NUCET NUCLEAR ENERGY AND TECHNOLOGY

Nuclear Energy and Technology 10(1): 55–62
DOI 10.3897/nucet.10.123054

Research Article

Ensuring radiation safety during dismantling, transportation and long-term storage of the SM-3 research reactor core*

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Academic editor: Yury Kazansky ♦ **Received** 23 August 2023 ♦ **Accepted** 25 November 2023 ♦ **Published** 29 March 2024

Citation: Yusupov AN, Mikhailov PA, Kizin VD, Gromov MO, Kusovnikov AV, Avdonin VV (2024) Ensuring radiation safety during dismantling, transportation and long-term storage of the SM-3 research reactor core. Nuclear Energy and Technology 10(1): 55–62. https://doi.org/10.3897/nucet.10.123054

Abstract

Described shortly here is a procedure of demounting, removal, transport and long-term storage of the SM-3 core, based on the previous experience of reactor refurbishment undertaken in 1991. Prior to performing refurbishment, computations and calculated data analysis were performed to prove radiation safety of this work, which included estimation of the activity level for activation products in the structural materials of the nuclear research reactor core and the radiation conditions at different stages of its handling. As evidenced by the calculated data, the activity of the main dose-forming radionuclide ^{60}Co attains equilibrium in about 12 years of radiation exposure. Taking into account the fact that the time period between two refurbishments was longer than 12 years, the calculated values of the equivalent dose rate were normalized to the radiation monitoring data obtained during the previous refurbishment, taking into account the calculated activity of ^{60}Co radionuclide. The normalization made it possible to confirm reliability of estimates. The obtained activity data of activation products and taking into account the time spent during the SM-3 refurbishment in 1991, the radiation impact on personnel was estimated. Calculated values of the anticipated effective radiation exposure doses to the personnel engaged in the refurbishment revealed that the main limits of the personnel radiation exposure established in accordance with NRB-99/2009 were not exceeded.

Comparison of the results of calculating the equivalent dose rate with the results of radiation monitoring at various points allowed us to establish that during the calculation and analytical justification of the radiation safety of work, the assessment of reflected radiation was significantly underestimated. But the radiation monitoring data, personal radiation monitoring, as well as recorded data of automatic radiation monitoring system show that all work was performed in compliance with the requirements of regulatory documents in the field of radiation safety.

Keywords

SM-3 research reactor, reactor core, equivalent dose rate, effective dose rate, induced activity

* Russian text published: Izvestiya vuzov. Yadernaya Energetika (ISSN 0204-3327), 2023, n. 4, pp. 95–110.

Introduction

In 2020, an investment project of Rosatom State Corporation was successfully completed at RIAR JSC for upgrading the reactor core of the SM-3 high-flux nuclear research facility. As part of the SM-3 upgrading, the reactor core was dismantled and withdrawn and transported further to the solid radioactive waste (SRW) site for long-term storage.

To comply with regulatory requirements (NRB-99/2009, OSPORB-99/2010), personnel exposure to radiation was estimated. Recommendations were provided based on the obtained results regarding the need for using additional biological shielding for the driver in the process of transportation and for the crane operator in the process of placing the reactor core in the SRW storage. The results of estimating the expected effective exposure dose for personnel showed that no basic dose limits, as defined in NRB-99/2009, were exceeded.

The SM-3 reactor core was dismantled, transported and placed in long-term storage in conditions of individual health monitoring for personnel and continuous monitoring of the radiation situation using the reactor building radiation monitoring systems and the automated radiation situation monitoring system (ARSMS). The operations were performed based on work permits for carrying out radiation-hazardous and extra hazardous operations. Ensuring radiation safety in the process of the operations to dismantle and transport the SM-3 reactor core and place it in long-term storage was of top priority and consisted in minimizing the personnel exposure doses, as well as in preventing the contamination of the environment in the surrounding area at all stages of operations.

The purpose of this paper is to describe the procedural approach to estimating and interpreting the radiation monitoring, ARSMS and individual health monitoring system readings in the process of operations.

Description of the reactor core disposal technology

The SM-3 reactor core shown schematically in Fig. 1 is made of steel and is the load-bearing structure on which the core and the reflector are assembled. It comprises the vessel, the support and the upper grid with storages for fresh and spent FAs. The vessel is cylindrically shaped. The vessel's bottom part has holes for the coolant delivery into the pressure chamber beneath the reflector which represents a cavity between the support plates. The support is attached to the vessel from below and is designed as a welded structure of two plates rigidly interconnected at the center through the cases for the FA accommodation, tubes for the tail pieces of beryllium trap inserts, the central tube for the central transuranic target assembly (CTTA), and two sleeves for irradiation cells.

