IRSTI 58.81.91 https://doi.org/10.26577/phst20261311

Radiation safety justification of spent nuclear fuel management technology for a reactor facility

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(Received February 11, 2026; received in revised form May 25, 2026; accepted May 31, 2026)

Research at the Institute of Atomic Energy of the National Nuclear Center of the Republic of Kazakhstan (IAE NNC RK) is focused on developing effective technologies for the management of spent nuclear fuel (SNF) from the BN-350 fast reactor. Among the approaches under consideration is a dilution and immobilization technology developed for irradiated uranium–graphite fuel from the Impulse Graphite Reactor (IGR). The process involves dry mixing of highly enriched uranium (HEU) fuel with depleted uranium dioxide, followed by immobilization through cementation. This study presents a comprehensive assessment of radiation conditions throughout the technological cycle of HEU fuel processing. Radiation field analysis at each processing stage is essential for establishing design requirements, optimizing facility operation, and ensuring personnel protection. Monte Carlo simulations performed using the MCNP code were employed to evaluate dose distributions and support radiation safety justification of the facility design. Based on the simulation results, several design modifications were implemented to improve shielding effectiveness and reduce occupational exposure. The proposed approach introduces a systematic methodology for radiation field assessment under conditions of incomplete information on fuel distribution, providing a conservative framework for safety analysis. Given the absence of direct international analogues for the management of irradiated HEU graphite fuel, the developed methodology offers both practical value for the IGR fuel immobilization project and potential applicability to future SNF management technologies for the BN-350 reactor.

Keywords: spent nuclear fuel, management, measured effective dose, radiation situation.

1. Introduction

Currently, the IAE NNC RK is designing a fuel dilution and immobilization facility (hereinafter referred to as the facility) for the IGR research reactor [13]. This facility is planned to process the IGR reactor fuel, which was unloaded in 1966. The main facility areas include temporary storage for daily fuel batches, the immobilization area (fuel and "dilutant" grinding and cementation process), and the barrel drying area (matrix solution hardening process).

According to international recommendations for the design and operation of processing facilities [14], all processes must comply with the basic safety principles: protection of people, prevention of nuclear materials criticality, control of emissions, and environmental protection. The facility operation requires a multi-level system of protective barriers (defense in depth), with emphasis on the ALARA principle: keeping radiation doses to personnel and the public "as low as reasonably achievable." The assessment of radiological impact associated with different fuel management options (e.g., reprocessing versus direct disposal) is well presented in [15]. The article [16] describes a spent nuclear fuel reprocessing plant divided into four production areas and presents methods for assessing the risks associated with the reprocessing process. These methods make it possible to predict which areas pose the greatest hazards and therefore require enhanced control and protection. Fuzzy comprehensive assessment and probabilistic statistical analysis methods are used to calculate the risk level for each assessment unit and each production area.

For the monitoring and control of operating nuclear reactors, various detection modules are being developed [5]. In study [6], the relationship between radon accumulation in indoor environments and the concentration of its decay products in soil is investigated in order to provide a comprehensive assessment of public health impacts. The article [7] is devoted to the development of a low-cost alpha and gamma spectrometric system for experimental physics and field applications, such as radiation detection and environmental monitoring of nuclear facilities.

The article [8] presents a method and a framework for optimizing the allocation of radiation monitoring resources based on an analysis of nuclear incident risks and the relative importance of facilities at a spent nuclear fuel (SNF) reprocessing plant. The study provides a theoretical and technical basis for safety management and operational decision-making at SNF reprocessing plants.

In recent years, significant progress has been achieved in the development of nanostructured materials within the framework of modern physical research. In particular, carbon-containing nanomaterials synthesized via electric arc discharge exhibit a high degree of graphitization and a nanoscale structure (on the order of tens of nanometers), leading to enhanced thermal stability and electrical conductivity [9]. Such properties make them promising candidates for application in radiation-resistant systems and functional components operating under extreme conditions.

Moreover, studies on carbon nanowalls have shown that their morphology and surface structure can be precisely controlled, which directly affects their physicochemical properties. This tunability is particularly important for processes involving interaction with radionuclides, including sorption, surface binding, and barrier functions [10]. The high specific surface area of such nanostructures further expands their potential applicability in nuclear technologies.

The article [11] presents calculation approaches for estimating personnel radiation doses during the following technological stages: unloading, dismantling, and processing. The calculations accounted for neutron and photon sources SNF, photon sources in control rods, and photon sources in structural materials, including the reactor vessel and fuel assemblies. When calculating dose rates from neutrons and secondary gamma photons, direct calculations using MCNP-4B produced acceptable results in most cases, with an acceptable level of methodological uncertainty. However, for problems involving photon sources, direct MCNP-4B calculations yielded unsatisfactory results due to strong radiation attenuation. To reduce statistical dispersion, several variance-reduction techniques were applied, including the assignment of different importance values to cells and the use of the weighted window iteration method.

When assessing the safety of a reprocessing facility, it is also necessary to take into account the national context: what standards and legislative features are in place in the country. Article [12] examines issues of state regulation of safety, as well as the policy and practice of handling accumulated and generated radioactive waste in the Republic of Kazakhstan. The Republic of Kazakhstan does not have industrial facilities for reprocessing SNF, and the main focus is on the safe storage, conditioning, and accounting of nuclear materials. The developed infrastructure of the NNC RK and international cooperation provide support for the radiation monitoring system and the improvement of the regulatory and technical framework. The national approach is based on the principles of multi-level protection and risk assessment to ensure long-term nuclear and radiation safety.

The present study is based on Monte Carlo simulations performed to assess radiation conditions in various areas of the facility and to establish requirements for the design, construction, and operation of process equipment, as well as to ensure adequate radiation protection of personnel. A particular challenge of the considered system is the uncertainty associated with the spatial distribution of irradiated fuel within the technological equipment. To address this issue, a set of conservative fuel distribution scenarios was developed and analyzed.

The calculations made it possible to identify the most unfavorable operating conditions, determine the factors having the greatest influence on dose formation, and evaluate the effectiveness of the proposed shielding solutions. The results were subsequently used to support design decisions and optimize the radiation protection measures incorporated into the facility project. The proposed approach may also be useful for radiation safety assessments of other facilities involved in the handling and processing of spent nuclear fuel under conditions where the exact distribution of radioactive materials cannot be determined with sufficient accuracy.

