RADIATION PROTECTION TASKS ON THE KIEV RESEARCH REACTOR WWR-M

by

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Both the description of and the operational experience with the radiation protection system at the research reactor WWR-M are presented. The list of the factors regarding the radiation hazards during the reactor routine operation is given and the main activities on the radiation safety provision are established. The statistical information for the staff exposure, the radioactive aerosol releases and the external radiation monitoring is shown. The preliminary considerations on the system upgrading for the decommissioning are presented.

Key words: research reactor, radiation protection, staff exposure, radioactive release, radiation monitoring

INTRODUCTION

The research reactor WWR-M of the Institute for Nuclear Research of the National Academy of Sciences of Ukraine (INR NASU) is one of the first research reactors constructed and commissioned in the former USSR. The main aim of using this reactor is the generation of neutron beams for research in different areas of physics and engineering. From the first years, the WWR-M research reactor has established the scientific and technical basis for research not only for scientists of the NASU but also of other organizations in Ukraine and the former USSR.

The WWR-M reactor is a heterogeneous water-moderated pool type research reactor operating with thermal neutrons at the power of 10 MW_{th}, giving the maximum neutron flux of $\sim 1.0\ 10^{14}$ $cm^{-2}s^{-1}$ at the core center. The reactor has 9 horizontal experimental channels, a thermal column, and 13 vertical isotope channels in the beryllium reflector. It is possible to install 10-12 vertical channels in the core [1].

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The reactor was commissioned on February 12, 1960 and used until 1993 mainly to study the radiation properties for the variety of reactor materials. During this period it was operated about 100 hours per week to give an annual total of 3000-4000 hours. The reactor was shutdown in 1993; the core was entirely off-loaded to the spent nuclear fuel (SNF) wet storage facility and the reactor did not work until May 1998. From May 1998 to the end of 2001 the reactor was operated according to the interim permit issued by a regulatory body. From May 2001 INR has the permanent license for the reactor operation.

The research reactor WWR-M is located at the site of the INR of NASU in the Goloseev district of Kiev. There are the isochronous cyclotron U-240 and electrostatic generator EG-10 on the INR site.

The WWR-M reactor has been in operation for more than 49 years; at the same time the reactor technical condition allows its further safe operation on condition of upgrading of some systems and elements. The basic objective of the reactor modernization implies future utilization complying with the nuclear and radiation safety requirements.

The NASU has approved "Strategic plan for the use of research reactor WWR-M of the INR" in July 2004 [2]. This multi-purpose strategic plan for the use of the reactor is directed to the effective use of logically defined and analyzed actions on the nuclear installation. The main goal is the coordinated work between the operator, researchers and users from different organizations, the determination of the user's needs and installation capabilities, and the provision of the reactor firm operation by means of the stepwise implementation of the planned strategic tasks.

The radiation protection system is a substantial component of such reactor modernization. Recently, the IAEA Technical Co-operation Project UKR/9/024 "Modernization and safety improvement of research reactor" has been completed successfully. In the framework of this project the reactor was equipped by modern devices and equipment allowing the technical and organizational enhancement for the radiation protection system [3]. The upgraded radiation protection system along with other safety important systems meets all the regulatory requirements for the regular reactor operation and operative decision-making in the case of radiation incidents as well as the forecast of possible emergency situations. This paper presents the description of the radiation protection system which is in operation on the WWR-M research reactor.

RADIATION PROTECTION SYSTEM AND TYPES OF MEASUREMENTS

The operation of nuclear installations in Ukraine is carried out in accordance with the acting legislation. The Norms of Radiation Safety of Ukraine (NRBU-97) [4] is the main state document that establishes the system of radiation and hygienical regulations to provide for acceptable exposure levels for both individuals and public. The norms establish the following categories of persons being exposed: *Category A (staff)* includes those individuals who handle directly, permanently or temporarily, sources of ionizing radiation, *Category B (staff)* includes individuals who are not directly dealing with sources of ionizing radiation but due to the location of their working places within the premises and on sites of the engineering facilities where radiation and nuclear technologies are available could get additional exposure, and *Category C* includes all the public. The exposure dose limits are 20, 2 and 1 mSv per year for the category A, B, and C, respectively.

