Physics

The Tasks Related to the Decommissioning of Georgian Nuclear Research Reactor

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ABSTRACT. The nuclear research reactor IRT-M belonged to the Institute of Physics and operated during the 1960-1988 period. All nuclear fuel (both fresh and spent) was sent out of Georgia after the reactor was shut down. The reactor decommission activity was conducted at different stages. Special activity plans were developed considering safety and security requirements, international experience and local circumstances. The developed decommissioning strategy was based on the idea on possible establishment of a new low power nuclear facility in the tank of the decommissioned reactor. At the first stage the reactor core together with some comparably high active waste was covered with special concrete. A special underwater concreting technology was developed for this purpose. Theoretical assessment of radiation was conducted before the activity. The results of theoretical assessment were in good correlation with data obtained at radiation monitoring during the concreting activity. During the second and third stages the reactor system inside and outside of the reactor building was dismantled. Lacking the capability to conduct full scale radioactive waste treatment, a special technology of waste packaging is developed to meet the safety requirements. The creation of a radioactive waste treatment facility was defined as a next stage of the activity. *© 2012 Bull. Georg. Natl. Acad. Sci.*

Key words: *nuclear reactor, decommissioning, radioactive waste.*

The IRT-2000 Research Nuclear Reactor, having the heat power of 2MW, was commissioned in 1960. IRT–2000-type nuclear reactor, designed in the Soviet Union, belongs to the group of light water pool-type reactors, in which the light water is used as a moderator and a reflector of neutrons and as a coolant and a part of biological shielding of the reactor. Nuclear reactors of such type were also built in Russia (Moscow, Tomsk, Sverdlovsk), Byelorussia (Minsk) and Latvia (Riga). Besides, IRT–2000 type research nuclear reactors were constructed in Bulgaria (1961), China (1962), North Korea (1965) and Iraq (1967) [1, 2].

The pool of IRT–2000 reactor was a tank made of

6mm thick aluminum alloy sheets surrounded by 1.8 m thick biological shielding of reinforced concrete. The height of the pool was 7.6 m and the entire volume – about 60 m^3. The following units of the reactor were located on the bottom of the pool under a six-meter water layer: the reactor core, ejection pump and pipelines of the primary circuit of cooling system, loading and unloading devices, channels of control and emergency protection systems and 10 horizontal experimental channels of various diameters (100-150 mm), which bordered the core and served as an outlet of neutron beams to the experimental installations. The pool of the reactor was filled with distilled water up to the height of 7.2 m and was covered with a lid of organic glass. The reactor was also equipped with vertical experimental channels for irradiation of samples in the vicinity and within the core. At the reactor power of 2MW, the maximum flux of thermal neutrons measured in the center of the core was $2.5\cdot10^{13}$ n/$cm^2\cdot$sec, in the water reflector of the core - $5\cdot10^{12}$ n/$cm^2\cdot$sec and at the outlet of the horizontal experimental channels - $5\cdot10^8 \div 1\cdot10^9$ n/$cm^2\cdot$sec.

In 1967-1968 the research IRT-2000 nuclear reactor was subjected to the first large-scale refurbishment which was connected with the necessity of increasing reactor power (about two times) and of the possibilities of carrying out the experiments mainly in the reactor core. The refurbished reactor, named IRT-M (M-modernized), had the power of 4MW and it operated reliably till 1973. In 1973-1975 the IRT-M reactor was subjected to the second large-scale refurbishment with the following aims: to replace the aluminum tank and in-tank equipment of the reactor by a tank and the same equipment made of stainless steel - being the most corrosion protective material. The new IRT-2M - type fuel assemblies and the modification of the cooling system with the use of new circular pumps allowed to increase the reactor power to 8MW. Accordingly, the neutron flux was increased up to ~10^{14} n/$cm^2\cdot$sec.

For 28 years (1960-1987) the nuclear reactor of the Institute of Physics operated more than 70,000 hours. More than 9 000MW·day heat energy was produced, which corresponds to the consumption of nuclear fuel - Uranium-235 in the amount of 11 kg. 201 fuel assemblies of various types were used, in which the total content of 90% enriched Uranium-235 was about 30kg.