2. Methodology

The irradiated graphite fuel from the IGR reactor has an enrichment of up to 90% in ^{235}U, which is classified as highly enriched uranium (HEU), and in order to remove the material from IAEA safeguards, it is planned to reduce the fuel enrichment in ^{235}U to 5%. The dry mixing method was chosen as the design solution for the HEU fuel dilution stage. This method involves the joint grinding of the fuel and depleted uranium dioxide to a fraction of ≈200 μm in specified ratios. Next, the diluted fuel is immobilized by cementation. Cementation of irradiated, now low-enriched uranium (LEU) fuel is carried out in a type "A" barrel (55 gallons) with a built-in mixer, in accordance with the developed technological regulations and using the established matrix recipe [13]. The concentration of fuel in the cement matrix does not exceed 50 ppm of ^{235}U [14,15].

During operation (from 1962 to 1966), the fuel received a dose of 4 × 10^{18} n/cm^2. The maximum temperature during reactor operation was approximately 2000 K, and the fuel burnup was less than 1%. Irradiated elements are expected to be handled in the form of blocks (Figure 1) and rods, as well as fragments thereof.

The estimated radionuclide composition and activity of the fuel are presented in Table 1.

Table 1. Number and activity of nuclides in irradiated graphite HEU fuel from the IGR reactor.

№	Nuclide	N (number of nuclei)	Main radiation type	Q (activity), sec^{-1}
1	^{137}Cs	$9.65 \cdot 10^{20}$	β, γ	$7.03 \cdot 10^{11}$
2	^{90}Sr	$8.00 \cdot 10^{20}$	B	$6.15 \cdot 10^{11}$
3	^{151}Sm	$9.03 \cdot 10^{19}$	B	$2.20 \cdot 10^{10}$
4	^{99}Tc	$2.71 \cdot 10^{21}$	B	$2.21 \cdot 10^{8}$
5	^{155}Eu	$1.36 \cdot 10^{16}$	β, γ	$6.04 \cdot 10^{7}$
6	^{93}Zr	$3.11 \cdot 10^{21}$	B	$4.38 \cdot 10^{7}$
7	^{135}Cs	$3.00 \cdot 10^{21}$	B	$2.86 \cdot 10^{7}$
8	^{129}I	$5.93 \cdot 10^{20}$	β, γ	$8.30 \cdot 10^{5}$
9	^{107}Pd	$1.30 \cdot 10^{20}$	B	$4.20 \cdot 10^{5}$

Among the radionuclides listed in the table, only ^{137}Cs exhibits both a high gamma-ray emission yield and significant activity, making it the dominant source of gamma radiation. Its activity is 7.03×10^{11} s^{-1}, which is the highest among the radionuclides considered. During the decay of ^{137}Cs, metastable ^{137m}Ba is produced, emitting gamma photons with an energy of approximately 662 keV. The remaining radionuclides are predominantly beta emitters or possess insufficient gamma activity to make a significant contribution to the radiation field.

The measured effective dose (MED) of gamma radiation on the surface of uranium-graphite blocks stored in the storage is approximately 2 - 2.5 μSv/h, and at a distance of 10 cm from a single block of irradiated fuel, it is approximately 0.33 μSv/h. The nuclide ^{137}Cs contributes most significantly to the gamma radiation level. The activity of ^{137}Cs in a single graphite block weighing 2 kg was assumed to be 10^8 Bq.

To carry out the immobilization and dilution process, preliminary work was performed to repackage the spent fuel into daily containers. A total of 640 blocks were extracted and repackaged.

The technological scheme for diluting HEU fuel: first, uranium-graphite blocks are pre-crushed and ground. For this purpose, a jaw crusher will be used, which allows the entire irradiated block to be loaded and ensures grinding to fractions of 25–5 mm. Further grinding of the fuel is carried out together with depleted uranium dioxide in a ball mill or rotary fine grinding mill. The next stage involves cementing, in which the proportion of solid radioactive waste introduced must not exceed 20%. The final stage involves pouring and mixing the cement mixture into containers – 200-liter barrels – and then transporting them to specialized storage facilities.

Figure 1 shows the design of the facility. The rooms in the planned building are divided into zones with different levels of radiation safety, each with corresponding access restrictions. The building design have been developed taking into account radiation protection requirements, including the use of protective barriers and structural elements to minimize radiation exposure. HEU fuel will be stored in special containers and equipment at all stages of operation; open handling is not permitted. The technological equipment is designed to be reliable and convenient to operate, made of corrosion-resistant and radiation-resistant materials that can be decontaminated, has the necessary tightness, and provides the possibility of using remote methods for equipment operation and monitoring. Figure 1 shows rather the technological chains in the process of processing blocks.

According to the project, personnel are restricted to spending only limited amounts of time in rooms with high radiation levels in order to ensure that permissible exposure levels are not exceeded.

This paper considers a radiation scenario describing the full processing cycle: from loading the graphite block into the crusher to placing the cemented LEU fuel into a container. At all stages of the cycle: grinding, mixing, cementation will be conducted on grinding installation (Fig. 2).

