GENERAL PLAN FOR THE PARTIAL DISMANTLING OF THE IRT-SOFIA RESEARCH REACTOR

by

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After the decision of the Bulgarian Government to reconstruct it, the strategy concerning the IRT-Sofia Research Reactor is to partially dismantle the old systems and equipment. The removal of the reactor core and replacement of old equipment will not pose any significant problems. For a more efficient use of existing resources, there is a need for an engineering project which has been already prepared under the title "General Plan for the Partial Dismantling of Equipment at the IRT-Sofia as a Part of the Reconstruction into a Low Power RR".

Key words: partial dismantling, decontamination, reactor equipment, radioactive wastes

INTRODUCTION

The IRT-Sofia research reactor was designed and constructed from 1958 to 1961. First criticality was reached in September 1961, the research reactor (RR) was in operation for 28 years, until July 1989. The reactor has been started up 4189 times, was in operation 24623 hours altogether, at different power levels (by 2 MW), agreed upon with the users at regular weekly meetings. It was shut down in 1989.

The IRT-Sofia is a pool type reactor, cooled and moderated with light water. The core contains up to 48 fuel and graphite assemblies. There are 14, 15 or 16 fuel rods per assembly. The fuel rods are of the EK-10 type (10% enrichment) and C-36 (36% enrichment). The reflector includes 13 graphite blocks.

The safety and control system includes 7 in-core rods, 2 safety rods, 4 regulating rods (made from boron carbide clad in aluminum) and 1 automatic regulating rod (made from stainless steel clad in aluminum). The cooling system includes 3 pumps, a special ejector pipe, max flow rate 540 m^3, 2 heat exchangers, ion exchange and mechanical filters. The maximum capacity of the storage pool is 112 fuel assemblies. It has connections to the reactor pool and hot cell laboratories. There are 11 horizontal and 12 vertical experimental channels, the maximum neutron flux on 2 MW thermal power is $2\ 10^{13}$ per cm^2s^1.

The principal areas of reactor usage ranged from basic and applied research, to technological and commercial applications. Moreover, it was an important place for university and postgraduate education and training.

In order to realize the reconstruction project of IRT-Sofia, it is necessary to develop and fulfill a General Plan for the partial dismantling of the RR, according to IAEA Safety Guides [1] and [2]. Taking into consideraton the actual radiation conditions inside the reactor pool, determined by the reactor operational staff, it is necessary:

- To describe the dismantling and waste processing activities (characterization, conditioning, decontamination, transport, storage and final disposal);
- To design and provide facilities and equipment required to support the dismantling activities according to safety guide [3];
- To take into consideration the value of the decontamination (cost-benefit) for internal or external surfaces of components, systems, and instruments. Decontamination can be carried out before and/or after dismantling;

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- To provide special transportation and shield containers for the transportation of the dismantled structure units and elements to the final storage facilities for radioactive waste (RAW) which would ensure safety for the staff, population, and the environment, according to safety guide [4];
- To define how the plant and operations will be monitored during the partial decommissioning. Appropriate radiation and dosimetry controls are to be ensured during the course of the above mentioned operations;
- To provide the appropriate mechanization for the reconstruction, in conformity with the General Plan, as well as with architectural and building solutions;
- To develop a technological process for the dismantling of parts and equipment of the IRT-Sofia reactor. The technological process of dismantling should be developed taking into account the actual radiation situation in the aluminum tank. These procedures should also be carried out according to appropriate radiation and dosimetry control measures;
- After defining the necessary human, material and financial resources, to draw up a schedule for the reconstruction of the reactor and its systems;
- To provide a management organization structure which is appropriate for partial decommissioning. The structure used during the operation of the RR is not necessarily optimal for partial decommissioning according to safety guide [5];
- To provide a safety assessment;
- To provide an environmental impact assessment report;
- To describe the order and way of examining these reports, as well as their storage;
- To provide a quality assurance program.

DISMANTLING ACTIVITIES

The disassembly of reactor systems, removal of the reactor core and replacement of old equipment will not pose any significant problems. Many of these activities are within the scope of what would be termed refurbishment, a common procedure during power upgrading of pool-type reactors. These tasks will be within the capability of the operator's organization – the Institute for Nuclear Research and Nuclear Energy (INRNE), supported by the contractors when it comes to the provision of tools and skills required for size reduction, handling, and transportation of the wastes. The General Plan for partial dismantling identifies the roles, chain of command and responsibilities within the dismantling team, interfaces with supporting organizations involved at the INRNE site and sub-contractors.