In 1990 the Presidium of the Georgian Academy of Sciences, taking into account the limited life time and absence of further safe operation of the reactor, the decision was issued to shut down the reactor and start its decommissioning procedure. The decommissioning activity was based on the concept of transforming the reactor to a low power nuclear facility using the reactor infrastructure (mainly the reactor pool to install the new nuclear device in it).

Preliminary decommissioning activities

The prepared Preliminary Decommissioning Plan (PDP) considered the following activities:

- Unloading the spent fuel from the reactor core.
- Preparation of the spent fuel assemblies of all types accumulated at the reactor by the time of its shutdown for their transportation to the reprocessing plant, and to accomplish the measures for their transportation in accordance with the transport regulations.
- Return of the fresh nuclear fuel in the form of separate rod type fuel elements and assemblies of different design to the manufacturer (or shipment to another reactor).
- Preparation of a complete inventory of all kind radioactive waste accumulated at the reactor site during its long operation life.

According to the programme all fuel (both fresh and spent) under international control was sent out of Georgia. Special radiological characterization of the reactor structures was done. ^{60}Co radioanuclide was defined as the main contaminant with total amount ~2 TBq. The greatest part of contaminated devices contains auxiliary system pipes that are characterized by inner surface contamination.

Based on the obtained data a Detailed Decommissioning Plan (DDP) has been prepared in accordance with the International Atomic Energy Agency (IAEA) guidance [2,3] and advice provided by the IAEA via the relevant Technical Cooperation Project (GEO/4/002) - "Conversion of Research Reactor to a Low Power Facility"[4].

The plan considered three stages:

I – Concreting of the lower part of redundant Research Nuclear Reactor IRT-M and cumulated high radioactive waste;

II – Dismantling of the technological systems of the Reactor;

III – Dismantling of the external technological system of the low-temperature complex of the reactor.

Concreting of the lower part of the reactor pool

It was decided to make a special confinement for the reactor core by filling of the lower part of the reactor pool (where the core is situated) by shielding material [4]. This confinement should also include some comparable highly contaminated small parts of the reactor devices. It is foreseen that a new low power nuclear device can be installed on the confinement in the existing reactor pool. The decision was based on the following criteria:

- No radioactive waste storage facility existed at that time;
- Lack of appropriate financial and human resources and technical devices.

It was also considered that the proposed method:

- is radiologically and ecologically safe;
- is radiation resistant and seismically safe;
- achieves minimal generation of radioactive waste and hence, copes with the problem of their accumulation, conditioning and transportation;
- financially affordable;
- provides the possibility of installing a new low power facility in the radiation free part of the reactor tank.

Before carrying out the procedure of the confinement, a special theoretical estimation of gamma-radiation on the surface of the concrete was made, using computer code "MicroShield" [5]. First of all, the possibility of using different shielding materials was estimated. The analysis of the obtained results shows the preference of concrete, considering the financial resources and technical aspects of the planned work. It should be emphasized that, according to the calculations, the amount of concrete, necessary to reduce gamma dose rate to natural background is not very large, and there is enough free space in the reactor tank to install a low power nuclear facility, which can be used for medical purposes (producing of medical radionuclides or treatment exposure of medical patients).

In accordance with the elaborated plan, the whole activity was divided into the following ten steps:

1. Preparation of the reactor hall;
2. Preparation of the reactor tank with its internals;
3. Preparation of the experimental horizontal channels;
4. Preparation of the radioactive waste;
5. Preparation of the auxiliary systems and reactor equipment;
6. Concreting the reactor tank;
7. Concreting the 8 experimental horizontal channels;
8. Concreting the waste in the dry storage vertical channels;
9. Concreting the waste in the storage well;
10. Installation of the monitoring and surveillance system.

A special underwater concreting method was used for filling the reactor tank with special concrete. The method provides more efficient shielding for specialists engaged in concreting the tank. During the operation gamma dose rates were measured at different stages. A comparison of the measured data with the calculated values proved the acceptance of the conducted theoretical calculations. Thus, at the first stage the water was pumped out to the mark 2.85m above the tank bottom level. The gamma dose rate measured on the surface of water was equal to 5μSv/h (Calculated value 8.75μSv/h). At the next stage the reactor tank was filled with the concrete (underwater concreting) up to the mark 1.75 m from the tank bottom. The measured value of the dose rate on the water surface was 0.4μSv/h (calculated value 0.2μSv/h). After the concreting the water was pumped out from the tank. According to the conducted measurement the gamma dose rate on the surface of the mortar was 25μSv/h (calculated value 26.6μSv/h). According to the theoretical estimations, to reduce the gamma dose rate on the surface of the protective concrete block in the reactor tank practically down to natural background, further concrete-covering of the tank from the mark 1.75m to the mark 2.9m was conducted. All activities were performed under the IAEA Technical Cooperation Project (GEO/4/002) - "Conversion of Research Reactor to a Low Power Facility".