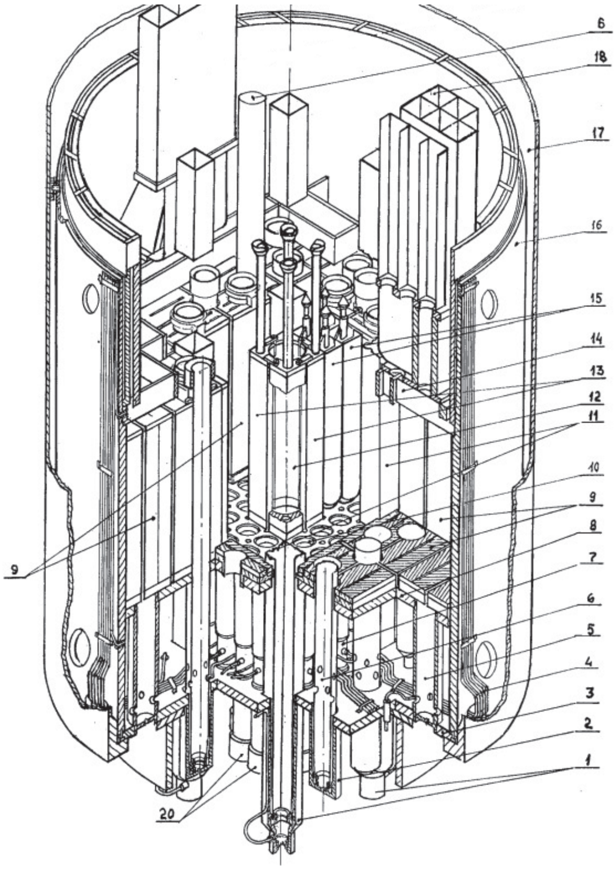

Figure 1. SM-3 reactor core: 1 – shim rod tube sockets; 2 – automatic controller tube socket; 3 – support grid lower plate; 4 – cladding integrity monitoring tube; 5 – support grid rack; 6 – automatic controller tube; 7 – FA sockets; 8 – support grid upper plate; 9 – beryllium blocks; 10 – FA sockets; 11 – cladding intregrity monitoring tubes; 12 – central trap; 13 – CTTA beryllium inserts; 14 – reflector hold-down grid; 15 – working FA; 16 – vessel; 17 – reactor vessel shield; 18 – FA storage No. 1.

The SM-3 upgrading program included dismantling, withdrawal, transportation and placement of the reactor core in long-term storage. The operations were performed on a day off to exclude the irradiation of personnel within the institute's commercial site. The following preparatory operations were performed in advance:

- the ceiling panels were removed above the vehicle entrance to the SM-3 reactor building;
- the shielded cask fastened on a semitrailer truck was placed under the transport hole in the reactor building's central hall;
- an automated radiation monitoring system was prepared and additional detection units with the maximum variation range of the gamma equivalent dose rate (EDR) were installed along the reactor core transportation route;
- all equipment was additionally tested for serviceability;
- drills were conducted with the SM-3 reactor core simulator;

- the institute's subdivisions, the firefighting service, the commander of the National Guard unit, and the Atomokhrana (Nuclear Guard) leaders were informed of the activities to be undertaken, and the institute's non-professional emergency response teams were placed on high alert.

Two teams of five persons each and two backup teams of five persons each were formed for carrying out handling operations. One of the teams was responsible for the operations to withdraw the core from the reactor, load it into the cask, and remove the cask from the reactor building. The second team was responsible for the operations to remove the cask fasteners, transfer the cask and place it on the compartment, as well as to unload the cask inside the SRW storage building.

Two work permits were issued on the operations day for carrying out radiation-hazardous operations in the reactor facility and SRW storage buildings with continuous health monitoring (including with the use of self-reading dosimeters with audible alarm on the effective exposure dose being in excess of the specified threshold), and a work permit was issued for carrying out extra hazardous operations.

After the auxiliary devices were dismantled, the reactor core was lifted up to the exit from the biological shielding plate (Fig. 2a). After being held for the water drain, the reactor core was transferred to the hole above the transport corridor and loaded into a K-1 cask (Fig. 2b), after which the cask was further shut with a safety plug atop of which a locking beam was installed. The reactor core was withdrawn, transferred to the hole in the transport corridor and lowered into the cask remotely using a bridge crane with a TV surveillance system used for monitoring.

A K-1 cask, which had proved to perform well in the course of the previous SM-2 upgrading, was used for the reactor core transportation. The following was undertaken to test the cask serviceability:

- the design was analyzed and tested for compliance with design documentation;
- operating conditions and modes were analyzed;
- the cask integrity was inspected visually;
- liquid penetrant inspection was undertaken for slinging points and welded joints;
- static and dynamic tests were conducted.

