# RADIATION LEVELS AT CARRYING OUT THE REFURBISHMENT OF THE BULGARIAN RESEARCH REACTOR IRT 2000*

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***Abstract.*** *The paper presents a summary of the main steps in carrying out the refurbishment of the research reactor IRT in Sofia. It was a 2 MW pool type light water cooled and moderated reactor which after the final shutdown was defueled and brought to a state of safe enclosure. According to a decision of the Bulgarian Government it undergoes a reconstruction into a low-power reactor. In this regard during the period 2008 - 2014 within the IRT-Sofia took place a number of activities with potential for radiation hazard such as shipping of the spent nuclear fuel, dismantling of the reactor internals and replacement of aged reactor equipment, categorization, sorting and packaging for temporary storage of dismantled equipment and the radioactive waste generated during the dismantling activities. The report gives information on the organization and management of the activities and outlines some key technical aspects of the dismantling and removal of the contaminated/activated components. Also, the experience gained during this project is highlighted, particularly with a view to methods for the separation of radioactive waste from material to be cleared. The radiation measurements and site monitoring prior, for the period of all the activities and at present supplied comprehensive evidence that the work has been accomplished safely for the personnel and without radiation consequences for environment.*

***Key words*: *research reactor, partial dismantling, RAW categorization, radiation protection***

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## 1. INTRODUCTION

The research reactor IRT in Sofia is a Russian design pool type nuclear reactor with light water used as a moderator, a coolant and a top radiation shield, Fig.1 [1, 2]. The reactor went critical on 18 September 1961. There were several upgrades from the initial 500 kW of thermal power: 1000 kW (1962), 1500 kW (1965) and 2000 kW (IRT-2000, 1970).

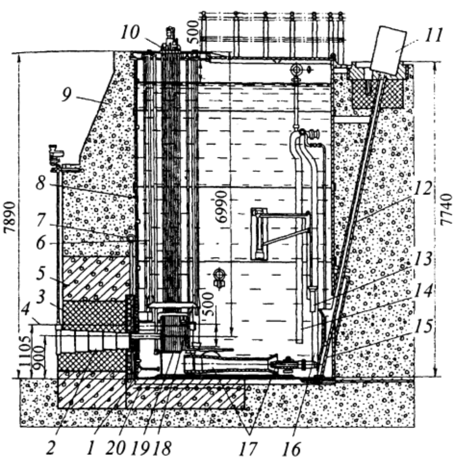

Figure 1. Vertical section of IRT-2000: 1) thermal screen, 6.5 tons/m3; 2) shutter of the horizontal beamtube; 3) shield, 5.1 tons/m3; 4) axis of the horizontal beamtube VII; 5) shield, 4.4 tons/m3; 6) channels for control rods; 7) collector of the exhaust ventilation of the horizontal channels; 8) aluminum tank; 9) concrete; 10) top platform; 11) container for transporting radioactive wastes and fuel elements; 12) pipe for removing radioactive samples and spent fuel elements; 13) turning arm for transporting samples inside the pool; 14) intake line of the cooling loop; 15) bucket for removing radioactive wastes and fuel elements; 16) delivery pipe for the cooling loop; 17) pipe and nozzle of the ejector (water-jet pump); 18) coils for cooling the shielding; 19) core; 20) repelling membrane.

The reactor IRT-2000 has been in operation until 13 July 1989. Apart from production of radioisotopes for industrial and medical needs as well as for training and education the reactor was used for research in nuclear and solid state physics, material science, radiation biology and radiochemistry [2]. It was shut down for reconstruction in order to conform with the increased requirements for nuclear and radiation safety after the Chernobyl accident.

The simplicity of the non-pressurized pool-type reactor system provides itself to inherent safety. Passive structures and the water pool are the main barriers granting protection for the workers and the public from accidents involving nuclear fuel failures. Fission product source term buildup is low, due to the low power operation of the reactor. There were no major modifications, no significant incidents or accidents and no events with a hazardous impact on the staff, public or environment and the reactor has been shut down under normal circumstances.