Dismantling of the technological systems of the reactor

It proved possible to conduct the planned activity was due to the construction on the site of Applied Research Center (Reactor site) Centralized Storage Facility (CSF). All activities were conducted under IAEA TC project GEO/3/002 whose aim was "to decontaminate and to dismantle the remaining radioactive parts of the IRT-M research reactor and to manage safely the radioactive waste generated from the dismantling operations". The project scope was defined as a dismantling of all radioactive devices located inside the reactor building, including the following:

1) The dual-circuit cooling system of the reactor.

2) The system of mechanical and chemical purification of the coolant of the primary circuit of the reactor cooling system.

3) The part of the pipeline of the system of circulation of gaseous helium.

4) The system of filters intended for cleaning the air from radioactive gases and aerosols being ventilated from the above-reactor space and different special technological rooms prior to their release into the atmosphere.

5) Devices of mechanical and chemical purification of water of pools intended for temporary storage of the fuel assemblies and cassettes.

Before the dismantling activity a special decommissioning programme was developed. The programme elaboration was based on previously conducted radiological monitoring. Dosimetry and gamma-spectrometry measurements were made on pipelines under dismantling. The values of radioactive contamination of the internal surface of the pipes were estimated by using the data on the characteristics of the material of the pipes and the width of their walls. Another method was used: the pipes were screwed at defined places and smears were taken from internal surfaces of the tubes. The data obtained by both methods shows good correspondence. It was found that the devices are characterized by inner surface not fixable contamination. The main contaminant nuclide was defined as ^{60}Co and ^{137}Cs. According to the obtained results, the average contamination value was 20-30 Bq/cm^2 and dose rates near the contaminated surfaces were in the range 5-50 mSv/h. The total activity of the pipes under dismantling was $1.8\text{x}10^9$ Bq. The clearance level for radionuclide ^{60}Co was defined as 1Bq/cm^2. Based on this value the pipes from secondary circuit were assigned as radiologically clean. So, secondary circuit tubes were released from radiation control after dismantling.

It should be stated that Georgia has not developed a special system for waste treatment. Therefore special tools and methods for waste processing were developed: All dismantled parts (pipes, valves etc.) were segmented, sealed hermetically and sent to CSF. Many parts of the pipes were screwed; others were cut, using abrasive cutting technique. The big pipes were used as containers (packages) for small ones. As a result, 34 large packages were prepared with the total weight of ~ 6,750 kg. Each package has a label indicating the number and weight of the package.

The special inventory contains detail explanation (including radiological characterization of each package). It is shown clearly that the proposed method creates acceptable safety conditions only for a temporary period of time (several years). Taking into account the possible corrosion and drying of ribbon materials, radionuclide leakage from the packages can appear after some years of storing.. Therefore it is very important to conduct routine radiological survey of the packages kept in CSF. A system for treatment of the waste should be developed to clean all dismantled pieces, which gives a satisfactory level of waste size reduction factor and, (which is most important) the safety level of the waste, taking into account its long-term storage and further opportunity of its disposal.

200l drums were used for conditioning comparably high active and small-size metallic parts, where the waste was covered with concrete. The same technique was used for ion exchange resins, which were placed in stainless steel boxes (4 pieces) and hermetically closed. The boxes were placed into 200l drums and covered with concrete. As the radiological monitoring showed, the dose rate on the surface in the case of opened boxes was: 125-130 μSv/h, and on the surface of closed boxes – 50-55 μSv/h. After concreting of drums with boxes containing resins, the dose rate decreased to 7-10 μSv/h.

