the building, such as damage to the roof or windows, will not affect radiation safety. Any other natural events, such as extremely high or low temperatures, etc., will not have any impact on the reactor building.

4.4. Cases due to external impact

There are no other nuclear installations in the vicinity of the reactor site; the impact of those could lead to a disruption of the normal decommissioning regime. External extreme man-made impacts will not disable the power supply system since the reactor site is located outside the zone of influence from possible chemical contamination due to accidents at the urban enterprises, as well as outside the zone of fire-hazardous facilities. Other facilities that could damage the reactor's power supply elements are situated outside their zone of impact in emergency situations.

Fires in the immediate vicinity of the site. Temporary uncontrolled or permanent storage of flammable substances or materials in the immediate vicinity of the reactor is not expected. The site is located on the territory of the Institute for Nuclear Research, and there is no threat of a forest wildfire spreading from the Holosiivsky Forest. The territory of the Institute is protected by a special unit of the National Guard, which also minimizes the threat of a fire in the area adjacent to the reactor building.

Accidents on transport routes. The nearest main railway track is located 3 km from the reactor site; an accident on the railway will not affect the conditions of the reactor. The main streets with intensive road traffic are located 400 m from the reactor site. Accidents related to car accidents cannot create an emergency situation at the reactor, and therefore are not considered.

Explosions of hazardous substances in storage facilities. Such events are not expected to have any impact on the reactor due to the existing protective area separating the reactor site from industrial facilities and residential areas. There are no explosive sources on the reactor site (cylinders, tanks with fuel and explosive substances, explosives, etc.). There are no production facilities using explosive substances in the immediate vicinity of the reactor site, and there are no gas pipelines or other product pipelines. Thus, there are no potential sources of explosion that could cause a blast wave within a radius of 10 km.

4.5. Cases due to internal impact

Unforeseen detection of a strong radiation field in the working areas. This event may be caused by the undetected radioactive substances in internal areas of the primary circuit equipment, in cracks, or in hard-to-reach places. This might be especially relevant during the dismantling of pipes or the drainage system (including the built-in external pipe). The potential consequence would be the increased direct exposure of staff.

Loss of the external power. External power is used to operate tools, cranes, lighting, and air filtration equipment during decommissioning. A loss of external power may result in the following: a) tools, lighting, and air filtration equipment are power-outaged; and b) cranes are off power. The loss of power on tools and lighting will interrupt work activities but will not result in a release of radioactivity. However, the loss of power to the plant’s ventilation and filtration systems may disrupt airflow paths and compromise the effective use of filters. In the event of a power failure outside the reactor site, all work activities with potential airborne radioactive contamination will be suspended. Safety regulations require that lifting cranes be equipped with an automatic parking brake that prevents movement when power is switched off. Therefore, a loss of power is not expected to result in a crane or hoist failure.

Failure of cutting equipment. This event can occur due to mechanical failures, power loss, or loss of control. The consequences include overexposure of personnel who intervene to eliminate the incident's effects, as not all actions can be performed remotely. All necessary measures must be designed accordingly.

Malfunction or failure of local dust suppression/ventilation systems. This can occur inside the cutter housing or during digging, opening, or demolition of walls. Such an event can be caused by a loss of power, clogged filters, fan blockages, or pipe blockages. The consequences are: uncontrolled spread of radioactive dust in the work area; exposure from potential inhalation of radioactive dust; and overloading of filters installed on the reactor building’s ventilation system outlet, which will lead to filter clogging.

Improper handling of local exhaust ventilation filter systems. This can result in overexposure of staff through direct exposure and inhalation.

Falling of dismantled components/waste contai-

ners in the sorting and filling area. The falling of heavy components is one of the most common hazards, creating a risk of structural damage, airborne release, and injury to staff. A heavy object's fall can damage building structures and existing utility systems. Possible consequences of such an event will include: spread of contamination in the work area; possible damage or failure of working equipment; direct exposure of staff; possible skin contact with contaminated parts; and inhalation of radioactive particles. If contamination was formed inside a vessel or pipe during operation, it is very likely that contamination is firmly fixed to the surface. In this case, if the vessel or pipe falls, the fraction of activity released should be small. Data on releases from contaminated surfaces for elements that do not break when dropped is very limited. Fraction up to 10^{-3} assumes an explosive force parallel to the surface. Since the force from a drop at the maximum possible height will be much less, it is reasonable to assume that the maximum fraction for this case will be 10^{-4}. Furthermore, not all activity has a form of aerosols within the component. For example, a heat exchanger has internal baffles to ensure maximum heat transfer during operation. Aerosol particles within the heat exchanger settle on the baffles before being ejected. Therefore, if the design is known, only contamination at open openings should be considered for emissions. Even for open pipes, most of the aerosols formed settle inside the pipe before being ejected.