Dismantling activities can be divided into two main categories.

(a) Activities which can be carried out prior to obtaining the permission for the dismantlement (carried out under the existing license) [6]:

- The decontamination of the primary cooling loop (CL) – pollution (deposits, grease spots, *etc.*) should be cleaned up. The use of mineral acid solutions (for instance, of nitric acid) for pollution-removal is effective when it is used along with low-concentrated organic acids (*i. e.* citric acid). The concentration of acid solutions should be properly calculated, if the damage to the metal parts which brings about problems with waste treatment (for instance, iron producing iron hydroxide, which in itself will result in large amounts of secondary waste) is to be avoided. In order to minimize the generation of liquid secondary waste, for the primary circulation loop, de-ionized water should be mixed (in corresponding ratio) with detergent (decontaminant).
- The drying of the primary circulation loop after the last washing with de-ionized water in order to reduce further corrosion. The removal of liquid wastes for treatment. The drying up of the reactor loop by means of an air current from the existing ventilation or from a compressor.
- All systems which are not going to be used during the dismantling (like the pumps of the First and Second CL, should be switched off).
- The horizontal and vertical experimental channels are to be emptied.
- All peripheral experimental facilities in the reactor building to be removed.

(b) Activities demanding permission from the regulatory authorities:

- The switching off of reactor control systems and removal of the most activated regulation rods and experimental channels, activated devices and ionization. Outside the reactor tank, the dismantling of pipelines envisaged for replacement by new ones. "Cleaning" of the water purification system, heat exchangers, system for collection of leakages and overspills/overflows. The pipelines should be disassembled taken to pieces, in order to avoid cutting by mechanical or thermal means (which would generate radioactive particles and aerosols). For transportation or temporary storage of low active parts, licensed cask containers should be provided.
- The dismantlement of the reactor core vessel (fig. 1) and internal reactor devices. The dismantling possibilities vary; it can be done either by means of ready manipulators (robots, especially elaborated instruments and equipment) or by implementing the "one-piece removal" manner of reactor dismantling in research reactor overhauls, already used in Hungary and the Czech Republic.

Figure 1. Reactor vessel

The reconstruction of the IRT-Sofia will comprise the following reactor systems:
- Core – liable to full scale replacement. The new core loading shall be pursuant with the type of the new, converted (low-enriched) fuel;
- Primary cooling system – replacement of the aluminum lining of the reactor pool of a 60 m^3 capacity, piping, and heat exchangers;
- Secondary cooling system – partial replacement of piping and fittings and the cooling open-air pools;
- Horizontal experimental channels – their number shall be reduced from the existing 11 to 7 and a new channel is to be set for boron neutron capture therapy;
- Spent fuel storage – replacement of the aluminum reservoir of 12 m^3 capacity;
- Electric power supply (EPS) – full scale replacement of cables and equipment;
- Control and Protection System (CPS) – full scale replacement of cables and equipment;
- Radiation Monitoring and Dosimetry System (RMDS) – full scale replacement of measuring lines and equipment;
- Civil Engineering Part – new rooms are to be built in the main reactor hall for main control panels and desks of the reactor;
- Heating and Climatic Systems – new systems to be built and installed;
- Ventilation Systems – partial reconstruction.

Schedule of dismantling activities by stages

The dismantlement of the reactor is to be carried out in two stages. The first stage includes the dismantling of the equipment around and inside the reactor pool and in the biological shielding. The second stage includes the dismantling of equipment in the primary cooling loop. It should be noted that for the realization of stage two, stage one should be carried out first.

The time schedule provides for an eight-hour working day. Its aim is to illustrate the needed times and dismantling sequences, under the stipulation that strict adherence to the plan is not compulsory, but simply a recommendation. On these grounds, an evaluation of the necessary resources has been made in order to estimate the requirements for qualified personnel.