Main radiation factors at the reactor regular operation are the external gamma-, beta-, and neutron irradiation; the radionuclide internal entry is possible during the repair works and accident after-effects elimination. The reactor is safe if at the regular operation and design-base accidents the non-exceeding of the established exposure doses and radionuclide content in the environment is provided for the staff and population by means of the technical and organizational measures [5, 6].

The radiation danger from the WWR-M reactor regular operation is determined by the following factors:

– *external gamma- and beta-irradiation* of different energies arising from the nuclear fuel and fission products, the induced activity of the coolant, reactor constructions and units, corrosion products, the activated materials and samples,
– *neutrons of different energies* (from fast to thermal) from the reactor core; the neutron impact is possible in the reactor hall around the experimental channels,
– *radioactive aerosols* arising from the fission-fragments and induced activity,
– *noble gases*: ^{41}Ar arising from the irradiation of ^{40}Ar in air by neutrons and the isotopes of Xe and Kr from the reactor core and primary circuit,
– *radioactive* ^{131}I from the fission-fragments,
– *radioactive contamination* of the working areas, equipment and overalls,
– possible penetration of the *activation products* from the coolant and moderator into the air within working areas, and
– *solid and liquid radioactive waste*.

The basic principle of the safety provision is the optimal combination of four activity directions: juridical, organizational, personnel, and technical. The juridical direction includes the development and improvement of the nuclear legislation, where the safety plays the leading role. The personnel direction foresees the measures for the utilization of the skilled and trained staff. The organizational direction consists of the mandatory compliance of the acting safety norms, rules, standards, and regulations as well as the complex of measures concerning the relevant actions at the accidents. The technical direction covers the installation and maintenance of both the radiation protection and the control systems [7, 8].

The basic directions of activity on the staff radiation protection are established by the normative document "Program of the staff radiation protection at the operation of the WWR-M reactor" [9]. Thus, the radiation protection system and the radiation control system are active. Both these systems include:

– biological shielding against both neutrons and ionizing radiation from the core, spent fuel storage and primary circuit,
– radiation control in the premises of the controlled and free access areas, and
– complex of organizational measures directed to the exposure dose reduction.

The radiation control system provides the monitoring of following parameters:

– condition of protective barriers,
– radioactivity of the coolant and technological media (water, air, equipment),
– radionuclide content in the organisms,
– individual dose of external exposure,
– gamma-ray dose rate in the premises, and
– radiation conditions in the supervised area (including the radiation conditions forecast).

The radiation control is performed by means of stationary and portable devices. The radiation dosimetric control in the reactor premises provides the following measurements:

– individual dose of the external exposure in the controlled area and the free access area,
– gamma-ray dose rate in the controlled area by means of stationary devices,
– gamma-ray dose rate in the free access area and on the reactor site by means of portable devices,
– neutron dose rate by means of stationary and portable devices,
– activity concentration of the noble gases, beta- and alpha-aerosols in the premises of the controlled area by means of stationary devices,
– beta-contamination level (controlled and free access areas, equipment, transport) by means of stationary and portable devices as well as by sampling,
– contamination of the working clothes and body surface at the working places and the sanitary check-point by means of stationary and portable devices, and
– activity concentration and radionuclide activity in the reactor releases by means of stationary devices.

The radiation technological control is designed for the measurements of:

– gamma-ray dose rate in the non-attended premises of the controlled area (the reactor hall, the pump-house of primary circuit, the ion-exchange filters, the heat-exchangers, *etc.*),
– activity concentration and radionuclide activity in the air above the reactor,
– activity concentration in the coolant aimed at the control of fuel assemblies tightness, and
– beta-activity concentration in the water of the secondary circuit.

The contamination of the premises in the free access area by radionuclides and aerosols above the established levels was found during the long-term reactor operation. The unplanned contamination of premises in the controlled area was occurring mainly due to the wrong actions of the staff (about 95%). The doses of external irradiation do not exceed the established control levels.