Malfunction or failure of lifting or handling equipment/tools. This can occur due to mechanical failure, loss of control, or loss of power. The consequences will be similar to those for the events analyzed previously.

Loss of equipment performance (loss of compressed air supply, fire supply water, or electrical power). Such a precursor may be an initiating event for other incidents described earlier (e.g., cutting equipment failure, loss of control, and possible incidents when using lifting equipment).

Failure of personnel protection equipment. Such an incident may lead to an increase in the consequences of the previously mentioned event if they occur simultaneously.

Damage or breakdown of the dosimetry measuring instruments and devices. This may result in high radioactivity spread undetected during work or an unauthorized release of radioactivity in the work area or into the environment.

Staff errors. These include the following events: a) incorrect actions when operating systems, equipment, and personal or collective protective equipment; b) failure to comply with (or ignoring) health and safety rules; c) unintentional actions; and d) intentional actions.

Fire in the dismantling area. To determine the worst possible consequences of an accident during decommissioning, it was assumed that a large contaminated area in the reactor hall or primary pump room was engulfed in fire. Most of the materials in these rooms are non-combustible (metals, concrete, etc.). After the final reactor shutdown, all combustible materials required for operation are removed from the reactor building, further reducing the potential for fire. The likelihood of a major fire is therefore considered low. Since not all rooms contain any combustible materials, the potential source of fire emissions would result from the cutting and material handling equipment. If electric Brokk is used, the fire would be limited to oil and other hydraulic fluids (if flammable). This is considered a worst-case scenario and is discussed in detail below.

5. Analysis of the consequences of the initial events

5.1. Pathways of negative impact on staff

In the event of an accident, personnel can be exposed to radiation through external and internal pathways. External exposure can be caused by the direct radiation from radionuclides in the air (irradiation from the plume) and from radionuclides that have settled on surfaces, clothing, or skin. Internal exposure could be a result of the intake of radioactive material through the respiratory tract, oral ingestion, or open wounds.

The total effective dose, accounting for all major exposure pathways during an accident, is calculated as follows [21]:

$$E_{tot} = E_{ext} + E_{inh} + E_{ing}, \qquad (1)$$

where E_{tot} - total effective dose; E_{ext} - effective dose from external exposure; E_{inh} - expected effective dose from inhalation; E_{ing} - expected effective dose from oral intake. External and internal exposures can be calculated using the equations described below.

External exposure:

$$E_{ext} = \Sigma C_i(x) \cdot D_{ext,i} \cdot t, \qquad (2)$$

where $C_i(x)$ - average concentration of radionuclide at point x (Bq/m^3); $D_{ext,i}$ - external dose coefficient (Sv·s^{-1}·Bq^{-1}·m^3); t - exposure time (h).

Internal exposure from inhalation:

$$E_{inh} = \Sigma C_i(x) \cdot D_{inh,i} \cdot I_{inh} \cdot t, \qquad (3)$$

where $C_i(x)$ - average concentration of radionuclide at point x (Bq/m^3); $D_{inh,i}$ - inhalation dose coefficient (Sv/Bq); I_{inh} - respiratory rate (m^3/h), the recommended standard is 1.2 m^3/h; t - exposure time (h).

Internal exposure from oral intake:

$$E_{ing} = \Sigma M_i \cdot A_i \cdot D_{ing,i}, \quad (4)$$

where M_i - radionuclide intake (kg); A_i - radionuclide specific activity (Bq/kg); $D_{ing,i}$ - inhalation dose coefficient (Sv/Bq).

Table 3 provides an example of numerical values of dose coefficients of the main radionuclides [22].

Table 3. **Dose coefficients**

Radionuclide	Inhalation dose coefficient, Sv/Bq	External dose coefficient, $Sv \cdot s^{-1} \cdot Bq^{-1} \cdot m^3$	Inhalation dose coefficient from oral intake, Sv/Bq
^{60}Co	$9.6 \cdot 10^{-9}$	$1.19 \cdot 10^{-13}$	$3.4 \cdot 10^{-9}$
^{90}Sr	$3.6 \cdot 10^{-8}$	$9.83 \cdot 10^{-17}$	$2.8 \cdot 10^{-8}$
^{137}Cs	$4.6 \cdot 10^{-9}$	$2.55 \cdot 10^{-14}$	$1.3 \cdot 10^{-8}$

The concentration of radioactive aerosols in the air depends on the generation and removal rates from the atmosphere. The removal of long-lived aerosols from the air is due to two processes: a) removal off the air into special and local ventilation systems; and b) deposition on the surfaces of the walls and floor of the room. The rate of radionuclide removal from the atmosphere by special exhaust ventilation is determined by the air exchange rate in the room. In technological rooms, the special ventilation provides ~ 3 air exchanges per hour, which corresponds to a constant removal of aerosols from the air environment of ~ 10^{-4} s^{-1}. Due to ventilation, with a flow rate of 10000 m^3/h, the constant removal of radioactive aerosols, depending on the room's size, will be in the range of $3 \cdot 10^{-3} – 3 \cdot 10^{-2}$ s^{-1}.