In accordance with a Resolution of the Bulgarian Government the reactor IRT-2000 undergoes a reconstruction into a low-power reactor. In this regard during the period 2008 - 2014 within the reactor took

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place a number of actions with potential for radiation hazard such as shipping of the spent nuclear fuel, dismantling of the reactor internals and replacement of aged reactor equipment, categorization, sorting and packaging for temporary storage of dismantled equipment and of the radioactive waste (RAW) generated during the dismantling activities.

Radiation safety in all planed actions is secured in the General Program (GP) developed to comprise a detailed set of technical and organizational resources and activities in accord with the ALARA principle. Using a comprehensive approach, the experts of the Nuclear Scientific Experimental and Educational Centre (NSEEC) at the Institute for Nuclear Research and Nuclear Energy (INRNE) evaluated technical and organizational radiation protection measures for all the personnel employed in areas exposed to radiation. The GP and the Detailed Plan for Partial Dismantling (DPPD) of the reactor were approved by the Bulgarian Nuclear Regulator.

The article reports on the organization and management of the activities carried out so far and outline some key technical aspects of the dismantling and removal of contaminated/activated components.

## 2. Refurbishment Activities

The partial dismantling of acknowledged as obsolete systems was an important stage in the reconstruction of the reactor. Guiding principles in implementing the DPPD were protection of human health and environment, protection outside the national borders as well as control over the amount and radioactivity of the generated RAW. The shipment of spent nuclear fuel (SNF) was carried out prior to any forthcoming activities, because the SNF storage, reactor pool, reactor internals and supporting equipment are issues of refurbishment.

During facility modifications regular radiation and contamination surveys of modification areas and equipment before and after every activity were performed. The radiation monitoring comprised the personal dosimetry control, control over the radiation situation in the premises, control of the air and the filters of the operating permanent and additional (local) ventilation systems, control of radioactive waste water and sewage, control over the collection, sorting, categorization, storage and transportation of solid and liquid RAW. Radiological characterization of the reactor facilities was made before the start of partial dismantling operations, during the activities and at the end. Assessment of the radiological situation was carried out through the methods of direct measuring, taking samples and smears from the materials of the facilities [3] that were subject to dismantling and through computational methods. The results served as an early basis for determination of the type of the technique used for dismantling, instruments and devices (remote, semi-remote or manual methods), the necessity of decontamination and the choice of appropriate methods, measures for radiation protection of employees and preservation of the environment, the RAW classification, the requirements for treatment, transportation and storage of the waste during the dismantling, the price of the dismantling, dose exposure, risk assessment and the kind of the necessary protective means for the employees at carrying out dismantling activities.

The monitoring of nuclear site environment and surrounding laboratories was performed at a grid of pre-selected observation posts. The evaluation was based on taken samples and determination of radionuclide contents in air (aerosols), water (groundwater and rainfall), soils and in selected plants – bioindicators [4, 5]. Particular detector systems were placed at selected locations of reactor workplace environment. The radiation monitoring system was in continuous operation. The exceeding of warning or emergency level was indicated by acoustic and optical signals. Real time information about gamma dose level can be displayed on the INRNE website.

### *2.1. Spent nuclear fuel loading and transportation*

SNF shipment planning began in October 2004 and the actual shipment was completed in July 2008, requiring about 45 months of activities. All 16 HEU (initial enrichment: 36% U-235) spent nuclear fuel assemblies (SNFAs) of the S-36 type and 58 LEU (initial enrichment: 10% U-235) SNFAs of the EK-10 type were shipped to Russian Federation with assistance from the Russian Research Reactor Fuel Return Program (RRRFR), one of several programs of the U.S. Department of Energy National/Nuclear Security Administration's (DOE/NNSA) Global Threat Reduction Initiative (GTRI).