During 49 years of the reactor operation, the incident situations with the exceeding of regular operation limits have not occur. The majority of the situations were connected with the automatic unplanned reactor shutdown. Such situations are equal to 85% of all registered accidental situations. About 8% is the malfunction in the equipment operation and 7% was caused by the wrong actions of the staff. Irregular reactor shutdown (the automatic drop of the emergency rods) was caused by following reasons:

– short-term (<1 s) disconnection of the electric power – 22%
– malfunction of the equipment – 58%
– wrong actions of the staff – 14%
– change of the parameter value (above/below) from the established one – 6%

CONTROL OF THE STAFF EXTERNAL EXPOSURE

The control of the individual external exposure dose during the radiation-dangerous operations is carried out by means of individual dosimeters D-2P and ID-02. The dosimeters ID-11, DKP-50, and ID-02 are used in the case of emergencies. The result acquisition and analysis is carried out by PC code PIDK [3], and then these results are stored in the data-base.

The control levels (CL) of the staff exposure (Category A) for the reactor building were established in accordance with the requirements of NRSU-97 [4], OSPU [10], and the features of the technologies and experience of the experimental and operational works at the reactor as well as on the base of achieved level of radiation safety [11]. The established CLs are presented in tab. 1.

Each CL value is established at the level below the relevant dose limit (DL) and allowable level (AL) for the execution of operational radiation control in the premises of the controlled area and the free access area. In the case of the CL exceeding, the administration and radiation protection service are obliged to perform the investigation with the aim to find out and eliminate the reasons that caused the exceeding.

The statistical information concerning the staff external exposure in the dependence on the duration and number of operations is shown in tab. 2. As one can see from tab. 2, the annual averaged individual dose does not exceed 2.41 mSv (in 1999), which is significantly lower than the established limit (tab. 1). The dynamics of individual doses depend on the character and duration of radiation-dangerous operations (operations in the radiation fields) and can be used as an explanation of the collective dose variation during the considered period. Thus, the main radiation-dangerous operations when the staff has the largest dose load are the following:

– repair, assembling, and dismantling of technological equipment, especially in the pump-house of the primary circuit,
– operations on the reactor cover plate, especially at the core reloading,
– replacement of cleaning resins,
– coolant sampling and analysis,
– collection, conditioning, transportation, and storage of radioactive waste, and
– all kinds of operations with the spent nuclear fuel in the cooling pond.

In accordance with the results of individual dosimetric control during the last decade, the cases of

Table 1.The control levels for the staff exposure

Premise	CL dose rate [μSvh^{-1}]	Neutron flux density [10^5 m^{-2}s^{-1}]			Activity concentration [Bqm^{-3}]			Beta-contamination [cm^{-2} min^{-1}]	Annual dose CL [mSv]
		f	*i*	*t*	α-aerosols	β-aerosols	Noble gases		
Pump-house of primary circuit	200.0	–	–	–	3.7	37.0	–	10^3	18
Reactor cover plate	200.0	–	–	65	2.96	37.0	$1.1\ 10^6$	10^2	18
Semi-serviced rooms	20.0	–	–	–	2.96	37.0	$1.1\ 10^6$	50	
Reactor hall	20.0	10	10	30	2.96	37.0	$1.1\ 10^6$	50	
Storehouse of fresh fuel	4-100	–	–	–					
Storehouse of special materials	4-10	–	–	–					
Free access area	0.7	–	–	–			$1.8\ 10^5$		

f – fast; *i* – intermediate; *t* – thermal neutrons

Table 2. Collective and individual doses of the staff external exposure

Year	Number of operations performed	Number of workers involved	Duration of operations performed [h]		Dose per year	
			Total	Average	Collective [man mSv]	Averaged* [mSv]
1998	269	22	322.8	1.2	68.7	1.27
1999	219	32	635.1	2.9	140.2	2.41
2000	247	41	790.4	3.2	160.5	2.29
2001	262	49	995.6	3.8	168.9	2.31
2002	298	28	476.8	1.6	108.9	1.49
2003	237	31	616.2	2.6	125.0	1.68
2004	211	34	738.5	3.5	152.7	2.06
2005	219	29	613.2	2.8	132.7	1.79
2006	263	37	867.9	3.3	161.7	2.21
2007	184	35	220.8	1.2	89.6	1.31
2008	150	33	255.0	1.7	107.9	1.56