5.2. Source term determination

To assess the environmental impact of an accident, the source term, which represents the amount of radioactive material released to the environment, must first be identified. Analysis of current practices for estimating release parameters shows that, aside from limited empirical data on aerosol release, there are currently no consistent approaches. This is due to the fact that only a small number of studies on this topic exist, and, accordingly, a lack of measurement data at nuclear sites during practical work, the great variability of conditions under which they are performed, and the design and technological features of each facility.

Therefore, an empirical approach using a five-factor formula was applied to estimate the emission source [23, 24]. The range of values used for each component is taken from [25] and is given in brackets:

$$ST = MAR \cdot DR \cdot ARF \cdot RF \cdot LPF, \quad (5)$$

where *ST* (*source term*) - the value of the emission; *MAR* (*material at risk*) - the maximum or expected amount of radioactive material that can be involved in a given process; *DR* (*damage ratio*) - the fraction of *MAR* actually exposed to structural destruction operations. It is estimated based on the response of the structural materials to the type and level of impact (0.1–0.9); *ARF* (*airborne release fraction*) - the fraction of *DR* released into the air in the form of aerosols available for further transport ($6 \cdot 10^{-6} – 3 \cdot 10^{-3}$); *LPF* (*leak path factor*) - part of *ARF* (radionuclide aerosols) that have passed through deposition systems (air ducts, production facilities), emission reduction (water spray, pollution traps), capture and filtration and have left the boundaries of process rooms, tanks and protective shells (0.1); *RF* (*respirable fraction*) - a part of radioactive material in the form of particles in the air that is carried by air and inhaled by a person(0.05–0.8).

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

$$EF = DR \cdot ARF \cdot RF \cdot LPF. \quad (6)$$

6. Assessment of the radiological consequences of the fire

Quantitative fire analysis could be ranged from relatively simple to very complex calculations depending on the fire hazard, the design and characteristics of the facility, and a variety of other considerations. More complex fire analyses involve the use of fire models, such as computer codes or manual calculations for scaling and/or simulation. For simple assessments, the formalism presented in Section 5 is sufficient.

To determine the worst possible consequences of the accident, it was assumed that the source of the release as a result of the fire would be the ignition of fuel and lubricants due to leakage from the equipment. The following scenario was considered - as a result of fuel leakage, a fire occurred on an area of 2 m^2, within 1 minute, three persons began to extinguish the fire and carry it out for 2 minutes. It is assumed that the activity of surface contamination is 100 kBq/m^2 with the isotope ratio of ^{137}Cs, ^{90}Sr and ^{60}Co as 2:1:1.

It is assumed that the surface contamination transitions into a gas-aerosol emission due to thermal heating, originating from the entire fire area. An $EF = 1$ was adopted, which corresponds to the most conservative assumption. The emission occurs in the reactor hall, which has a volume of 9120 m^3 (21.5 × × 26.5 × 16.0 m). It is also assumed that the ventilation system is not operational. Exposure of the emergency staff occurs due to external exposure and internal exposure from inhalation. The exposure route from oral intake was not considered because the consumption of food or water is impossible during this time. The calculation results are presented in Table 4.

Table 4. **Calculation results for the staff exposure**

Value	^{137}Cs	^{90}Sr	^{60}Co	Sum
$C_i(x)$ - average radionuclide concentration, Bq/m^3	11	5.5	5.5	22
$E_{ext,i}$ - external exposure, μSv	$3.4 \cdot 10^{-5}$	$6.5 \cdot 10^{-8}$	$7.9 \cdot 10^{-5}$	$1.1 \cdot 10^{-4}$
$E_{inh,i}$ - internal exposure from inhalation, μSv	$1.0 \cdot 10^{-3}$	$6.5 \cdot 10^{-7}$	$1.8 \cdot 10^{-3}$	$2.8 \cdot 10^{-3}$
E_{tot} - total exposure dose, μSv	$1.03 \cdot 10^{-3}$	$7.15 \cdot 10^{-7}$	$1.88 \cdot 10^{-3}$	$2.9 \cdot 10^{-3}$

The dose limit for the total external and internal exposure for staff is set at 20 mSv/yr. During decommissioning, it is advisable to set a limit based on a factor of 2 to determine the working time in areas with increased dose rates, resulting in a dose limit of 10 mSv/yr for staff (this corresponds to a daily dose limit of 35 μSv). As can be seen from Table 4, the total dose to staff during firefighting is very low.