A special Radiation Monitoring Procedure was developed to ensure safe execution of the preparation and transportation of the SNFAs, as well as for meeting the requirements for permitted levels of radioactive contamination on the surface of the transport casks. The procedure covered the radiation survey of systems and equipment in the reactor hall and loading spots, both before and after the required modernization.

There were some important aspects in terms of radiation levels during the preparation of SNF for transportation. Firstly, the radiation protection of the personnel directly engaged in the activities required monitoring of radioactive surface contamination of the personal (particles.$cm^{-2}.min^{-1}$), equivalent dose rate limits of whole body outer irradiation Hp [6] and reference to basic limits of radiation protection. Some of the major issues that deserved special attention concerning the radiation protection of the staff were:

- Facility modifications and equipment. After the choice of using VPVR/M casks was taken, several facility modifications were identified as necessary to allow their use in the reactor hall, including:
  - Replacement of the reactor hall crane with a higher capacity 12.5-ton bridge crane;
  - Fabrication of a support platform to hold the VPVR/M cask above the reactor pool;
  - Installation of an underwater camera and lights to assist with basket loading;
  - Modification of the ventilation air-ducts above the reactor pool to exhaust potentially

contaminated air to an existing filtered ventilation system

- Spent fuel inspections. To assure compliance with Russian requirements and acceptance at Mayak reprocessing facility, a spent fuel inspection has been performed. The records of SNF storage water control proved absence of any failed fuel elements.
- Cask loading. It was executed by the INRNE staff and IAEA and Euratom inspectors witnessed all cask loading and applied tamper indicating seals on each cask.

The second aspect was meeting the requirements in terms of allowable levels of radioactive contamination and equivalent dose rate of the casks, transportation packages outer surface and dismantled equipment from reactor hall during facility modifications.

The equivalent dose rate at any point of the Package and transport outer surface must not exceed the value 2 mSv.h$^{-1}$, and at any point at a distance of 2 m from the transport outer surface - 0.1 mSv.h$^{-1}$. Radioactive contamination of the Package and transport surfaces must not exceed the values given in Table 1 (according to SanPin-2.6.1.1281-03 and adaptation of norms OST-95 to NP 053-04, TS-R-1). The limits were used at averaging of any part of the surface at any section with the area of 300 cm$^2$.

Table 1. Allowable levels of radioactive contamination of loaded transport and Packages (particles.min$^{-1}$.cm$^{-2}$)

| Object | Non-fixed contamination | | Fixed contamination | |
|---|---|---|---|---|
| | β- | α- | β – | α - |
| Transport outer surface | 10 | 1 | 100 | 10 |
| Transport inner surface | 100 | 10 | Not regulated | |
| Transport cask outer surface | 100 | 10 | | |

The measurements of the radiation levels could be grouped with respect to the performed activities:

- Radiation levels during preparatory activities

Measurements in the reactor hall during old crane dismantling activities showed maximum equivalent gamma dose rate of 185 nSv.h$^{-1}$.

Radiological characterization of dismantled old crane parts was performed and the measurements of collected smears by alpha-beta counting and In-situ measurements are presented in Table 2. The In-situ data were obtained using a handheld device MicroCont-II (Mirion Technologies (RADOS) GmbH) and refer to contamination that has not been moved from its original place of deposition.

The highest radiation levels were measured in the time of SNF inspections. The dose of gamma radiation from one fuel rod (EK-10 type) at a distance of 1 m in the air did not exceed 0.75 mSv.h$^{-1}$, while the highest gamma dose rate from SFA at a distance 1 m., but under water was 3126 mSv.h$^{-1}$.

- Radiation levels during cask loading;

All three casks Skoda VPVR/M were loaded, dried and checked for leak tightness over a period lasting eight days under the close supervision of IAEA, EURATOM and Bulgarian Nuclear Regulatory Agency (NRA) inspectors. The drying and leak testing of the casks was performed by specialists from the Nuclear Research Institute (NRI) in Řež (Czech Republic). Radiation and contamination surveys were performed before, during and after loading of every cask. For the nine days of the job the accumulated dose by each individual was between 0.008 -0.012 mSv except one value of 0.015 mSv.