Dismantling sequences

Basic operations

- Dismantling of equipment, pipelines, *etc.* – by cutting thermal processing, other possible methods;
- End-capping of pipelines' open ends and of equipment orifices;
- Carrying the dismantled parts (by hand or by means of hoisting mechanisms) out of the premises;
- Dosimetry measurements and sorting of materials according to their level of contamination and waste types [7, 8];
- Loading into transport packages (freely, in plastic bags, tanks, casks) [9];
- Transportation outside the premises [4];
- Delivery of RAW to the State enterprise "Radioactive Wastes";
- Continuous radiation control during dismantling

Final operations

Include the following activities:
- Carrying the utilized equipment, appliances, and supplementary means out of the premises (after their decontamination, if necessary);
- Removal of the temporary lighting, ventilation, termination of power supplies and other consumption;
- Dismantling of scaffoldings; and
- Cleaning and tidying of the site, check up of the radiation situation and possible decontamination, if necessary.

Inventory and characteristics of radioactive wastes generated by the partial dismantling of IRT

The radiological characterization was carried out by a combination of neutron activation calculations and direct measurements. The characterization is essential for drawing up a Radiation Protection Plan which meets the ALARA principles and requirements for the handling, transportation, conditioning, and disposal of radioactive wastes.

The materials produced during the disassembly are: solid wastes from the dismantling of the reactor's technological systems and components, liquid wastes from the reactor's pool and spent fuel storage, as well as liquid wastes generated during the decontamination process.

The solution of the problem with solid radioactive waste storage arising from partial dismantling will be the Novi Han repository. It is situated 35 km South East of Sofia and is a repository for low and intermediate level category radioactive wastes (LILRW) of industrial, medical, scientific, and agricultural origin. The purpose of the repository is to accept and store LILRW with radio nuclides of a half life of less than 30 years, small quantities of long-lived radio nuclides and spent ionizing sources. Presently, Novi Han serves as a repository for institutional wastes. The decision to upgrade additional storage facilities at the site in order to store the wastes from the IRT partial dismantling has already been made.

The characterization program includes following steps:

– review of historical information,
– implementation of calculation methods,
– Sampling and Analyses Plan preparation,
– measurements of sampling and analyses performance, and
– review, evaluation and comparison of data obtained

The Sampling and Analyses Plan provides for measurements of smears and samples taken (figs. 2 and 3) after draining the water out of the first cooling circle. The exact points where the smears and

Figure 2. Smears taking

Figure 3. Smears and samples taking

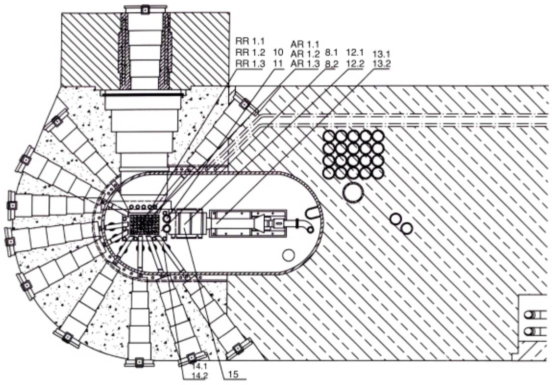

Figure 4. Places of smears and samples taken – horizontal view

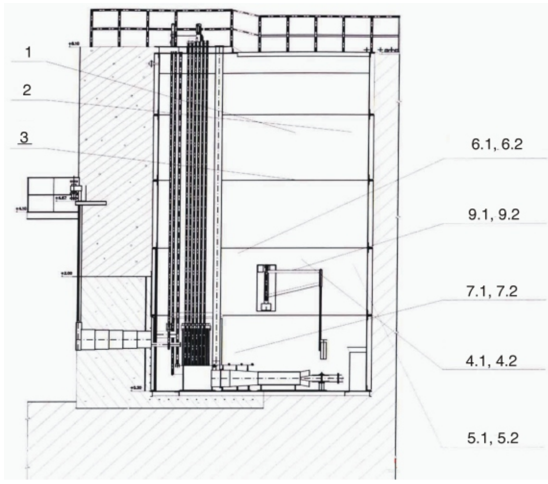

Figure 5. Places of smears and samples taken – vertical view

samples were taken can be seen in figs. 4 and 5. Some results are presented in tabs. 1 and 2.