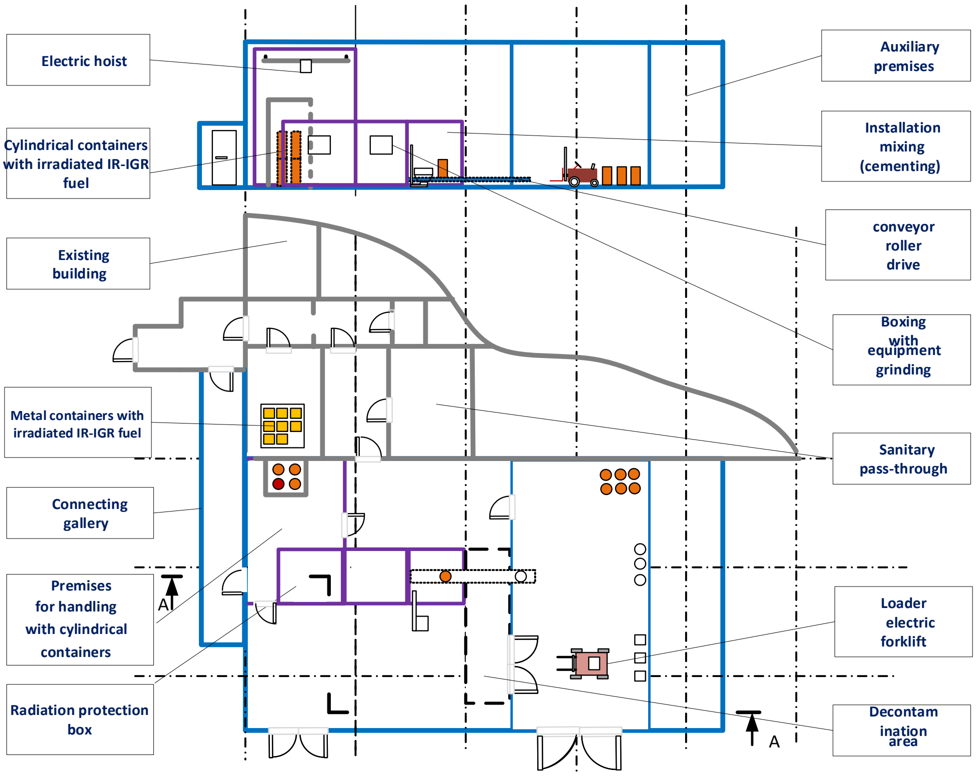
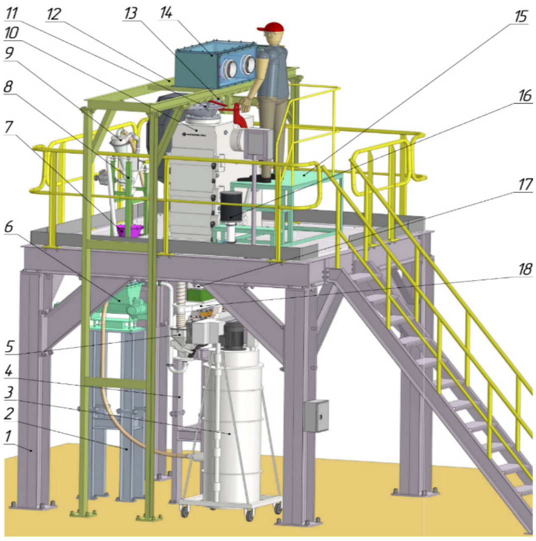


Figure 1. Design of HEU fuel dilution and immobilization facility.



1 – service area; 2 – weight dispenser stand; 3 – turbine vacuum cleaner; 4 – feeder and mill support; 5 – universal mill (point 3) (A4); 6 – weight dispenser (point 4) (A8); 7 – storage hopper (A6); 8 – cyclone support; 9 – double cyclone (A5); 10 – jaw crusher (A2); 11 – rotary valve; 12 – loading chamber support; 13 – adapter; 14 – feed chamber (point 1) (A1); 15 – pedestal; 16 – air filter; 17 – chute; 18 – electromagnetic dosing feeder (point 2) (A3).

Figure 2. Grinding installation.

According to the technology, only two graphite blocks will be processed per cycle, due to activity and permissible dose limits. However, the most complex scenario takes into account the potential for graphite dust to remain in various parts of the equipment.

This discretization is not arbitrary, but reflects the actual dynamics of fuel transport and accumulation in the system. This demonstrates the reliability of the model's conservatism and the representativeness of the results.

During the calculations, the radiation sources were specified in volumetric form as entire blocks on the production line and as volumetric cylindrical sources in the storage warehouse (barrels). This has been corrected in the article.

Systematic uncertainties were taken into account when defining the geometry. The fuel position was selected at five points, corresponding to the maximum amount of fuel transported within the facility. It was conservatively assumed that two fuel blocks remain in the facility at all times as dust deposited on the internal equipment. Therefore, it was assumed that 0.5 blocks were always present at each calculation point.

The facility was not modeled; the calculation geometry included only radiation shielding elements (glass and screens) and building structures, specifically the room walls and metal barrels, which significantly impact dose distribution.

Uncertainties in the fuel composition were also considered when developing the scenario (graphite was specified without additional impurities that have an absorbing effect), and the activity of a single graphite block was chosen to be maximum. Various scenarios were considered, taking into account the uncertainty in the geometry, shielding, and fuel distribution within the facility.

When conducting calculations, the facility was not modeled in detail as having complex industrial geometry. This is due to conservatism; all the facility components are made of stainless steel, which serves as an additional absorbing element. However, when constructing the models, we took into account that this could affect scattering, shielding, and the overall formation of dose fields.

The adopted approach represents a generalized methodology for handling incomplete source term information, where discrete conservative scenarios are used to approximate the envelope of possible fuel distributions.

Such an approach allows the identification of bounding radiation conditions without requiring detailed real-time material tracking, which is often unavailable at early design stages.

3. Calculation methodology

The calculations were performed in 3D geometry using the MCNP6 code. Conservative approaches and assumptions were applied in setting up the tasks and modeling. The Visual Editor and Excel were used for visualization.

Due to the lack of detailed data on fuel distribution at each point of the grinding installation, the following assumptions were made to simplify the calculation. The fuel distribution was evaluated at five key locations corresponding to the maximum amount of fuel during its movement through the facility: point 1 – the loading throat of the jaw crusher; point 2 – the electromagnetic metering feeder; point 3 – the mill; point 4 – the receiving hopper of the weigh feeder; point 0 – the bottom of the barrel.

A conservative assumption was made that two fuel blocks that two fuel blocks, in the form of dust settled on the internal equipment, remain in the installation at all times. Thus, it is assumed that 0.5 blocks are constantly present at each calculation point. Table 2 presents five scenarios for fuel placement at the service area and site.

Table 2. HEU fuel location in the installation of fuel dilution and immobilization.