The K-1 cask, developed and manufactured at RIAR JSC in 1990, represents a cylindrically shaped reinforced-concrete tank without a bottom. The external diameter of the cask is 2250 mm, the internal diameter is 1450 mm, and the height is 3250 mm. The cask is clad with carbon steel on the outside and lined with stainless steel on the inside. There is a 100mm thick and 1000mm high cast-iron ring at a distance of 450 mm from the cask's lower end between the lining and the cladding. The rest of the volume is filled with concrete of the M200 grade. There is a basket of carbon steel with a thickness of 4 mm (the bottom thickness is 10 mm) and an external diameter of 1445 mm inside the cask.

A ChMZAP-99064 semitrailer truck (Fig. 3) was used to transport the K-1 cask with the loaded reactor core from the reactor facility building to the SRW storage building. The route was developed and the required measures were taken to reinforce the roadbed for the safe movement of the heavy hauler.

After the arrival at the storage building, fasteners were removed from the K-1 cask. Hoisting slings were attached to the cask using the cask servicing platform, and a bridge crane was used to transfer the cask to the storage compartment (Fig. 4).

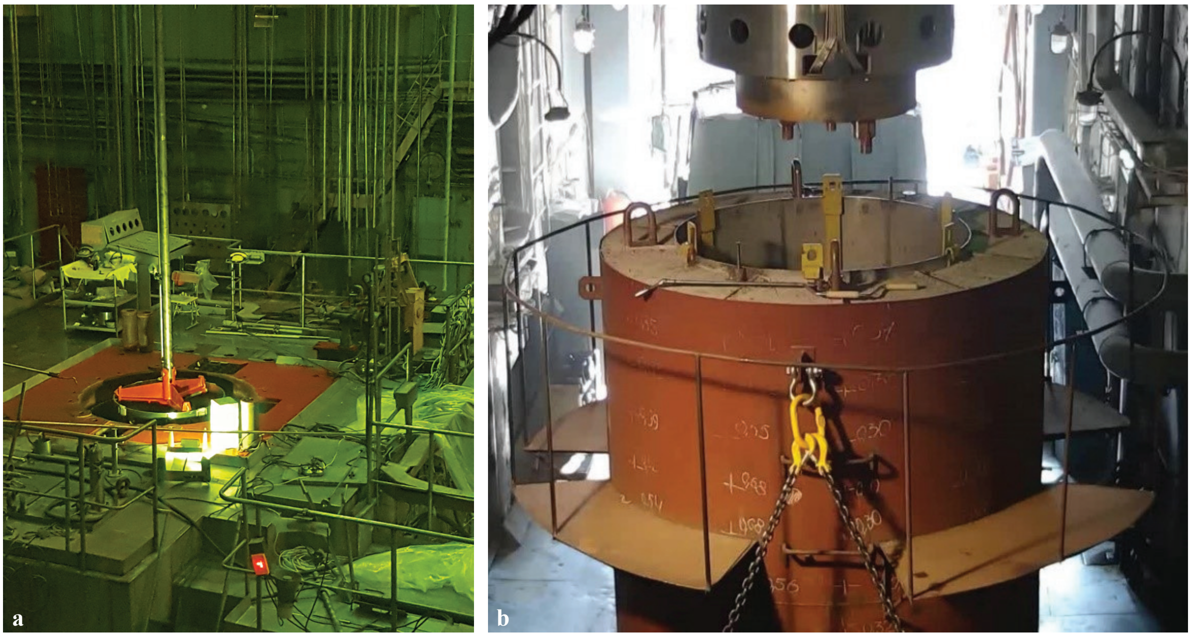

Figure 2. SM-3 reactor core withdrawal. **a.** From the reactor cavity; **b.** Loading into the K-1 cask.

Figure 3. Transportation of the K-1 cask with the loaded reactor core.

Figure 4. K-1 cask positioned above the storage compartment.

After the cask was positioned above the compartment, the slings, the locking beam and the safety plug were removed from the cask using the servicing platform. A bridge crane was used to lower the basket with the reactor core into the storage compartment for long-term storage.

Computational and analytical support of operations

To comply with the principle of ensuring radiation safety (normalization, justification, optimization), the expected radiation exposure for personnel was calculated for all upgrading stages: from the reactor core withdrawal, and the shipping cask slinging, lifting and transfer onto the heavy hauler trailer, to the cask transportation to the SRW storage site and the reactor core placement in the storage compartment. The expected effective exposure doses for personnel were calculated taking into account an analysis of the time during which similar process operations were performed in 1991.