Table 2. Alpha-beta counting (smears) and in-situ β-activity measurements during facility modifications

| Dismantled crane parts | Total $A_\alpha$, Bq.cm$^{-2}$ | Total $A_\beta$, Bq.cm$^{-2}$ | β-particles, min$^{-1}$.cm$^{-2}$ (in-situ) |
|---|---|---|---|
| beam 1 | 2.2 E-4 | 9.3 E-4 | 10-12 |
| hook | 2.2 E-5 | 9.0 E-4 | 10-12 |
| tracks | 2.5 E-3 | 5.4 E-4 | 10 |
| beam 2 | 6.2 E-5 | 1.1 E-3 | 10-12 |

*2.2. Partial dismantling*

The partial dismantling of reactor facilities followed strictly the DPPD prescriptions, in full accordance with the regulations [6-15].

The DPPD described in detail the sequence of procedures concerning dismantling of the obsolete IRT-2000 equipment found of no use for the low-power reactor and the subsequent activities for reduction of the radioactive waste volume, decontamination, sorting, packaging, temporary storage and transportation for delivery to the State Enterprise “Radioactive waste”.

The dismantling of the IRT-2000 facilities and equipment was executed in two stages. The first stage comprised the dismantling of the equipment inside the reactor pool (RP) and the block with shutters of the thermal column (TC). The second stage included dismantling of the piping and pumps in the primary cooling loop (PCL) room. After the SNF removal it was dismantled the spent fuel rack on the bottom of the SNF storage following safely conducted measurements and characterization of the rack materials.

Two main zones were formed at the time of carrying out the dismantling: a) zone without radioactive contamination – “clean” zone; b) zone in which there was radioactive contamination of the equipment, the work appliances, etc. or contamination was expected - “dirty” zone. These two zones were separated with clearly visible signs and appropriate resources. At each zone boundary there was a sanitary check point supplied with dosimetry equipment for whole body measurement of the surface radioactive contamination. The order of crossing from one zone to another was regulated according to a preliminary developed plan and marking scheme of the movement of the personnel

Five working zones were recognized and delineated in the Reactor hall: Zone 1 (dismantling of RP equipment); Zone 2 (dismantling of PCL equipment); Zone 3 (dismantling and other activities at the reactor site); Zone 4 (dismantling of the TC); Zone 5 (secondary processing and deactivation of the dismantled equipment site).

Real measurement of radiation was essential in evaluating the effectiveness of protection measures and in assessing the radiation dose likely to be received by individuals so that both "installed" (in a fixed position) and portable (hand-held or transportable) measuring instruments for radiation protection were used [16]. Installed instruments were fixed in positions known as important in assessing the general radiation hazard in the site. Examples are the installed "area" radiation monitors, gamma interlock monitors, personnel exit monitors, and airborne particulate monitors.The individual dosimetry control of the personnel was conducted by two types of individual dosimeters: thermo-luminescent detector (TLD) and electronic direct-reading dosimeter DMC 2000S (Merlin Gerin Provence Instruments, France) with integrated semiconductor detector with energy response better than ± 20 % from 60 keV to 1.3 MeV for $^{137}$Cs and measurement range 1 μSv – 10 Sv. Electronic dosimeters were reported wirelessly using the software (DOSIMAS), and the results were logged into the database [17]. All working personnel were provided with TLD and DMC dosimeters, which were reported every day. At the end of the reporting period each worker received written information about his dose load.