Scenario	Option				
	Point 1	Point 2	Point 3	Point 4	Point 0 (Barrel)
1	2.5 block	0.5 block	0.5 block	0.5 block	-
2	1.5 block	1.5 block	0.5 block	0.5 block	-
3	0.5 block	1.5 block	1.5 block	0.5 block	-
4	0.5 block	0.5 block	0.5 block	2.5 block	-
5	0.5 block	0.5 block	0.5 block	0.5 block	2 blocks evenly distributed in the cement matrix

For dose field calculations in MCNP, the F4 tally was employed, followed by dose conversion from particle flux to effective dose. The conversion was performed using photon fluence-to-effective dose coefficients for air corresponding to the anterior–posterior irradiation geometry (AP, eng). The adopted effective dose conversion coefficients were taken from *On the Approval of Hygienic Standards for Radiation Safety Requirements* (02/08/2022).

Photon libraries *mcplib84,* included in MCNP6 and based on the ENDF/B-VI.8 nuclear data files, were used in the calculations. The selection of this library was motivated by its status as a standard and validated dataset, ensuring reproducibility and comparability of the simulation results.

Dose calculations were performed using the MCNP code (Monte Carlo method) [16, 17]. The models were constructed considering the area where the equipment is located and the facility as a whole. For a conservative assessment, the installation itself was not explicitly modeled was not modeled, but radiation protection elements and building structures that have a significant impact on dose distribution were included in the calculated geometry.

During the modeling, a minimal number of objects was specified – only walls, blocks, and, in the second version of the project, radiation protection screens.

The radiation sources were specified as volume sources with constant energy characteristics. Five scenarios were considered, differing in the position of the sources and the configuration of the protective elements. For each scenario, the dose fields at the service area and the site as a whole were calculated. For each scenario, at least 10^{12} particles were simulated, which ensured a statistical error of no more than 10^{-3} near the sources.

The first version of the project assumed the absence of radiation-protective screens near the grinding unit equipment, and the absence of mobile radiation-protective screens when the barrel is full at the mixing unit and in the holding area.

4. Results and discussion

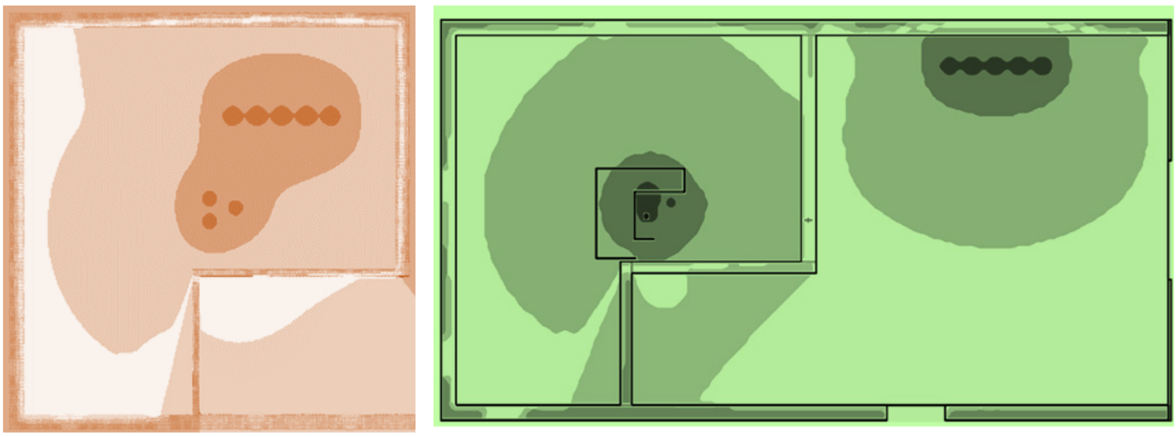
Nuclear safety at the fuel dilution and immobilization facility is ensured by limiting the number of fissile materials based on its isotopic composition. In accordance with [18], the designed building is not classified as a nuclear hazard area, since the total mass of ^{235}U present at any given time in the production facilities where nuclear materials are handled does not exceed 300 g.

To analyze the radiation situation within the facility, two volumetric dose fields were constructed: one at floor level “0” and the other at service area level “+3” (1.5 m above floor level). The calculation results made it possible to identify areas with increased radiation and formed the basis for the design of a multilayer radiation protection system located in the most vulnerable areas of radiation propagation.

The results are presented in figures 3a – 8a in the form of effective dose distribution maps constructed on the basis of dose function data [19] in planes corresponding to working levels. As a result of calculations for areas where work with HEU fuel is carried out, it was proposed to strengthen radiation protection measures and use the following to protect personnel:

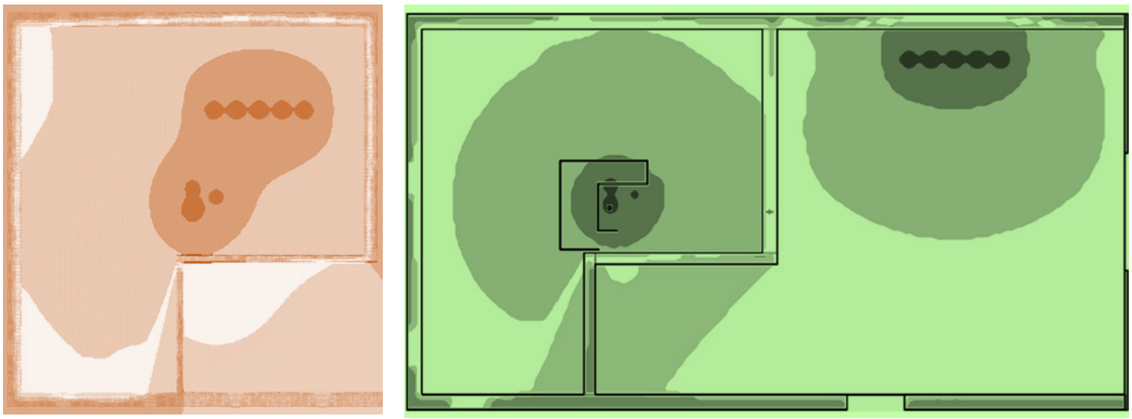
- from the grinding equipment - RZM Abris-1, RZM Abris-2, RZM Abris-3 radiation protection mats;
- from the filled barrel at the mixing unit - ZS-311A mobile radiation protection screens;
- for filled barrels at the filled barrel storage area - ZS-321A mobile radiation protection screens.

The project also provides for the organization of dosimetric control. The effective dose distribution maps after strengthening protective measures are shown in Figures 3 and 8 with index b. In Figures 3-8, a darker shade of color corresponds to a higher dose value.