* The dose measurements uncertainty does not exceed 10%

the individual dose exceeding have not been registered during the whole time of the reactor operation. The main criterion of the radiation protection effectiveness is the absence of the individual dose exceeding. Moreover, the additional criteria are the maximal and averaged exposure doses, the decreasing of the collective dose, the decreasing of the individual exposure doses, the decreasing of the radioactive aerosol releases, and the decreasing of the violations.

CONTROL OF THE RADIOACTIVE EFFLUENTS

The noble gases and radioactive iodine isotopes are the main components of the reactor release into the atmosphere. In general, the radioactive noble gases can not be detected in the environmental samples. They give rise to the low doses of the population. ^{85}Kr is the only radionuclide with a half-life longer than a few days (10.7 years). It is detectable in very low concentrations in the atmosphere. ^{41}Ar is produced by the neutron activation of the cooling air. The direct radiation from ^{41}Ar plume results in a significant fraction of the dose which the most exposed members of the public receive. In accordance with the results of the systematical measurements during 1979-2008, the activity of noble gases at the reactor operation was caused by following radionuclides: ^{41}Ar-95%, ^{85}Kr-0.8%, ^{88}Kr-2.5%, and ^{35}Xe-1.7%. The main source of the external exposure among the iodine isotopes is ^{131}I (its half-life is equal to 8.08 days); the releases of iodine isotopes took place continuously.

As a rule, the concentrations of the released radionuclides into the environment are often too low to be measurable. Therefore, the dose estimates for the population have to be based on modeling the atmospheric transport and environmental transfer of the released radioactivity.

The following control levels were established for the releases: the total activity of noble gases should not exceed the value of $1.65 \cdot 10^{14}$ Bq per year (the activity concentration $<3.03 \cdot 10^{6}$ Bq/m^3) and the total activity of ^{131}I should not exceed the value of $5.55 \cdot 10^{10}$ Bq per year (the activity concentration $<4.07 \cdot 10^{-2}$ Bq/m^3) [12]. These release values lead to the exposure dose for the public individual equal to 8.8 and 51.5 μSv per year, respectively. The collected results of the radioactive release are shown in fig. 1.

EXTERNAL RADIATION MONITORING

The systematic radiation control of the rector's impact on the environment is carried out during all the time of the reactor operation. The main task of radiation monitoring is the overall control of gamma-, beta-, and alpha-radioactivity as well as the content of basic radionuclides of reactor's origin (first of all, ^{3}H, ^{90}Sr, and ^{134,137}Cs) in the environmental objects around the reactor's affected zone. The investigations are performed in 6 stationary points within the reactor site area (300 m) and 12 stationary points within the supervised area (3000 m), which have been selected according to the windrose. The subjects of inquiry are the following: the near-surface air, the atmospheric precipitates and settling dust, the water from the main collectors, the water from the open reservoirs (including the water flow of river Dnepr – above and below the rector's location), the water from the melted snow, the birch sap, the soil, and the vegetation. The measurements of the short-lived and long-lived alpha- and beta-aerosol content in the near-surface air were performed too together with the measurements of gamma-radiation dose rates in the control points. The measurements of ^{137}Cs and ^{90}Sr in the soil specimens from the investigation holes on the reactor site were carried out with the aim to detect the soil contamination.