Safety regulation [10] determines the following RAW categories:

Solid RAW, Category 1 – transitional RAW which can be released from regulation control after

Table 1. Results from smears (measured values)

Smear No.	Place of smear taken	Specific activity Eu-152 [Bq/cm^2]	Specific activity Cs-137 [Bq/cm^2]	Specific activity Co-60 [Bq/cm^2]
1.1	1 cooling loop, 3 pump	0.044	0.0024	0.566
1.2	1 cooling loop, 3 pump	0.042	0.0012	0.448
1.3	1 cooling loop, 3 pump	0.013	0.0011	0.175
2.1	1 cooling loop,valve inlet	0.006	0.0005	0.019
2.2	1 cooling loop,valve inlet	0.002	0.0004	0.004
2.3	1 cooling loop,valve inlet	0.0016		0.001
1	tank wall east 1.45 m from upper water level			0.14
2	tank wall east 2.3 m from upper water level	0.013		1.03
3	pipe at 1.4 m from upper water level	0.005		0.19
4.1	tank wall east 4.3 m from upper water level		0.004	0.14
4.2	tank wall east 4.3 m from upper water level			0.03
5.1	bypass tube 4.5 m from upper water level	0.1		18.2
5.2	bypass tube 4.5 m from upper water level	0.013		0.95
6.1	tank wall west 4.1 m from upper water level	0.013		0.18
6.2	tank wall west 4.1 m from upper water level	0.003		0.07
7.1	tank wall east 5.7 m from upper water level	0.004		0.19
7.2	tank wall east 5.7 m from upper water level			0.03
8.1	shield center 6.0 m from upper water level	0.2		32.4
8.2	shield center 6.0 m from upper water level	0.03		1.7
9.1	console west 4.9 m from upper water level	0.16		33
9.2	console west 4.9 m from upper water level	0.03		6.4
12.1	ejector's table 7 m from upper water level	0.008	0.003	0.47
12.2	ejector's table 7 m from upper water level	0.005	0.0024	0.43
13.1	ejector's tube 7.3 m from upper water level	0.09	0.02	8.1
13.2	ejector's tube 7.3 m from upper water level	0.013	0.005	0.75
14.1	11 horizontal channel 6.7 m from upper water level	0.05		3.2
14.2	11 horizontal channel 6.7 m from upper water level	0.02		1.5

Table 2. Non-fixed surface contamination

Sample No.	Specific activity Eu-152 [Bq/cm^2]			Specific activity Cs-137 [Bq/cm^2]			Specific activity Co-60 [Bq/cm^2]			
	Eu	Eu^*	Eu^{FN}	Cs	Σ Cs	Cs^{FN}	Co	Co	Co^{FN}	Σ_T [Bq/cm^2]
1.1	0.044			0.0024			0.566			
1.2	0.042	0.099	0.124	0.0012	0.0047	0.0059	0.448	1.189	1.486	1.6159
1.3	0.013			0.0011			0.175			
2.1	0.006			0.0005			0.019			
2.2	0.002	0.0096	0.012	0.0004	0.0009	0.0011	0.004	0.024	0.03	0.0431
2.3	0.0016						0.001			
1 tank							0.14			
2 tank	0.013		0.0125				1.03		0.475	0.4875
3	0.005						0.19			
4.1 tank				0.004			0.14			
4.2 tank							0.03			
5.1	0.1						18.2			
5.2	0.013	0.113	0.226				0.95	19.15	38.3	38.53
6.1 tank	0.013						0.18			
6.2 tank	0.003						0.07			
7.1 tank	0.004			0.19						
7.2 tank				0.03						
8.1	0.2	0.23	0.46				32.4	34.1	68.2	68.66
8.2	0.03						1.7			
9.1	0.16	0.19	0.38				33	39.4	78.8	79.18

* – Sum of the separate tampons date for the smear
** FN – Intrinsic value of the non fixed surface contamination (with giving an account of the coefficient of removing)

proper treatment and/or temporary storage for a period of time no longer than 5 years, in which case its specific activity has to fall below the level for RAW release regulated by relevant regulations.