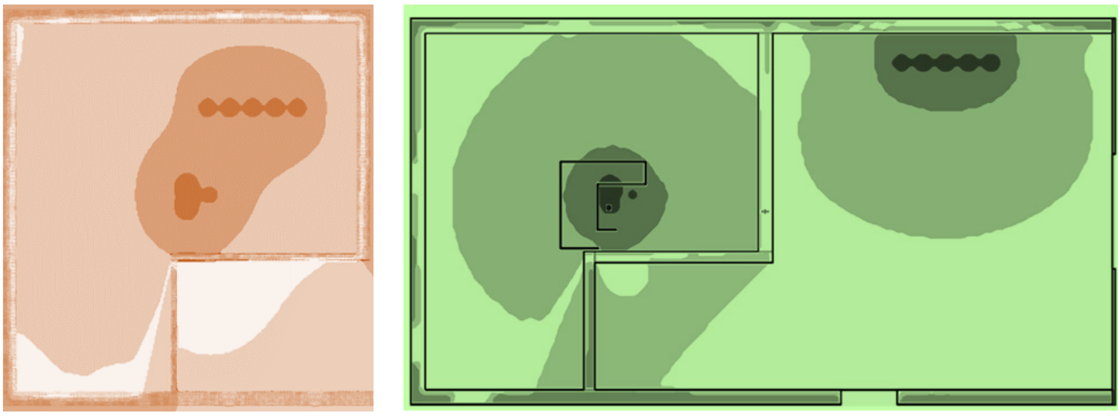
a) b)

Figure 3. Scenario 1: a) initial version; b) final version.



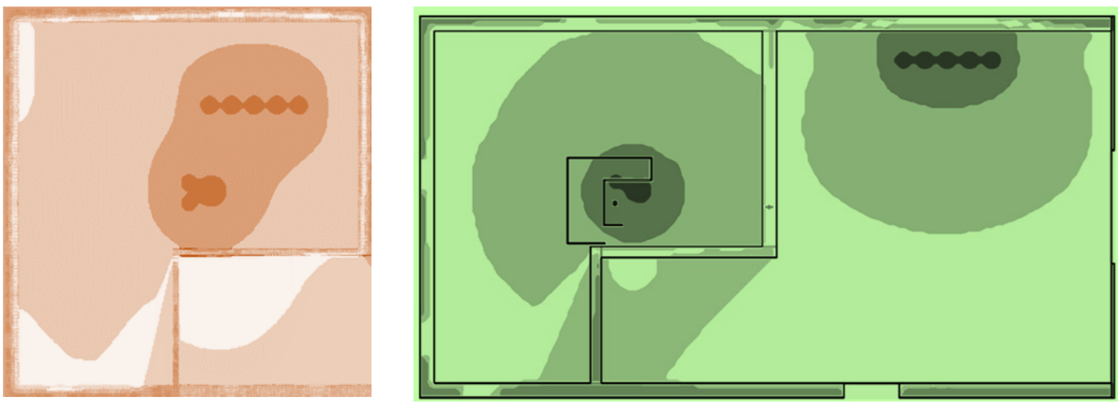
a) b)

Figure 4. Scenario 2: a) initial version; b) final version.



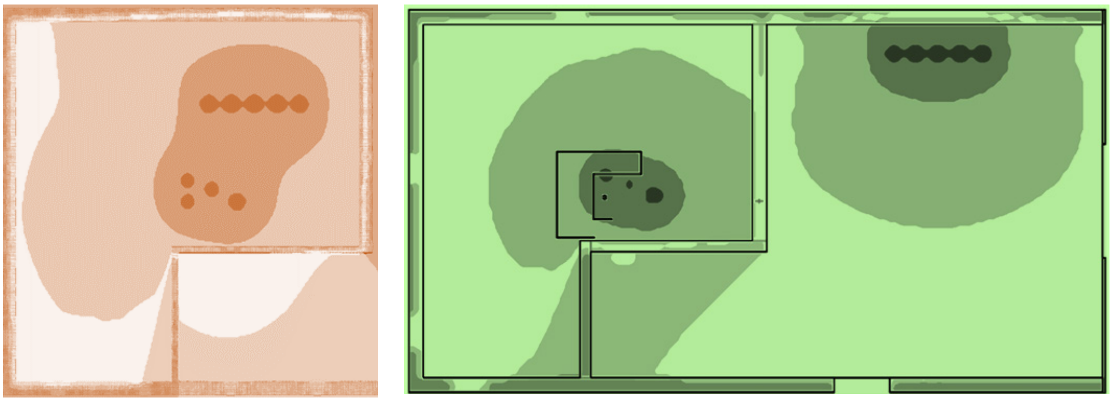
a) b)

Figure 5. Scenario 3: a) initial version; b) final version.



a) b)

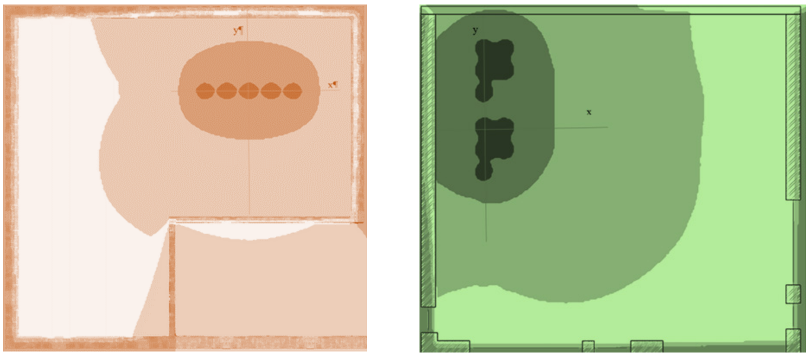
Figure 6. Scenario 4: a) initial version; b) final version.



a) b)

Figure 7. Scenario 5: a) initial version; b) final version.

The initial version of the project provided the placement of a total of 15 barrels containing cemented LEU fuel. After the calculations, the number of barrels was reduced to 5, and they were relocated to a separate room with additional lead plate shielding. Figure 8 shows the arrangement of barrels with cemented LEU fuel in the initial design.



a) b)

Figure 8. Initial placement of barrels containing cemented LEU fuel: a) central hall; b) barrel storage room.

Table 3. Maximum MED value in key positions, μSv/h.