The radiation situation in the period of the reactor core handling is defined by the induced activity of radionuclides caused by the structural material activation process. The key activation products in the reactor core steel are the following radionuclides: ^{60}Co, ^{55}Fe, ^{59}Fe and ^{54}Mn (Gusev et al. 1990). The gamma radiation from ^{55}Fe and ^{54}Mn is "softer" than that from ^{59}Fe and ^{60}Co and will not contribute greatly to the radiation situation if proper shielding is provided. The ^{60}Co gamma spectrum has comparable energy lines with the ^{59}Fe gamma spectrum but the intensity of the ^{60}Co radiation is two times higher. ^{60}Co results from the (n,γ) reaction on the ^{59}Co nucleus. As defined in GOST 5632-2014, the content of ^{59}Co in steel shall not exceed 0.5%$_{mass}$. However, as it is known from literature, e.g. from Gusev et al. 1990, the content of the cobalt impurity in reactor steels does not exceed 0.05%$_{mass}$. The starting content of the cobalt impurity in reactor steels varies in a broad range, this leading to a major uncertainty in estimating the activity of this radionuclide.

The ChainSolver code (Romanov 2018), developed, verified and validated by RIAR JSC's experts, for calculating the nuclei chain transmutation in the process of neutron irradiation in the SM-3 was used to calculate the activity of radionuclides as a result of the SM core structural material activation. For the purpose of calculation, the reactor core was conditionally divided into seven parts, including the upper end components of the beryllium blocks, the beryllium blocks and the vessel, the lower end parts of the beryllium blocks, the upper support plate, the central part components, the lower support plate, and the basket bottom. It was conservatively assumed that the mass fraction of the cobalt impurity in steel is 0.1%$_{mass}$. The calculation results have shown that the radiation situation in the process of the operations for the reactor core dismantling, withdrawal, transportation and placement in long-term storage is defined by the gamma radiation from the ^{60}Co radionuclide (the activity of ^{60}Co in all simulated parts is approximately 10 times as high as the activity of ^{59}Fe). The total activity of ^{60}Co in the reactor core structural components is equal to $(1.0\pm0.2)\cdot10^{15}$ Bq.

Radiation monitoring data obtained in the period of the SM-2 reactor core upgrading in 1991 were used to verify the calculation results for the induced activity of ^{60}Co. To take into account the effect of the reactor core operating time (the SM-2 reactor core was in operation for 13 years and the SM-3 reactor core was in operation for 29 years) on the induced activity value, the dependence of the ^{60}Co specific activity on irradiation time was determined using the ChainSolver code. The obtained results are presented in Fig. 5.

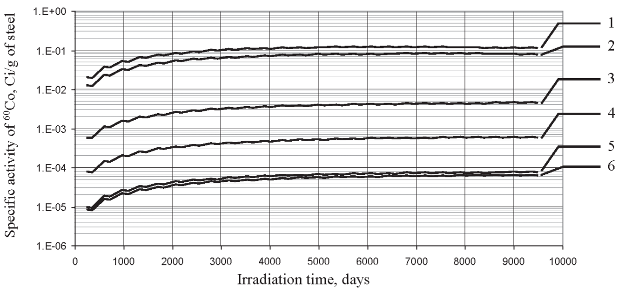

Figure 5. Induced activity of ^{60}Co in the reactor core structural components as a function of irradiation time: 1 – lower end parts of beryllium blocks; 2 – upper support plate; 3 – vessel between elevations of -200 and 200 from the core mid position; 4 – vessel outside elevations of -200 and 200 from the core mid position; 5 – lower end parts of beryllium blocks; 6 – bottom shell.

It follows from the data in Fig. 5 that the induced activity of ^{60}Co in the reactor core structural components reaches equilibrium for the operating time of about 12 years. Further operation of the reactor core does not lead to a major increase in the induced activity of ^{60}Co.

According to the radiation monitoring data from 1991, an EDR of ≈44 mSv/h was recorded on the side surface of the K-1 cask with the reactor core loaded into it. The MicroShield code based on a commonly accepted engineering approach for estimating the dose characteristics of the gamma fields was used to simulate the withdrawn SM-2 reactor core in the form of a homogeneous cylinder with a homogeneously distributed activity. The calculations have shown that an EDR of about 44 mSv/h is reached when the total activity of ^{60}Co is equal to $(1.1\pm0.3)\cdot10^{15}$ Bq, this coinciding with the result obtained using the ChainSolver code within the accuracy limits. It was assumed for further calculations that the total activity of ^{60}Co in the reactor core structural materials is equal to $1.1\cdot10^{15}$ Bq.

The EDR from the K-1 cask with the SM-3 reactor core loaded into it was calculated using the MicroShield code. To take into account the effect of the cast-iron ring on the attenuation of radiation, the reactor core was simulated in the form of seven homogeneous disks that simulated respective parts. The model material and the density of each disk were defined based on analyzing the weight and dimension characteristics and the morphological composition of the simulated part.