### *2.3. Radioactive waste management*

Radionuclides generated during the operation of a nuclear reactor are the source contributing to production of radioactive waste (RAW). By origin, the radionuclides are essentially divided into two groups: products of neutron activation and fission products. *Fission products* are the complex mixture of radionuclides of various chemical elements, including noble gases, which have different physical and chemical properties and different behavior in the fuel and in radioactive environments in the nuclear reactor. *Activation products* are due to neutron irradiation of structural materials (including the corrosion products) and present the main contribution in the RAW formation Accumulation of fission products and activation in the primary coolant in the reactor vessel and internals can be conditionally called primary contamination. All other contaminations of equipment, tools, facilities, clothing etc. are formed as a result of migration and redistribution of radionuclides occurring by different mechanisms - dissolution and crystallization, evaporation and condensation, sorption, diffusion and chemical interactions. The speed of the process and the contamination degree of course depend on the physicochemical properties of the transport medium and the interfaces, as well as on the external conditions (particularly the temperature). These contaminants are termed secondary pollution. From these radionuclides with the largest contribution to the total activity are $^{60}$Co and $^{137}$Cs. The primary contaminants are voluminous because they are distributed throughout the volume of the medium. Secondary contamination can be conditionally divided as volume and surface.

RAW source analysis specified waste streams that are identifiers for objects of generation and temporary storage of radioactive waste. RAW is mainly generated in the reactor hall and the radiochemical laboratory I class (RCL). Table 3 gives some radionuclides expected in the material resulting from the dismantling activities. Waste streams are identified by an identification code. The relationship between the identification code and characterization data is given in Table 4.

Table 3 Preliminary list of radionuclides

| Radio nuclides | T1/2 (years) | Emi tters | Material |
|---|---|---|---|
| Neutron activation products | | | |
| H-3 | 12.3 | β | concrete, stainless steel, graphite |
| C-14 | 5730 | β | concrete, stainless steel, carbon steel, graphite |
| Fe-55 | 2.7 | Ec | concrete, stainless steel, carbon steel, graphite |
| Co-60 | 5.3 | γ, β | concrete, stainless steel, carbon steel, graphite /all materials/ |
| Ni-59 | 7.5 E+4 | Ec | concrete, stainless steel, carbon steel, graphite |
| Ni-63 | 100 | β | concrete, stainless steel, carbon steel, graphite |
| Eu-152 | 13. 4 | Ec, β, γ | concrete, graphite |
| Fission products | | | |
| Cs-137 | 30 | β, γ- | |

*Ec – electron capture

The identification code is a combination of letters and numbers and is formed as follows:

Table 4. Relationship between the identification code and characterization data

| Symbol position | 1 | 2 | 3 | 4 | 5 |
|---|---|---|---|---|---|
| Symbol | A | X | B | AB | XX |

Where: Position 1 represents a letter of the Latin alphabet, which indicates the physical state of the radioactive waste: L – Liquid RAW, S – Solid RAW

Position 2 is a number, which indicates the object associated with the generation of radioactive waste at IRT-2000:

| | |
|---|---|
| 1 | Reactor hall |
| 2 | Reactor vessel |
| 3 | Biological shield around the thermal column |
| 4 | Primary loop |
| 5 | RCL I class |
| 6 | SNF storage |

Position 3 is a letter of the Latin alphabet, associated with the time of RAW generation:

| | |
|---|---|
| C | RAW generated during the IRT-2000 operation |
| D | RAW generated during the partial dismantling |

Position 4 - two letters of the Latin alphabet, which denote objects related to waste management: TL-Liquid RAW storage, ED – RAW storage.

Position 5 - two Arabic numerals specifying the serial number of the flow of 01 to 03:

Example: S1DED01 - means solid waste from the reactor hall received during the partial dismantling. They will be stored in the RAW storage.

Each of the waste streams is characterized in terms of physical, chemical, radiation and etc. properties of RAW. In coverage of the characteristics shall be filled a diary reporting the generated waste. The characterization is performed to assess the possibilities for further processing, storage and / or disposal, and the need to maintain current inventories. RAW inventory and referral to the appropriate categories is based on the results of the measurements, sampling and smears as well as analyzes made by calculation.