Solid RAW, Category 2 – low and middle-active wastes containing radio nuclides in concentrations requiring no special measures for post-decay power removal during storage and burying. Radioactive wastes of this category are additionally categorized as:

(a) *Category 2a* – short life low and middle-active wastes containing mainly short lifetime radio nuclides (with a half decay period shorter or equal to the ^{137}Cs half decay period and long life alpha-active radio nuclides with specific activity less or equal to 4 10^6 Bq/kg per single package and less or equal to 4 10^5 Bq/kg for the entire RAW volume;

(b) *Category 2b* – long life low and middle-active wastes, containing long lifetime alpha-active radio nuclides (with a half decay period longer than the ^{137}Cs half decay period) with specific activity exceeding the limits for category 2a;

Expected amounts of RAW as a result of the partial dismantling of IRT equipment

I Expected amounts of RAW – 2a category according to regulation [10]:

(1) Aluminum and aluminum alloys from the reactor pool — 100 kg
(2) Steel St 1X18H9T + Al alloy AB from the reactor pool — 170 kg
(3) Shield of the thermal column corpus – casing of St3, filled up with paraffin and boron carbide:
- St3 — 289 kg
- paraffin of a density of 1 g/cm^3 — 210 kg
- boron carbide with a density of 1.5 g/cm^3 — 60 kg

(4) Lead slab – Pb + Al — 211 kg
(5) Graphite assemblies from the deflector of the RC — 172 kg
(6) Ion-exchange resin — 320 kg
(7) Previous measurements suggest it is to be expected that approximately 1000 kg of the thermal column graphite will be activated and subsequently classified into this category.
Expected average activity — 10^8 Bq/kg
In addition:
Contaminated materials from the first CL premise (Al and St) — 3780 kg

After decontamination, it is expected that CL materials will be converted into RAW of the first category or into non-radioactive wastes.

II Expected amounts of RAW – category 1 or non-radioactive wastes, according to regulation [10]:

(1) The concrete shielding of the thermal column – concrete with a density of 4.5-5.5 t/m^3; 24831 kg (part of the concrete situated closest to the RC is expected to be volume activated and to pass over to a upper category of waste, in accordance with the classification for RAW).
(2) Block with shutters of the thermal column – St3 5033 kg.

Note: For RAW is defined the β and γ activity to be in Bq/kg and Bq/cm^2, because it is not expected RAW to contain long-lived α-active radio nuclides with specific activity over 4 10^6 Bq/kg.

Toxic materials

During the operation of IRT, materials as asbestos and mercury have not been used in reactor construction and equipment. Acids and alkalis, as well as organic and non-organic solvents have been used in small amounts in the maintenance of the facility, with strict adherence to norms for the management of such materials. No generation of such materials is expected as a result of dismantling activities.

Non-radioactive materials

A generation of a certain amount of non-radioactive materials is also expected as a result of dismantling activities. This pertains to metals from the facility outside the reactor pool and the 1st circulation loop premise, materials from the reactor pool itself and from the 1st circulation loop premise after the deactivation, as well as to the larger part of the concrete and fixtures pertaining to the neutron therapy channel preparation.

Liquid radioactive wastes

There are no facilities for the treatment of liquid RAW at the RR site. They are collected in underground stainless steel tanks with a total volume of approximately 300 m^3, situated near the reactor building. An expertise of the state they were in was carried out in 2004 by a licensed organization, the results positive, proving their usability for future operation. A transportation of approximately 250 m^3 low-level liquid RAW was carried out in 2000. About 84000 l are to be transported to the Kozloduy NPP for reprocessing before the dismantling commences.

Radiation protection of personnel and environment

In the development of the schedule for dismantlement, the basic principles of radiation protection should be observed according to regulation [11]: the limit for the personnel effective dose is 100 mSv over

5 years and the maximal effective dose per year must not exceed 50 mSv.

During the dismantling of the IRT, radiation control systems will provide:

- Necessary individual dosimetric controls of the personnel;
- Surveillance of radioactive contamination; control over the radiation situation at the premises and radiation control of the air and filters for the staff and supplementary (local) ventilation systems;
- Monitoring of emissions; control over radioactive wastes and drainage waters, including available mobile facilities for low and middle-active liquid RAW treatment;
- Control over collection, sorting and categorization, storage and transportation of liquid and solid RAW;
- Coordination/compliance with approved limits during the operation/dismantling;
- Engineering protective systems (screens, shields, ventilation, *etc.)*. It is crucial to have at one's disposal mobile ventilation systems, protective materials (lead blocks and covers), PVC sheets for covering and protecting the contaminated clothes washing premises;
- Control over the radioactive materials and wastes (by means of accounting programs, control zones, *etc*.);
- Calibration;
- Education/training;
- Emergency planning, including emergency radiation monitoring;
- Unforeseen events;
- Quality assurance (Program for Audit and Examinations);
- The minimization of personnel irradiation risk in accordance with the ALARA principle quality assurance program.