The EDR was calculated at points at a height of 1.5 m from the floor, which corresponds to the position of the body of an average person. Two cases were simulated in the process of the EDR calculations: the cask is on the floor surface and the cask is at a height of 0.3 m above the floor. The choice of the two cases is explained by the fact that the cask is designed such that the radiation from the upper end parts of the beryllium blocks with the cask being on the floor surface will not be attenuated by the cast-iron ring radiation unlike the case when the cask is above the floor. Table 1 presents the EDR calculation results depending on the distance between the cask and the area potentially occupied by personnel.

Table 1. EDR calculation results as a function of distance

Cask position	Simulated case	EDR (mSv/h) as a function of distance (m)					
		0	1	4	6	7	8
Cask is on the room floor	Reactor core loading into the SRW storage compartment	39.4	9.9	9.3	4.7	3.5	2.8
Cask is 0.3 m above the floor	Cask transfer inside the transport corridor, cask transportation	–	–	3.6	1.9	1.5	1.1

The dash shows that personnel will be at a distance of not less than four meters in the process of the cask transfer inside the transport corridor and the cask transportation.

Taking into account the fact that the process operations with the K-1 cask are remotely controlled, there was no need for additional biological shielding to be installed on the facility building. However, the process operations inside the SRW storage building required using biological shielding to protect the crane operator workplace. Different versions of the shielding design were considered. The best possible version (both in terms of the radiation attenuation and in terms of the design simplicity) was that with installation of a human-height dovetailed lead brick wall. Table 2 presents the EDR calculation results for the case of the lead brick biological shielding.

Based on the obtained results presented in Table 2, and an analysis of the process and the potential points for the lead brick installation, a decision was made to install additional biological shielding at a distance of about 4 to 5 meters from the potential position of the K-1 cask.

Table 2. EDR calculation results as a function of distance with additional biological shielding used

Cask position	Simulated case	EDR (μSv/h) as a function of distance (m)			
		4	6	7	8
Cask is on the room floor	Reactor core loading into the SRW storage compartment	140	70	49	35
Cask is 0.3 m above the floor	Cask transfer inside the transport corridor, cask transportation	70	40	30	20

Different parts of the personnel bodies will not be irradiated uniformly in the course of the process operations in the K-1 cask servicing area. Three detection points were selected for simulating the personnel feet, body and head. As shown by the obtained results, the expected EDR in the personnel feet region was 120 mSv/h, and that in the body and head regions were 0.5 mSv/h and 0.02 mSv/h.

The expected effective exposure dose for the personnel involved in the process operations inside the buildings was calculated based on data on the time for which similar operations were performed in the course of the SM-2 upgrading in 1991. It was determined based on this that the expected effective dose for the crane operator in the SRW storage building for the entire operations time will amount to ≈36 μSv. The expected effective exposure dose for personnel for the time of one operation in the K-1 cask servicing area (sling/crossbeam removal/installation, locking beam installation onto the cask lid, etc.) was about 80 μSv. Taking into account the number of operations and assuming that all operations will be performed by one person, the total effective exposure dose for the entire operations time will be ≈0.5 mSv.

The expected effective exposure dose for the driver in the process of the K-1 cask transportation from the reactor building to the SRW storage building was calculated taking into account the truck traveling route. There will be two right turns and one left turn in the course of transportation. The distance from the cask to the driver position when traveling in a straight line is ≈11.0 m, this distance to change by 8.0 m to 9.5 m in the event of the left turn, and by 9.2 m to 10.0 m in the event of a right turn. There is direct radiation streaming taking place through the heavy hauler window. The expected EDR in the driver cab will be 1.8 mSv/h when traveling in a straight line, 4.1 mSv/h when turning left, and 2.7 mSv/h when turning right. Given the time spent for the SM-2 reactor core transportation in 1991, the expected effective exposure dose for the driver was estimated to be about 0.7 mSv.

To reduce the driver exposure, it was recommended that additional biological shielding be used. To this end, shielded lead mats with a thickness of 1.1 cm were installed inside the heavy hauler cab. The shielded mat dimensions and installation points were chosen such that to prevent direct radiation streaming through the driver window. Installation of additional biological shielding allowed the expected EDR in the driver cab to be reduced to 0.5 mSv/h when traveling in a straight line, to 0.8 mSv/h when turning left, and to 0.6 mSv/h when turning right. The additional shielding for the driver cab will reduce the expected effective exposure dose for the driver to 0.2 mSv.

Based on the above, a conclusion was made that no personnel exposure in excess of 20 mSv (the key dose limit for group A personnel as per NRB-99/2009 is expected in the course of process operations to dismantle, withdraw, transport and place the SM-3 reactor core into the compartment for long-term storage.

Radiation monitoring results

The radiation parameters were monitored continuously during the operations by the reactor building radiation monitoring systems and by the onsite ARSMS. Fig. 6 shows the positions of the radiation monitoring points (RMP) for the reactor building, and Fig. 7 shows the ARSMS points.