Distance, m	Scenarios for the simultaneous presence of spent nuclear fuel at points				
	1	2	3	4	5
	Mixing barrel				
0,02 (close to the barrel)	7.14	8.62	10.10	18.49	122.00
0,5 (from the barrel)	4.55	5.38	6.20	9.69	17.64
1 (from the barrel)	3.28	3.78	4.27	6.07	7.64
2 (from the barrel)	2.04	2.26	2.47	3.15	3.01
3 (from the barrel)	1.48	1.60	1.71	2.06	1.79
4 (from the barrel)	0.81	0.81	0.81	0.81	0.81
	Service area for the grinding plant				
0,02 (inside the loading chamber)	5137.56	3102.08	1056.90	1049.10	1038.55

Continuation of the table

Distance, m		Scenarios for the simultaneous presence of spent nuclear fuel at points				
		1	2	3	4	5
0,02 (close to the loading chamber)		194.94	134.55	65.73	56.26	48.73
0,5 (from the loading chamber)		31.16	27.91	21.47	16.51	11.95
1 (from the loading chamber)		13.26	13.16	11.97	8.90	6.29
2 (from the loading chamber)		4.73	4.95	4.90	3.90	2.65
Boundary of the site						
In front of the wall (along the perimeter of the room 32)		0.98	1.02	1.12	1.13	0.82
Behind the wall (along the perimeter of room 32)		0.0028	0.0032	0.004	0.0039	0.0025
Behind the leaded windows		0.66	1.20	1.46	1.99	1.13
Behind the partition wall		0.05	0.09	0.12	0.05	0.05
Center of the site						
Zone A		1041.69	1057.63	3106.78	**5136.69**	1045.22
Zone B	On the border of the first RPM	835.22	846.67	**2489.22**	854.43	835.90
	After the first RPM	407.83	417.94	**1210.07**	424.01	408.87
Zone C	Before the second RPM	11.05	13.40	**23.69**	19.99	11.34
	After the second RPM	9.67	11.83	**20.78**	17.72	9.91

* *RPM – On the border of the first radiation protection mats*

The first shielding layer significantly reduces the radiation level, by more than a factor of two compared to the initial values. The second layer further decreases the dose. However, its effectiveness is lower, which is attributed to the fact that irradiation in this region occurs from all directions rather than predominantly from a single side.

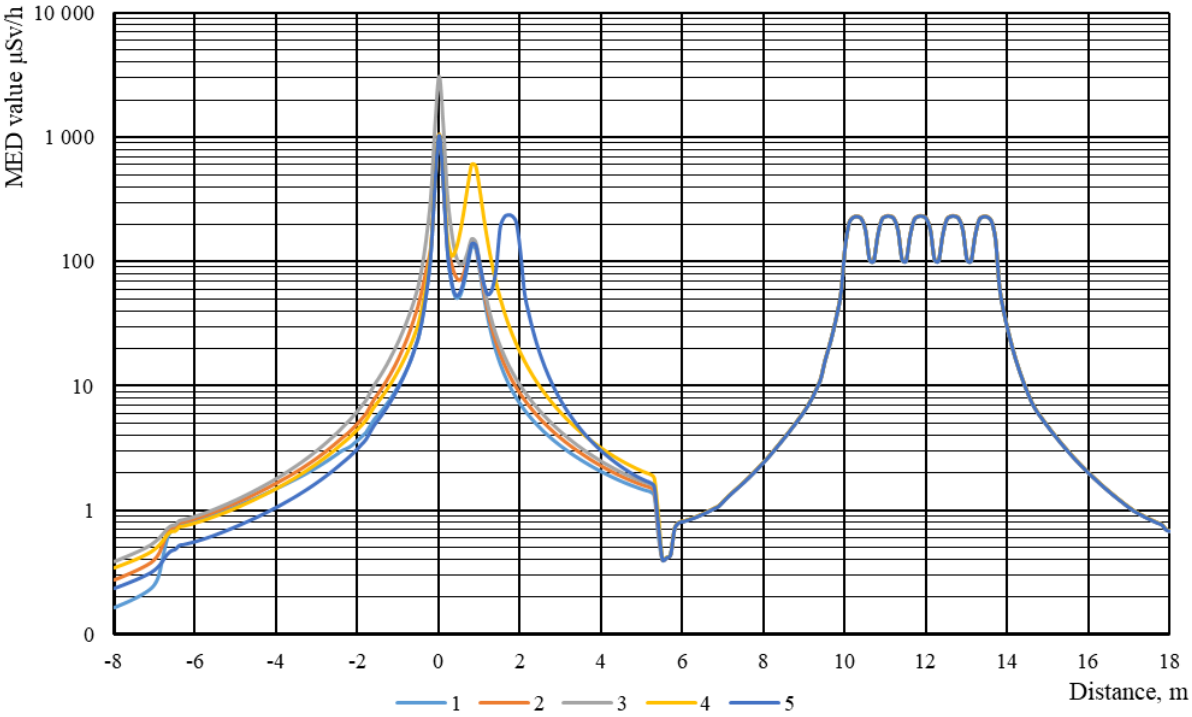
The initial design called for 15 drums of immobilized spent nuclear fuel to be placed in the reprocessing room. Following radiation safety calculations, the number of drums was reduced to five, and they were moved to a separate room with additional shielding in the form of lead plates.

The effective dose rate from a group of five drums of the mixer containing the spent nuclear fuel matrix solution was calculated, taking into account the installation of lead plate shields. The results are presented in Table 4.