There is a functioning Integrate Quality Management System (IQMS) in INRNE – ISO 9001:2000 and 14001:1996 which takes the NRA – Bulgaria and IAEA recommendations into account.

The present quality assurance program (QAP) defines the specific responsibilities, organization, and technical undertakings for achieving the necessary quality of work during the implementation of activities described in the plan for partial dismantling.

Expected dose exposures during the disassembly, individual and collective doses

In general, the following work zones are formed during partial dismantling:

– in the reactor tank,
– at the site of the dismantlement of the thermal column from the outside,
– at the reactor platform,
– on the dismantling platform, and
– at the first CL premises.

Based on the measurements, an assessment of expected dose exposures of the personnel during partial dismantlement is made.

External irradiation of personnel

At the reactor pool, when full

Activities connected with the dismantling of the reactor platform equipment are carried out with the pool still full of water in order to reduce the dose exposure of the personnel. The doses to which the personnel is expected to be exposed to during these activities are negligible, since the dose rate at the reactor platform during operation has been within the limits of 144-216 μSv/h.

At the reactor pool, when drained

According to the above introduced zones, the following considerations have been made:

– in activities carried out on the reactor platform, mark 8.10 meters, and
– the measured dose rate for the empty reactor pool is 35 μSv/h.

On the basis of time schedules, the time needed for carrying out all activities at the reactor platform after its drainage, is estimated to be 360 hours and the necessary personnel permanently occupied with the activity – 2 workers. It is anticipated that 5% of the time they will be accompanied by a dosimetry-man and personnel providing safety techniques. The total expenditure of work by the personnel is estimated to 756 man-hours. The collective dose under these conditions is estimated to amount to 26.5 mSv, maximal expected individual dose to 12.6 mSv.

Dismantling activities in the reactor vessel

The time necessary for the implementation of these activities is estimated to 15 hours. During their execution, only one person will be in the reactor pool. The prediction is that around 5% of the said 15 hours, a dosimetry-man will be present on the spot. This makes a total of 15.75 man-hours. The maximal measured value in the reactor pool is 6.48 mSv/h. The expected maximal collective dose value is estimated to 102 mSv. Because of this, consecutive replacements of several workers will be necessary during the dismantling of the equipment. The aim is not to exceed the indicated constraint values for personnel irradiation. Special protective means should be used when needed.

Activities on the dismantling platform

Daily activities are to be carried out by 2 workers; it is thought that around 5% of the time a dosimetry-man will be present. The total time for manipulating materials from the reactor interior on the dismantling platform is estimated to 40 hours. The estimated overall total time required, 82 men-hours. The various activities planned on the dismantling platform will necessitate the handling of numerous details. The determination of a maximal equivalent dose (MED) is done on the basis of measurements done on the most activated reactor detail – the control rod (CR). Its maximal measured value is 3.0 $mSvh^{-1}$. A conservative estimate is that all details subject to work will induce similar dose rates. The expected maximal value of the collective dose is 246 mSv. The use of remote techniques is also recommended. Prior to the commencement of the dismantling process, appropriate training for relevant skills and the reduction of outages is recommended.

Because of that, consecutive replacements of the work force will be necessary.

First CL premises – the highest measured MED is near the heat exchangers and stop valve – value of 5 Sv/h is at a distance of 10 cm from them).

Obviously, the irradiation at the first CL premises does not present a practical danger for the workers. The time necessary for the dismantling of heat exchangers is estimated to 60 hours, presuming that all operations necessary in the dismantling of the heat exchangers, including the dismantling platform, will be carried out by 4 workers. The presence of a dosimetry-man is foreseen about 5% of the time, bringing us to a total of 243 men-hours. Consequently, the maximal individual dose is not expected to exceed 0.3 mSv, while the collective one should not be higher than 1.215 mSv.