Table 3 presents the maximum recorded gamma EDR values at the key monitoring points inside the reactor building, and Table 4 presents the maximum onsite ARSMS readings. The buffer and supervised area ARSMS readings did not exceed the background values throughout the process.

The gamma EDR in the trailer truck driver cab, both immediately against and at a distance of 1 m from the cask with the reactor core loaded into it, was measured using an MKS-AT1117M radiation dosimeter with a BDKG-04 detection unit. The measurement results for the K-1 cask with the reactor core inside being on the trailer truck are presented in Table 5.

Table 6 presents effective exposure doses for the personnel involved obtained based on analyzing personal dosimeters after the operations.

Discussion of results

It follows from the calculation results that operation of the SM reactor core for more than 12 years leads to the ^{60}Co induced activity in structural materials being in an equilibrium state. Such conclusion makes it possible to confirm (in the accuracy limits) the results of calculating the induced activity of ^{60}Co in the reactor core components, obtained in solving the nuclide kinetics problem using the

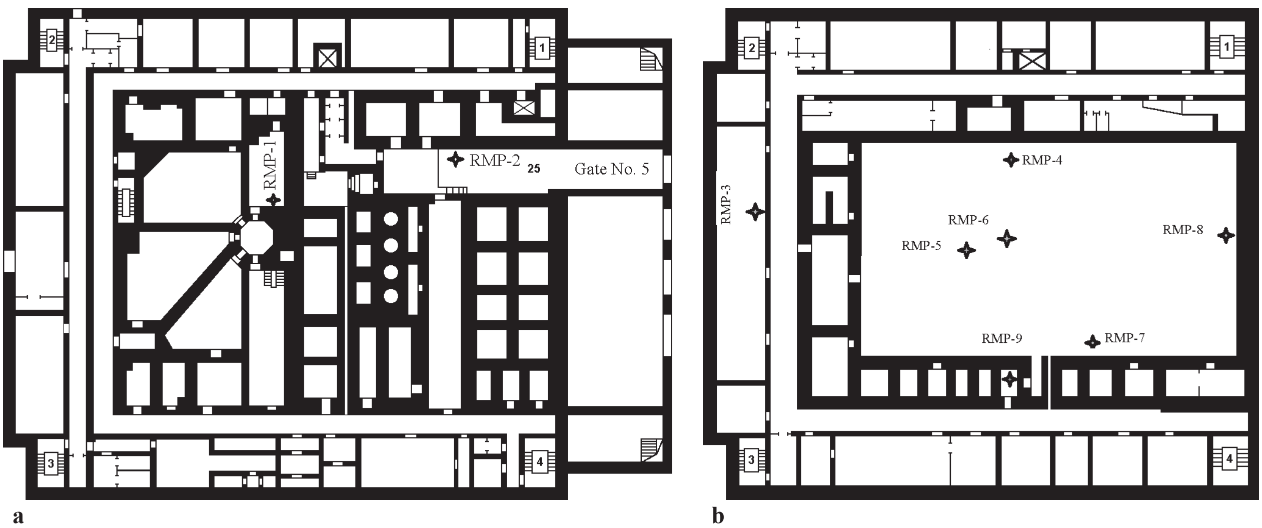

Figure 6. Positions of the radiation monitoring points inside the reactor building: **a.** Elevation +0.0 m; **b.** Elevations +8.4 m and 13.2 m.

Table 3. Maximum readings from the radiation situation monitoring points

No.	Reactor core position	Time	EDR, mSv/h								
			RMP-1	RMP-6	RMP-9	RMP-7	RMP-8	RMP-4	RMP-2	RMP-1	RMP-3
1	Cavity	8:30	0.35	4.0	0.0012	0.075	0.055	0.20	0.04	0.001	0.0001
2	Biological plate level	10:33	9000	100	0.0015	74	50	200	0.04	0.001	0.0016
	Distance between reactor core and monitoring point detection unit (BDMG-08R-05), m		1.5	–	–	19.5	32.5	12.5	–	–	–
3	Opposite the hot cell (HC)	10:50	110	4.8	0.0014	330	93	400	0.08	0.001	0.0013
4	Above hole in corridor No. 25	11:00	20	4.3	0.0013	88	300	80	48	0.001	0.0012
5	K-1 cask (corridor No. 25)	11:10	0.8	4.3	0.0012	0.5	0.75	0.35	2.8	0.001	0.0001
6	Inside K-1 cask shut with plug (corridor No. 25)	12:30	0.35	3.3	0.0012	0.1	0.14	0.22	2.0	0.001	0.0001
7	Behind gate No. 5, building 106	12:49	0.33	3.6	0.0012	0.08	0.058	0.20	0.04	0.001	0.0001