The resulting waste was collected, deactivated, pre-prepared, packaged and temporarily stored at the site. For the temporary storage of radioactive waste a RAW storage was made available. After packaging the RAW and putting them into a licensed transport container, RAW will be transported to the State Enterprise "RAW". For the temporary storage of the radioactive and contaminated liquids, a Liquid RAW storage is available on the site. Liquid waste from IRT-2000 is submitted to SE "RAW" for processing and disposal.

### *2.4. Radiation levels during partial dismantling*

For **Zone 1** (Dismantling of the RP equipment) the maximum measured dose rate was 5.64 mSv.h$^{-1}$. Therefore we chose remote dismantling instruments allowing 25 cm distance from the hot spot. At 25 cm the dose rate reduced to 0.35 mSv.h$^{-1}$. Dismantling of equipment of this hot spot was estimated to require 25 hours or a total dose of 8.83 mSv. The total time for pool equipment dismantling was assessed to 100 hours if carried out by two persons. The evaluated collective dose of exposure was 9.49 mSv.

For **Zone 2** (Dismantling of the equipment in PCL) the total work time was assessed to 180 hours. The maximum equivalent dose rate in the PCL was measured 10 cm away from the ion exchange filters and was 0.0024 mSv.h$^{-1}$. The collective dose exposure of the personnel was expected to be 0.864 mSv.

In **Zone 3** the activities were executed while the RP was full of water that permitted lowering of the dose load of workers. The measured equivalent dose rate was in the limits 0.14 - 0.20 μSv.h$^{-1}$ that can be thought of as the equivalent dose rate of natural background. The dose load contribution from natural background radiation was negligible and was not included in the estimates of the total radiation exposure of personnel. The measured equivalent dose rate on the reactor site when the pool was empty was 11 μSv.h$^{-1}$. This allowed a team of two people to work there a whole day assisting with the dismantling activities conducted by the personnel in the pool. The maximum equivalent personnel dose rate for the 100 hours of equipment dismantling was 2.2 mSv.

For **Zone 4** (Dismantling of second horizontal channel - TC) and **Zone 5** (Secondary processing and deactivation of the dismantled equipment site) the obtained doses of the 14 workers were in the interval 0.001 – 0.972 mSv, i.e. below 1 mSv.

The collective equivalent dose during the reactor equipment partial dismantling was assessed to 12 - 15 mSv. Obtained collective doses during the partial dismantling activities were: 2.47 man mSv for reactor internal systems; 123 man μSv for PCL equipment; 80 man μSv for at the reactor site; 2.41 man mSv for the TC; 271 man μSv on the site for secondary processing.

The process of radiological characterization of the IRT-2000 materials was carried out through the methods of measuring, taking samples and smears from the materials of the facilities, which were liable to dismantling. To assess the RP the measurements were carried out in full of water RP, by a hermetic detector and after emptying of the pool - by direct measurements through the protective cabin. A number of samples were taken from the RP equipment - *automatic regulation rod* (3 tube sectors from the channel with 2 cm length, at around 30 cm from the channel bottom); *vertical channel of 1 MR /north/* (3 tube sectors from the channel with 2 cm length at a distance of 30 cm from the channel base); *cover of one of the vertical experimental channels /west/* (2 shaving pieces from the surface); *cover steel hasps* (part of it, around 1 cm); *two preliminary selected graphite assemblies from the east side of the reactor core* (assemblies were withdrawn by means of a manipulator from the reactor core and measurements of their activity were carried out above the water level) etc. Additionally, a graphite block was taken out from inside of the TC. The block was there during the whole period of operation of IRT-2000. Data obtained for selected graphite assemblies and graphite block are presented in Tables 5 and 6.