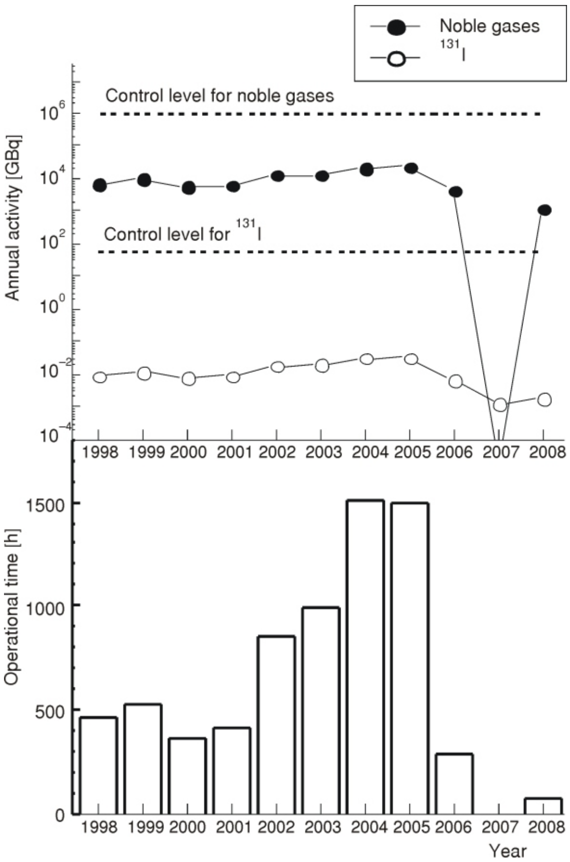

Figure 1. Releases of noble gases and ^{131}I

The control of air radioactive contamination is carried out by means of the sediment and aspiration methods. The sediment method was applied for the determination of total specific beta-activities and ^{90}Sr specific activity. In accordance with the results of many year investigations, the value of the total beta-activity lies in the range from 77 to 208 Bq/m^2 per year, for ^{90}Sr – from 1 to 14 Bq/m^2 per year. The conent of short-lived alpha- and beta-aerosols in the air was measured by means of aspiration method (twice per week); this content was from 4 10^{-5} to $1.7 \cdot 10^{-3}$ Bq/l and from $4 \cdot 10^{-4}$ to $6 \cdot 10^{-3}$ Bq/l for the alpha- and beta-aerosols, respectively.

The control of water radioactive contamination is carried out by sampling from the main institute's collectors and open reservoirs. The contents of beta-radionuclides and tritium were significantly lower than the established permissible concentrations.

The control of soil radioactive contamination is carried out annually by sampling in the stationary control points. The results are the same as for the typical values in Kiev and formed by the radionuclides of the Chernobyl accident origin. Additionally, the content of ^{137}Cs and ^{90}Sr in the soil specimens from the investigation holes is measured twice per year. The measured value for ^{137}Cs is in the range from <0.4 to 0.9(2) Bq/kg, for ^{90}Sr – from 0.4(1) to 0.9(2) Bq/kg. This is an evidence of the absence of the man-caused influence on the soil contamination arising from the reactor operation as well as of the integrity of the liquid radwaste tanks on the reactor site.

The control of vegetation radioactive contamination is carried out by sampling; the measured values for the specific beta- and ^{90}Sr activities do not exceed the typical ones for Kiev.

The control of gamma-radiation dose is carried out every day at the control point on the reactor site; the measured value lies in the range from 0.11 to 0.20 μSv per hour.

As the whole, the results of the radiation monitoring give the evidence that the reliable increase of radionuclide content within the controlled parameters in comparison with the Kiev's typical ones was not found during the whole time of investigations and this confirms the safety of the reactor. The reactor radiation impact on the environmental objects is very small and it is difficult to distinguish between the natural background and the man-caused contamination caused by the Chernobyl accident and global fallout. In the future, at the reactor decommissioning, the available radiation monitoring system will be preserved and adapted for the new tasks.