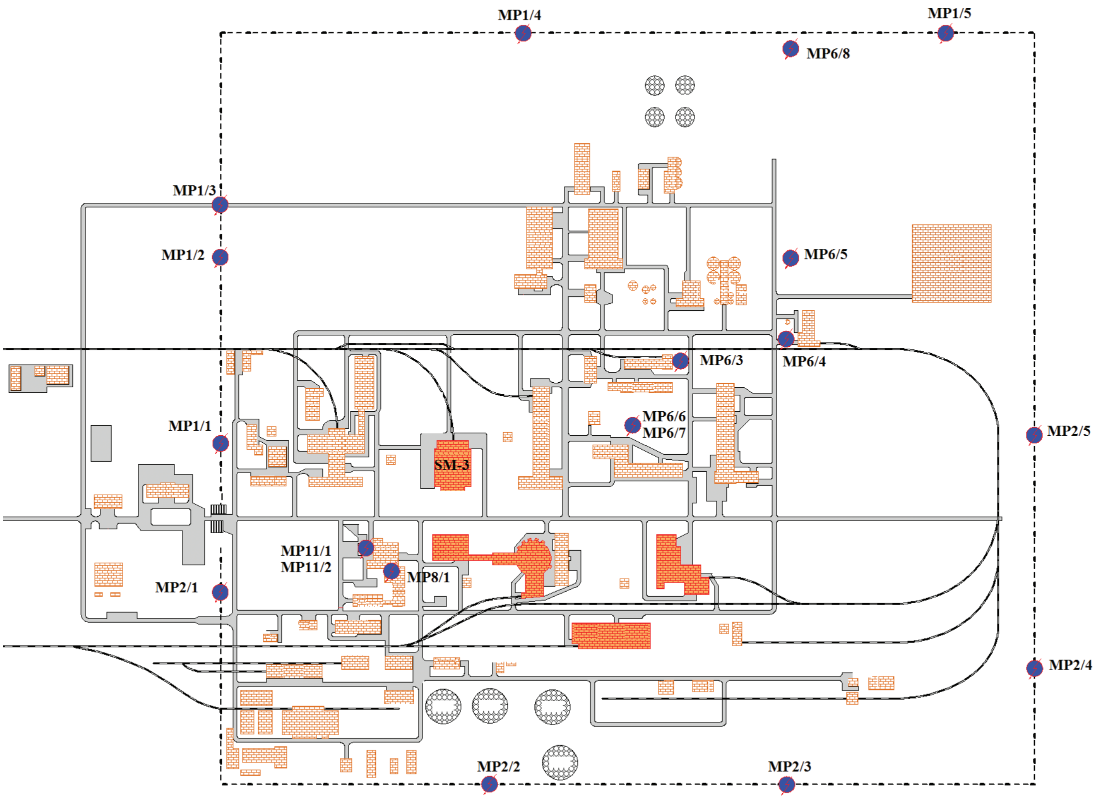

Figure 7. Arrangement of onsite ARSMS posts.

Table 4. Maximum readings from the ARSMS monitoring points

Monitoring post	Max. value, μSv/h	Max. value recording time
MP 1/1	0.34	12.10.2019 10:46:16
MP 1/2	0.15	12.10.2019 10:50:35
MP 1/3	0.11	12.10.2019 10:44:06
MP 1/4	0.10	12.10.2019 10:48:26
MP 1/5	0.07	12.10.2019 11:01:23
MP 2/1	0.18	12.10.2019 10:35:29
MP 2/2	0.18	12.10.2019 10:52:45
MP 2/3	0.08	12.10.2019 10:54:55
MP 2/4	0.06	12.10.2019 11:55:23
MP 2/5	0.07	12.10.2019 11:51:04
MP 8/1	10.63	12.10.2019 10:39:48
MP 6/3	0.34	12.10.2019 10:35:29
MP 6/4	0.26	12.10.2019 10:52:45
MP 6/5	0.09	12.10.2019 11:33:47
MP 11/1	0.24	12.10.2019 10:44:06
MP 6/8	0.06	12.10.2019 11:51:04

ChainSolver code by way of the reactor core simulation and by normalizing the calculated value of the EDR created by the unit activity, against the results of radiation monitoring in the process of the SM-2 upgrading in 1991.