Thus, 26 concreted drums were placed into CSF. During this stage all technological systems within the reactor building were dismantled and the following waste generated:

- Concreted waste in drums:

$^{137}Cs – 1.013 \cdot 10^{10}$Bq and $^{60}Co – 6.2 \cdot 10^{8}$Bq;

- Sealed pumps:

$^{137}Cs – 8.06 \cdot 10^{6}$Bq and $^{60}Co – 7.44 \cdot 10^{7}$Bq;

- Ion exchanger filter:

$^{137}Cs – 2.3 \cdot 10^{9}$Bq and $^{60}Co – 7.2 \cdot 10^{8}$Bq;

- Sealed up pipes (packages):

$^{137}Cs – 1 \cdot 10^{8}$Bq and $^{60}Co – 2.65 \cdot 10^{9}$Bq;

- Sealed heat exchangers:

$^{137}Cs – 2.59 \cdot 10^{7}$Bq and $^{60}Co – 5.18 \cdot 10^{7}$Bq

Dismantling of the external technological system of the low-temperature complex of the reactor

During the operation of the reactor IRT-M a large number of physical experiments were carried out at low temperatures using liquid nitrogen and helium. The liquid nitrogen was provided to the reactor building from a cryogenic station situated at the reactor site by a special system of tubes. The conducted radiological survey confirms the existence of the same contaminants in the same condition as they were characterized for the reactor building tubes. The maximum contamination was fixed at 380 Bq/cm^2. All activities were conducted under the IAEA TC project GEO/3/004. The same method and tools were used for dismantling and waste treatment. As a result 27 395 kg pipes were dismantled with total activity of $^{60}Co – 1 \cdot 10^{8}$Bq. It was noted that free volume of CSF is not enough to keep all dismantled parts. Therefore dismantling of the cryogenic station was delayed until the waste treatment facility was established at the reactor centre. The facility should be designed to treat all waste generated during the reactor decommissioning and all other types of radioactive waste to reduce their size and convert them into a more safe condition.

ფიზიკა

ქართული ბირთვული საკვლევი რეაქტორის დეკომისიის ზოგიერთი ამოცანები

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(წარმოდგენილია აკადემიკოს ჯ. ლომინაძის მიერ)

ე.ანდრონიკაშვილის ფიზიკის ინსტიტუტის კუთვნილი ბირთვული კვლევითი რეაქტორი IRT-M ოპერირებდა 1960-1988 წლებში. რეაქტორის გაჩერების შემდგომ მთელი ბირთვული საწვავი (როგორც ახალი, ისე მოხმარებული) გატანილ იქნა საქართველოდან. რეაქტორის დეკომისია ხორციელდება ცალკეულ ეტაპებად. აქტივობისათვის შემუშავდა სპეციალური გეგმები, რომლებიც ეყრდნობიან ბირთვული უსაფრთხოების და უშიშროების მოთხოვნებს, ითვალისწინებენ საერთაშორისო გამოცდილებას და არსებულ ადგილობრივ სიტუაციას. დეკომისიის სტრატეგია ეყრდნობა იდეას მცირე სიმძლავრის ბირთვული დანადგარის რეაქტორის ავზში განთავსების შესახებ. დეკომისიის პირველ ეტაპზე განხორციელდა რეაქტორის აქტიური ზონის სხვა შედარებით მაღალ აქტიურ ნარჩენებთან ერთად დაბეტონება, რისთვისაც შემუშავდა სპეციალური წყალქვეშა ბეტონირების მეთოდი. სამუშაოების დაწყებამდე განხორციელდა რადიაციული ველების თეორიული შეფასება. მუშაობის დროს წარმოებულმა რადიაციულმა მონიტორინგმა დაადასტურა თეორიული შეფასების შედეგების სისწორე. დეკომისიის მეორე და მესამე ეტაპზე განხორციელდა რეაქტორის შენობის შიგნით და გარეთ მყოფი სისტემების დემონტაჟი. რადიოაქტიური ნარჩენების გადამამუშავებელი სიმძლავრეების არქონის გამო შემუშავდა ამ ნარჩენების შეფუთვის ახალი ტექნოლოგია, რომელიც უზრუნველყოფს რადიაციული უსაფრთხოების საჭირო დონეს. რადიოაქტიური ნარჩენების გადამამუშავებელი მცირე საწარმოს დაფუძნება განისაზღვრა როგორც დეკომისიის შემდეგი ეტაპი.

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