FORTHCOMING TASKS

The present technical condition of the reactor allows its safe operation for no less than 8-10 years on condition of upgrading of some systems and elements. At the same time in accordance with the acting legislation the decommissioning of the reactor must be considered by the operator as early as possible [13]. The initial decommissioning planning is in progress now [14]. Deducting from the planes of the further site use with removing the spent fuel and radwaste outside Kiev and returning the site its unrestricted use, the decommissioning program for the WWR-M reactor foresees the immediate dismantling as an optimal decommissioning strategy. The goal of the decommissioning program is the determination of the main organizational and technical measures directed to the preparation for the decommissioning and the implementation of the decommissioning of the WWR-M reactor. In accordance with the selected decommissioning strategy the sequence of the decommissioning stages, the content of works and measures at these stages, their durations as well as the necessary conditions and infrastructure for the timely, safe, and effective decommissioning execution were established. In accordance with the preliminary estimations, the decommissioning timeframe will not exceed 6 years.

The setup of the radiation protection system at the reactor decommissioning will be a logical continuation of the currently existing system. This system will be rearranged and adopted for the needs resulting from the nature and content of decommissioning works. It will be necessary to implement specific surveillance and monitoring programs, including the appropriate standards and separate measuring procedures. At the same time, the established approach for the staff exposure will be retained, namely:

– staff and population exposures cannot exceed the established dose limits, and
– levels of individual exposure and the number of persons subjected to the exposure should be as low as it can be achieved with an allowance for economical and social factors.

The design limit for Group A worker dose is 20 mSv per year, although it is permissible for individual workers to receive up to 50 mSv in a year subject to an overall (50 year) lifetime limit of 1000 mSv. The design intent of the decommissioning operations is that the annual individual dose will not exceed 20 mSv and will be as far below 20 mSv as is reasonably achievable (ALARA). There are limits on the maximum dose rates in the areas according to occupancy. The dose rate limits allow a safety factor of 2 and would result in the annual exposures of 10 mSv for Group A workers (this corresponds to the daily dose limit of 70 μSv).

The additional administrative and engineering measures will be implemented for the safety provision for the staff, population, and environment at the decommissioning:

– works will be carried out in the conditions established by the rules of radiation hygiene, namely, the radiation control, protective barriers, sanitary sluices, *etc*. will be provided,
– working premises and areas will be divided into separate zones,
– restriction of staff exposure by means of use of the remote equipment, the optimization of dismantling and cutting procedures, *etc*.,
– secondary radwaste minimization,
– local ventilation and dust suppression will be used together with the available ventilation system,
– additional individual protection tools will be necessary as well as the mobile protective shield and temporary barriers,
– radiological mapping of working areas should be arranged,
– radiation monitoring aimed at the detection of areas with an increased dose rate,
– permanent measurements of contamination,
– improvement of the external monitoring system,
– account and control of the radioactive waste before the removal outside the reactor site, and
– development and modernization of the emergency response plan.

CONCLUSIONS

The WWR-M reactor continues to operate within the existing national frameworks and international recommendations. The reactor is equipped by the radiation control system; the radiation protection procedures are sufficient and very satisfying. The established radiation parameters for the reactor operation do not exceed the normative limits. The technical and organizational measures provide the radiation protection of staff and population at the necessary level. The reactor operation has a negligible influence on the environment and cannot be a reason for any negative ecological change. The elaboration of the planes for the modernization of the radiation control system aimed at the safety provision at the reactor decommissioning is in progress now.

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ЗАДАЦИ ЗАШТИТЕ ОД ЗРАЧЕЊА НА КИЈЕВСКОМ ВВР-М ИСТРАЖИВАЧКОМ РЕАКТОРУ

У раду је приказан систем за заштиту од зрачења и изнето је искуство у управљању њиме на ВВР-М истраживачком реактору. Утврђена је листа фактора који се тичу радијационих ризика током рутинског управљања реактором и установљене су основне активности на успостављању радијационе сигурности. Показани су статистички подаци о излагању особља, ослобађању радиоактивних аеросола и спољашњем радијационом мониторингу. Такође су дата прелиминарна разматрања о унапређењу система декомисије.

Кључне речи: истраживачки реактор, заштита од зрачења, изложеност особља, емисија зрачења, мониторинг зрачења