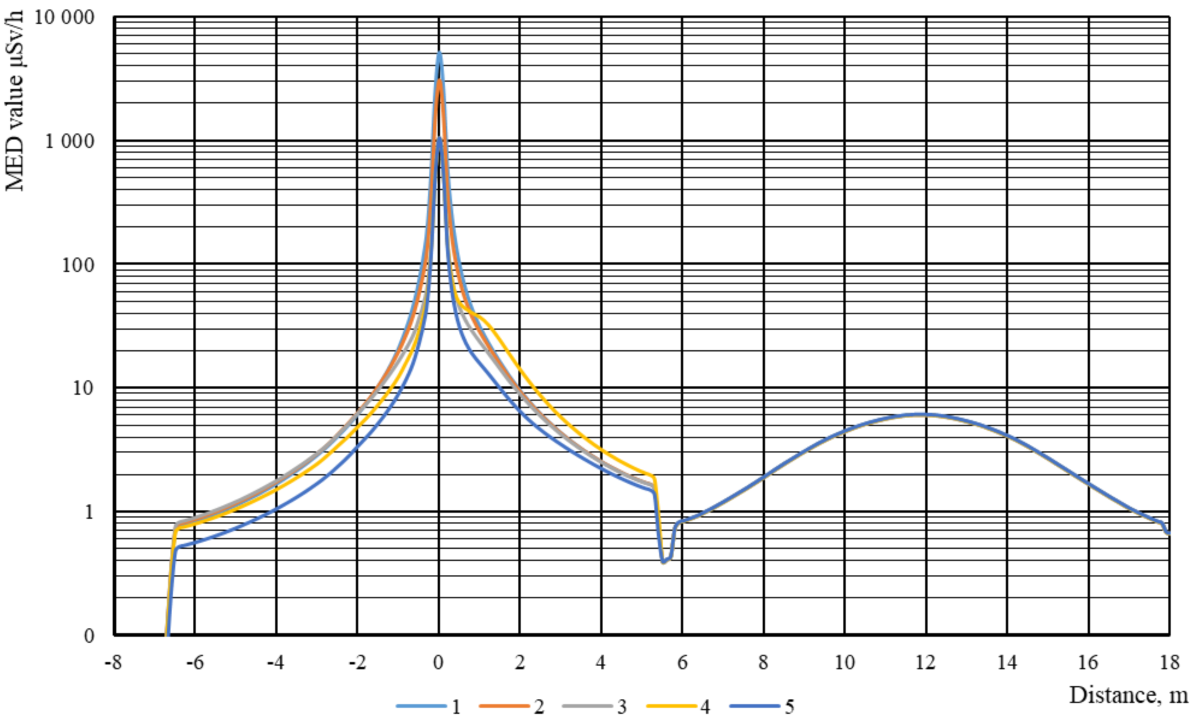
Table 4. MED values near a group of 5 barrels.

Distance, m	MED value µSv/h
0.02 (close to the barrels)	151.06
0.02 (close to the RPM)	81.22
0.5 (from the RPM)	30.11
1 (from the RPM)	15.88
2 (from the RPM)	6.34

The maximum values of the MED along the OX axis are presented in the form of graphs in Figure 9. The numerical values of the MED are presented in logarithmic form at the site and service area.



(a) facility



(b) separate room (service area)
1) Scenario №1; 2) Scenario № 2; 3) Scenario № 3; 4) Scenario №4; 5) Scenario №5

Figure 9. Maximum values MED.

Analysis of the causes of maximum dose generation

An analysis of the distribution of the maximum effective dose rate (EDR) shows that the nature of the radiation field is determined not only by the total activity of the sources, but primarily by their spatial location relative to the working areas and the effectiveness of their shielding.

The highest DER values are observed in the immediate vicinity of the grinding unit loading chamber and in the central part of the section, where the arrangement of the graphite blocks creates the most intense local gamma field. Thus, the maximum DER inside the loading chamber reaches 5137.56 μSv/h (scenario 1), while at a distance of 0.5–1 m, the level decreases by more than an order of magnitude. This indicates the dominant influence of the geometric factor and the law of radiation attenuation with distance, as well as the high sensitivity of dose loads to the local concentration of radioactive material.

A different mechanism for dose field generation is observed for the mixing drum. The maximum value (122 μSv/h) is achieved only in scenario 5 when located close to the barrel; however, at a distance of 1–2 meters, the difference between the scenarios decreases significantly. This indicates that the effect of fuel redistribution is localized, and the dose increase is due to the formation of a limited area of increased activity located in close proximity to the work position.

Additional analysis demonstrates the high effectiveness of engineered barriers. Despite significant radiation levels inside the central hall, the dose rate decreases by several orders of magnitude behind protective walls and shields. For example, behind the perimeter wall of the central hall, the values are only $(2.5–4.0)\times10^{-3}$ μSv/h, confirming the sufficient effectiveness of biological shielding and the localization of radiation exposure within the process zone.

Thus, the determining factor in the formation of maximum doses is the localized accumulation of the source in an unfavorable geometric configuration, and not solely the total mass or activity of the fuel. This confirms the need to analyze material redistribution scenarios as a key element of a conservative radiation safety assessment.

A comparison of the scenarios shows that the most conservative (worst-case) scenario depends on the control zone under consideration, as the maximum doses occur in different HEU configurations.

For the loading chamber of the milling facility, the worst-case scenario is Scenario 1, in which the DER inside the chamber reaches 5137.56 μSv/h, and close to the chamber, 194.94 μSv/h. This is likely due to the configuration in which the source is located as close as possible to the loading area, ensuring minimal geometric attenuation and the absence of intermediate shielding elements.

In the central part of the section, the highest values are observed for Scenario 4, where the DER in Zone A is 5136.69 μSv/h, significantly exceeding alternative configurations. Similarly, at the boundary of the first radiation shield, the maximum is realized in scenario 3 (2489.22 μSv/h), and after the first shield, also in scenario 3 (1210.07 μSv/h). This indicates a pronounced dependence of the dose field on the spatial redistribution of the source and the relative position of the protective shields.

Scenario 5 for the mixing barrel requires special attention, as it leads to a sharp local maximum (122 μSv/h) in the immediate vicinity of the equipment, despite the absence of extreme values at other control points. Therefore, this scenario should be considered a local worst-case scenario for equipment maintenance operations, although it is not the most radiation-intensive on a site-wide scale.

Overall, the results show that there is no single worst-case scenario for the entire facility. The most conservative estimate is achieved by analyzing a set of scenarios, since different fuel configurations lead to maxima in different critical zones. This approach is more physically sound than choosing a single limiting case and provides a conservative assessment of radiation safety with incomplete information on the fuel distribution.

The obtained results demonstrate a pronounced dependence of dose fields on the spatial distribution of radiation sources and the configuration of technological equipment. The highest dose rate values are observed near regions with localized material accumulation, confirming the key role of geometry and non-uniform radionuclide distribution in shaping the radiation environment. In this context, the total source activity appears to be a less decisive factor than its spatial localization.