Table 5. Data from measurements of two of the graphite assemblies

| Assembly 1 | $\mu Sv.h^{-1}$ | Assembly 2 | $\mu Sv.h^{-1}$ |
|---|---|---|---|
| Upper end | 10 | Upper end | 6 |
| Middle | 110 | Middle | 120 |
| Bottom end | 180 | Bottom end | 160 |

Table 6 Data from gamma spectrometric measurements of the graphite assemblies

| Sample | Mass, g | Eu-152, $Bq.kg^{-1}$ | Co-60, $Bq.kg^{-1}$ |
|---|---|---|---|
| Graphite | 233 | 130 ± 11 | 650 ± 60 |

For radiological characterization during the time of dismantling activities were collected about 300 smears (dry or wet) and 100 material samples (solid or liquid) - metal segments from reactor systems, concrete, steel, glass, flakes from the inner surface, graphite of the TC, personal means of protection etc. Measurements were carried out by Gamma spectrometry ($^{137}Cs$, $^{60}Co$, $^{152}Eu$), Liquid Scintillation Counting - LSC ($^{14}C$) and Total alpha-beta counting. As a whole the obtained results were low and showed the lack of surface contamination. Only few contaminated details were detected during the whole operation period. Contaminated details were treated chemically. New samples were collected and analyzed after decontamination treatment.

The specific activities of $^{137}Cs$ and $^{60}Co$ in all studied samples varied between 2 and 60 $Bq.kg^{-1}$ (Table 7). Exception made two areas at the floor of premise 103 (PCL) where the mean measured values were 362 $Bq.kg^{-1}$ for $^{137}Cs$ and 2.62 $kBq.kg^{-1}$ for $^{60}Co$. Contaminated materials were removed and treated as a radioactive waste.

Data obtained for the graphite material collected from the thermal column (obtained by chemical destruction with distillation technique and by burning procedure) show high specific activity of $^{14}C$ – between 55200 and 60600 $Bq.cm^{-2}$.

Table 7. Data for material samples during the partial dismantling activity

| Sample | Cs-137, $Bq.kg^{-1}$ | Co-60, $Bq.kg^{-1}$ |
|---|---|---|
| concrete | 2.6 ± 0.2 | 8.6 ± 0,7 |
| concrete | 20.3 ± 1.8 | 59.5 ± 5.4 |
| walls PCL room | 19.0 ± 1.3 | 20.0 ± 1.6 |
| floor PCL room | 4.5 ± 0.4 | 24.0 ± 1.8 |

Final radiological survey of PCL, RP and reactor hall was carried out according to a final characterization plan. Sampling was done at different points on a sampling grid covering the entire investigated areas (walls, floors, ceilings, equipment surfaces etc.). All surfaces were separated into small fragments marked A, B, C, D etc. and each place was investigated by means of portable dosimetry and radiometry devices. The sampling points were determined after measurements of surface contamination. From the places with measured values higher than the permissible limits material samples and smears were taken. In general, the results showed lack of contamination except to two points of the pool floor. Contaminated materials were removed and classified as RAW.

### *2.4. Radioactive waste temporary storage*

Data obtained from final radiological characterization were used to assess the situation after dismantling activities.

All dismantled equipment and the generated during the dismantling activities solid RAW (mainly metals - steel, aluminum and iron as well as small amounts of graphite, concrete, rubber and plastics) were categorized, sorted, packed and sealed in special reinforced concrete containers for transportation and storage. Categorization was performed in accordance with the regulations for transport of radioactive material and delivery of radioactive waste to the State Enterprise RAW. [6,14,15]

Prior filling the container its inner surface of the walls and bottom were treated with epoxy corrosion protection and waterproofing primer AQUADUR, polyurethane corrosion protection HYPERDESMO and with double adhesive membrane Bitutin. RAW were distributed to containers in accordance with the regulations - dose bellow 2 $mSv.h^{-1}$ at the container surface and 0.1 $mSv.h^{-1}$ at a distance of 2 m from the container.[14] Filled containers were placed on a site for temporary storage specially built for the purposes of the partial dismantling.

## 3. Conclusions

The radiation measurements and site monitoring prior, for the period of all the activities and afterwards gave comprehensive evidence that all the work has been accomplished safely for the personnel and without radiation consequences for environment.

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