The MicroShield-estimated EDR from the K-1 cask with the SM-3 reactor core inside have turned out to be underestimated noticeably (by a factor of about ≈3.5) as compared with the radiation monitoring results. According to the calculation results, the expected EDR at a height of 1.5 m above the bottom immediately near the cask is ≈40 mSv/h, and that obtained as a result of radiation monitoring is 140 mSv/h. However, as shown by the calculation results, the expected EDR in the driver cab was 0.5 mSv/h, which agrees with the radiation monitoring results (0.35 mSv/h). An analysis of the calculation and radiation monitoring results demonstrate that the most probable cause for the disagreement between the estimates and the radiation monitoring results on the K-1 cask surface is a downward bias in the floor-reflected radiation estimates. Actually, the radiation monitoring results exceed the calculation results in the immediate vicinity of the cask, i.e. in conditions when the reflected radiation contributes greatly to the

Table 5. Results of the cask radiation monitoring in the process of transportation

Detection point	EDR, mSv/h			
	Cask with safety plug		Cask without safety plug	
Driver cab	0.4		0.7	
Under cask (immediately near)	2000		2000	
Under cask (road level)	360		360	
Cask	Immediately near	1 meter off	Immediately near	1 meter off
+0.0 m from bottom level	20	12	20	12
+0.8 m from bottom level	140	45	148	50
+1.6 m from bottom level	90	35	114	68
+2.4 m from bottom level	4.9	10	63	46
+3.2 m from bottom level	8	5	240	65

Table 6. Effective exposure doses for personnel involved

Subdivision	Number of persons	Effective dose	
		Average, mSv	Collective, 10^{-3} pers.-Sv
Management	3	1.61	3.22
Reactor operation control	2	0.83	1.66
Transport technologies team	16	0.29	4.56
Management of radiation safety	3	0.86	2.57
Driver	1	0.05	0.05

radiation situation, and the radiation monitoring results for the driver cab, when the reflected radiation stops to contribute greatly to the radiation situation, coincide with the calculation results (in the limits of ≈40%).

It follows from comparing the expected effective exposure dose for the driver and the personal radiation monitoring results that the estimates have turned out to be much higher (by a factor of about 4). The causes for such disagreement are two factors: first, the estimated driver cab EDR is higher than the observed EDR, and, second, the time of the K-1 cask transportation from the reactor building to the SRW storage building is half as short as the time expected and conservatively assumed in the calculations.

The maximum expected effective exposure dose for transport technology experts was 0.2 mSv provided that all handling operations are performed by one person. Based on the personal radiation monitoring results, the average effective exposure dose for transport technology experts was 0.3 mSv which exceeds the calculated value. This disagreement has also been caused by a downward bias in the reflected radiation contribution estimate.

It follows from the personal radiation monitoring results that no exposure in excess of the base dose limits was observed for the personnel involved in the operations to dismantle, withdraw, transport and place the SM-3 reactor core in long-term storage. Personal effective exposure doses in excess of over 1 mSv was observed in the process of operations in the event of three employees in the executive staff category: 1.9 mSv for one employee, and 1.5 mSv for two employees. An exposure of over 1 mSv was observed in the process of similar operations in 1991 for three employees: the personal effective exposure dose was 2.5 mSv for each. An analysis of the work progress in 1991 and the SM-3 reactor core simulator training made it possible to reduce the duration of the most hazardous operations which, combined with the additional biological shielding used, led to reduced personnel exposure as compared with the reactor upgrading activities in 1991.

A note shall be specifically made of the fact that the exposure for personnel in the executive staff category was much in excess of the exposure for the personnel involved immediately in the handling operations. The explanation for this fact is that the executive staff members supervised the process being at a relatively safe distance from the reactor core but throughout the process. At the same time, engineering and technical personnel was removed from the radiation exposure area after they completed their functions.

It follows from analyzing the ARSMS sensor readings that the radiation situation within the institute's commercial site was normal. Readings in excess of the background level was recorded only at monitoring post (MP) 8/1, with the EDR value amounting to 49% of the emergency setpoint of 22 μSv/h. This monitoring post is directly within the visual range from the reactor building's vehicle entrance. The maximum EDR observation time for this monitoring post coincides with the time the trailer truck carrying the K-1 cask with the reactor core inside was leaving the building through the vehicle entrance. An analysis of the ARSMS sensor readings in the process of transportation suggests as well that the best possible route was selected for minimizing the exposure for the institute's shift personnel. No ARSMS sensor readings were recorded to be in excess of the background value within the buffer area and the supervised area. The dose rate throughout the site did not exceed the reference level defined for nonstandard conditions (100 μSv/h) for the entire time of the operation for the reactor core disposal.

The transportation routes were radiologically surveyed after the operations were completed, as the result of which no radioactive site contamination was recorded as shown by the radiological survey results.

Conclusions

The operations to dismantle and transport the SM-3 nuclear research reactor core and place it in long-term storage were performed in accordance with the prepared procedures for organizing and carrying out activities. No personnel exposure in excess of the limits defined in regulatory documentation was observed in the course of the operations, and no radioactive site contamination was recorded along the transportation route of the K-1 cask with the SM-3 reactor core inside.

Preparatory activities, drills and the amendments made to the designs made it possible to reduce the time required for the operations and avoid deviations from normal conditions in the process of disposal.

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