This evaluation has been made without taking into account the presumable decontamination which is expected to reduce the contamination for not less than an order of magnitude.

The use of a number of individual protection means is foreseen, as well as a solution for additional ventilation with the aim of providing collective protection.

The expected internal irradiation of the personnel is many times less than the external. Internal irradiation of the population is not expected. In spite of that, a permanent monitoring of air parameters is compulsory during the work process, so as not to allow a possible increase in constraint levels. The collective dose during the partial dismantlement can be estimated to 375 mSv. Taking into consideration the conservative approach accepted for determining the personnel dose exposure, it can be concluded that the collective dose will cover all participants in the dismantling process.

Finally, the following prerequisites should be taken into account:

- During the dismantlement, each member of the personnel is to be kept under control, so as not to obtain a dose larger than the constraint annual dose (taking into consideration doses obtained in previous activities in ionization radiation environment and bearing in mind control limits determined by internal instructions),
  - in case an established limit is breached, measures are to be taken for the termination of activities with ionization radiation sources, carried out by that person,
  - additional measurements should be carried out after the removal of horizontal channels 9 and 10, because they have the highest level of activation,
  - dismantling activities should be carried out from a distance whenever this is possible, and
  - all protection means applicable in a particular case are to be used – such as putting on leaden protective waistcoats, covering the highly activated details with protective shields, *etc*.

Observation of the ALARA principle

The purpose of radiation safety is to establish conditions enabling personnel radiation doses to be lower than the limits indicated in normative documents and providing adequate control of personnel exposure doses during accidents.

This necessitates the acknowledgment of the ALARA principle whose essence is to achieve a reasonable reduction of the collective dose, one honoring the technical and economical point of view, as well.

The application of this approach leads to following general formulations:

- All processes which can be realized by means of standard technical means are to be realized in such a manner that the presence of personnel is not required;
- Processes in which the automation would require the application of complicated and expensive robotized devices are to be carried out with the participation of people, but utilizing maximal automation and mechanization for individual, key operations;
- With the aim of minimizing the doses from external and internal irradiation, the personnel participating in RAW treatment should be equipped with protective means adequate for carrying out such activities;
- Strict restrictions regarding the access to the RAW work zone and limitation of personnel numbers, so as to minimize the collective dose;

- Minimization of work hours spent in zones with the increased values of dose rates;
- Strict control over the radiation situation and personnel dose exposure;
- Training and permanent improvement of personnel qualifications, permanent work in improving safety culture knowledge.

CONCLUSION

The results from the implemented measurements and calculations are the initial basis for determining the nature of the dismantling techniques to be employed: *i. e*. fully remote, semi-remote or manual methods; the need for decontamination; radiation protection for the personnel, public and the environment; RAW classification according to national categories; the requirements for handling, transportation and storage of wastes following dismantling; the price of dismantling *(i. e.* overall project costs, including waste management). Additionally, the results are also necessary for the drawing-up of other parts of the plan for dismantlement, such as: the assessment of dose exposure; assessment of hazard; assessment and selection of a scenario in compliance with the ALARA principle, as well as the choice of adequate protection means needed in the dismantling process.

On the basis of the provisional General Plan for the partial dismantling of the IRT-Sofia, after having obtained the permission of the National Regulatory Agency, a detailed time schedule for the implementation of the Plan is currently being drawn up.

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ГЕНЕРАЛНИ ПЛАН ЗА ДЕЛИМИЧНО УКЛАЊАЊЕ ИСТРАЖИВАЧКОГ РЕАКТОРА ИРТ У СОФИЈИ

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После одлуке бугарске владе да реконструише Истраживачки реактор ИРТ у Софији, развијена је стратегија делимичног уклањања старих система и опреме. Претпостављено је да ће измештање реакторског језгра и замена старе опреме бити обављени без већих тешкоћа. Ради ефикаснијег коришћења расположивих средстава, постојала је потреба за инжењерским пројектом који је припремљен под називом “Општи план за делимично уклањање опреме ИРТ-Софија, као део поступка реконструкције у истраживачки реактор ниске снаге”.

Кључне речи: делимично уклањање, деконтаминација, реакторска опрема, радиоактивни отпад