It is further assumed that when the ventilation is not operational, the entire volume of radioactive materials (200 kBq) from the room is released into the environment, bypassing the filter system. For the WWR-M reactor, the regulation and control of population exposure in its surveillance zone are carried out based on calculations of the annual equivalent doses of exposure to critical population groups, obtained due to its gas-aerosol emission. The results of the calculations are reflected in the document [26] and have been approved by the regulatory body. Document sets limits on the total amount of radioactive substances that the reactor is allowed to release into the environment, ensuring that any resulting exposure to humans is small and well below the legal limits.

Table 5. **Authorized gaseous-aerosol releases**

Radionuclide	Authorized gaseous-aerosol releases, Bq/yr
^{137}Cs	$2.70 \cdot 10^{9}$
^{60}Co	$2.80 \cdot 10^{9}$
^{90}Zr	$4.00 \cdot 10^{7}$

The numerical values of the permissible releases of some radionuclides are given in Table 5. The calculated emergency release is significantly less than the permissible values and the actual releases during reactor operation, which do not exceed 1% of the permissible ones.

7. Countermeasures to prevent emergencies and accidents

Fire safety during decommissioning will be ensured through a standard fire suppression system, as well as organizational and technical measures aimed at preventing fires, mitigating negative consequences, and creating conditions for rapid and effective fire extinguishing.

Key areas of fire safety during decommissioning include:

– maximal removal of fire-hazardous materials;
– maximal removal of ignition sources;
– maximal use of non-combustible and fire-resistant materials;
– improvement of the fire protection system;
– timely modernization of the fire alarm system.

Emergency prevention measures for operations with radioactive materials are based on the following:

– high skill and training of staff;
– serviceability of all equipment and technological systems;
– compliance with safety precautions and protection of personnel from radioactive exposure.

The existing fire safety system [20] includes an automatic fire alarm system and fire-extinguishing equipment (e.g., fire hydrants, fire extinguishers, and sandboxes). During the work performance, individual elements of the existing fire protection system will be sequentially dismantled, so installation of additional new elements may be required. The quantity of the main types of fire equipment are regulated by the requirements of the state standards, building codes, and relevant regulatory acts.

In the event of a fire, the reactor's systems:

– the automatic fire alarm system would enable staff to quickly locate the fire;
– the quantity of fire-extinguishing equipment would be sufficient to suppress the fire at its initial stage.

The general direction of accident countermeasures is the following:

– detection and notification of the accident;
– limitation of the spread of the accident;
– elimination of the accident;
– restoration of the operational capacity of technological equipment.

Decisions on taking appropriate protective measures in emergency situations must be made rapidly under changing and unpredictable conditions, often with partial or incomplete information on contamination levels and associated doses. A comprehensive list of organizational, technical, and protective measures is outlined for the decommissioning period to prevent accidents, minimize their consequences, and ensure the protection of personnel, the public, and the environment.

8. Conclusion

During the assessment of accident scenarios, it has been found that for most accidents, the harmful impact would be exerted only on the staff located directly in the accident zone. A fire in the reactor building is considered the most problematic event, though it is assessed as unlikely, short-term, and localized. Given that the objects during a potential accident do not contain a significant amount of radioactive substances, the radiation doses to staff will not exceed the permissible levels. The radiation exposure assessment for potential incidents during the decommissioning stage shows that dose loads to staff and the public are significantly below the limit values established by the regulatory documents, and the overall impact is assessed as insignificant. Industrial risks associated with the work at the site will be minimized through the application of organizational and technical measures designed to prevent and/or mitigate potential consequences for staff and the environment. All identified events are classified as not exceeding Level 1 on the INES scale.

The technical condition of the reactor systems allows their continued operation, and, therefore, it is appropriate to use them during the decommissioning without any reconstructions or modifications. The list of organizational, technical, and protective measures provided for in the operational and technical documentation is sufficient to prevent accidents, minimize their consequences, and ensure the protection of staff, the population, and the environment.

In summary, the analysis of the reactor site's location, the chosen decommissioning option, and the planned dismantling procedures confirms a low probability of radiation hazard to the public and the environment during these activities.

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ОЦІНКА РАДІОЛОГІЧНОГО ВПЛИВУ МОЖЛИВИХ АВАРІЙ ПРИ ЗНЯТТІ З ЕКСПЛУАТАЦІЇ ДОСЛІДНИЦЬКОГО РЕАКТОРА ВВР-М

Дослідницький реактор ВВР-М планується до зняття з експлуатації. Необхідною складовою звіту з аналізу безпеки є аналіз і оцінка потенційних аварій при виконанні зняття з експлуатації. У цій статті представлено підхід до визначення небезпек і можливих радіологічних наслідків найнебезпечнішої вихідної події. Результати оцінки показують, що радіологічні наслідки будуть значно нижчими за встановлені норми.

Ключові слова: реактор типу ВВР, зняття з експлуатації, аварія, доза опромінення, викиди.

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