The installation of additional radiation shielding reduced dose rates by approximately a factor of two in the most highly loaded working area (Zone B), whereas in the less loaded area (Zone C), the shielding effect was limited to approximately 10–15%. These values are consistent with findings reported in previously published studies, where Monte Carlo simulations demonstrated that local shielding configurations in spent nuclear fuel handling systems can reduce dose fields from several tens of percent up to multiple-fold decreases, depending on system geometry and shield placement [20].

An important factor determining radiation shielding effectiveness is the spatial configuration of sources and shielding structures. As reported in the literature, even when identical shielding materials are used, variations in their arrangement and relative geometry with respect to the source can result in significant redistribution of dose fields and changes in local dose maxima [21]. In this regard, the observed difference in shielding effectiveness among various facility zones can be explained by a combination of geometric factors, shielding by structural components, and heterogeneous source distribution.

It should be noted that the computational model is based on a number of conservative assumptions related to the simplified representation of equipment geometry and the absence of a detailed description of internal structural elements. In addition, the source distribution was defined through scenario-based assumptions due to insufficient initial information regarding the actual fuel configuration during reprocessing. Such an approach enables the derivation of upper-bound dose estimates sufficient for radiation safety justification and engineering decision-making.

Thus, the calculation results confirm the applicability of the scenario-based Monte Carlo approach for radiation environment assessment under conditions of incomplete input data and demonstrate the importance of accounting for geometric factors in the design of radiation protection systems.

The results obtained in this work show that the distribution of radioactive material within the process equipment has a significant influence on the radiation conditions at the facility. In particular, local accumulations of fuel may lead to substantially higher dose rates than those predicted from the total fuel inventory alone. This observation should be taken into account when developing shielding systems and selecting equipment layouts.

The performed calculations demonstrate that the use of several conservative fuel distribution scenarios makes it possible to assess radiation conditions under limited information about the actual location of radioactive materials. Such an approach can be applied at the design stage to identify potentially unfavorable operating conditions and to evaluate the effectiveness of proposed radiation protection measures.

The obtained results were used to support engineering decisions related to shielding design and personnel protection. The methodology developed in this study may also be useful for radiation safety assessments of other spent fuel handling and processing facilities where uncertainties in material distribution are present.

5. Conclusion

This section should summarize the main findings of the study in a concise and clear manner, directly reflecting the stated objectives. It should highlight the scientific novelty and key contributions without repeating detailed results. Authors should indicate the practical significance and potential applications of the work, as well as its limitations. Where appropriate, brief recommendations for future research may be included.

An assessment of the radiation situation during the dilution process of HEU graphite fuel from the IGR reactor was performed based on the layout of the grinding and immobilization equipment in the designed facility. Due to the lack of quantitative information on the fuel distribution at each point of the installation, five scenarios representing different simultaneous distributions of HEU fuel blocks were considered:

Scenario 2: The maximum number of HEU fuel blocks (1.5 blocks) is located in the feed hopper of the jaw crusher and in the electromagnetic dosing feeder. At the remaining two positions – the mill and the receiving hopper of the weigh feeder – 0.5 HEU fuel blocks are assumed at each location.

Scenario 3: The maximum number of HEU fuel blocks (1.5 blocks) is located in the mill and in the electromagnetic dosing feeder. At the other two positions – the loading throat of the jaw crusher and the receiving hopper of the weigh feeder – 0.5 HEU fuel blocks are assumed at each location.

Scenario 4: The maximum number of HEU fuel blocks (2.5 blocks) is located in the receiving hopper of the weigh feeder. At the remaining positions – the feed inlet of the jaw crusher, the mill, and the electromagnetic dosing feeder – 0.5 HEU fuel blocks are assumed at each location.

Scenario 5: The maximum number of HEU fuel blocks (2 blocks) is uniformly distributed within the cement matrix inside the barrel, while 0.5 HEU fuel blocks are assumed at each of the other positions. The research results were used to revise the facility design and justify the use of additional equipment to ensure normal operation of the technological process.

Based on the calculation results, the following modifications were implemented:

1. Three SRZ-6 lead glass windows (lead equivalent of 12 mm) were added for monitoring the technological process.

2. Radiation protection screens made of

- SRZ-Z lead glass (lead equivalent of 2.5 mm) were installed in the loading chamber;

- RZM Abris mats were added in the service area.

3.With regard to the barrels containing cemented LEU fuel:

- The number of barrels with spent nuclear fuel was reduced, and the barrels were relocated to a separate room.

- In the barrel storage room, 10 mm thick lead plate shields were installed on three sides of the barrel storage area.

In conclusion, this study extends beyond a conventional engineering assessment by proposing a systematic methodology for evaluating radiation fields under conditions of incomplete information on fuel distribution.

The work introduces a conservative, scenario-based modeling framework that enables the representation of uncertain fuel configurations in complex technological systems. It demonstrates that peak dose formation is governed primarily by localized fuel accumulation rather than by the total fuel mass, thereby refining the understanding of key radiological drivers. In addition, the study provides a quantitative assessment of shielding effectiveness, directly linking Monte Carlo simulation results with design optimization decisions.

Importantly, the proposed approach is transferable and can be applied to a wide range of spent nuclear fuel management systems. Overall, the methodology offers not only site-specific insights but also a generalized framework for radiation safety analysis in fuel processing facilities.

The facility design was subjected to expert review. It was concluded that the designed dilution facility complies with the safety requirements specified in the technical documentation, as well as with the applicable rules and regulations of the Republic of Kazakhstan governing the use of nuclear energy.

Funding

This research was funded by the Committee of Science of the Ministry of Science and Higher Education of the Republic of Kazakhstan (Grant No. BR24993104).

Conflicts of interest

The authors declare no conflict of interest.

Author contributions

“Conceptualization, I.P. and A.M.; Methodology, I.B.; Software, A.M.; Validation, A.M., I.P. and A.P.; Formal Analysis, V.P.; Investigation, Y.B.; Resources, Y.B.; Data Curation, I.B.; Writing – Original Draft Preparation, I.P.; Writing – Review & Editing, A.P.; Visualization, A.M.; Supervision, I.P.; Project Administration, V.P.; Funding Acquisition, Y.B.”

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