

OPTIMIZATION OF RADIATION SHIELDING AND PLACEMENT OF SPENT FUEL ASSEMBLIES IN THE HI-STAR 190 TRANSPORT CONTAINER TO MINIMIZE NEUTRON RADIATION

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The neutron radiation passage through the transport container (TC) HI-STAR 190 was simulated. It was found that there are areas where the dose rate exceeds the limit level, which was established according to the calculations of the HOLTEK company. Taking into account world experience and the results of our previous research, a composite based on lead, aluminum and boron is an effective material for protecting the TC, therefore we proposed to replace the lead layer in the radiation protection of the TC with a composite of 95% lead and 5% boron and add aluminum to this composite in the form of a spiral roll-like structure as a heat conductor. The simulation was performed for the new HI-STAR 190 TC when placed in the BCC 31 VTVZ-A with spent fuel with a burnup of 55 (MW·day)/kg U. It was found that the average dose rate is reduced by 25% compared to the TC, where pure lead is used in this block of layers. To ensure radiation safety, a study was conducted to optimize the placement of assemblies with different burn-up levels and exposure times in the TC.

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Currently, Ukraine has no experience transporting more than 12 spent fuel assemblies (SFAs), as was previously done using TK-13 containers. Therefore, the radiation and operational safety of HI-STAR containers, as well as the safety of operating personnel, requires urgent study. Such research should substantiate the radiation and operational safety of HI-STAR containers and service personnel, which is a necessary condition for the safe operation of the Centralized Spent Nuclear Fuel Storage Facility (CSFSF). In turn, this is a strategically important task for the energy independence of Ukraine.

Global experience in the design of modern transport containers includes the use of composite structural materials based on boron, steel, cast iron, or lead as biological shielding, solid neutron shielding, and increasing both the thermal power capacity and storage capacity of the container [1, 2].

The HI-STAR 190 UA transport container developed by Holtec is designed to transport 31 spent fuel assemblies with fuel burnup up to 55 (MW·day)/kg U. This differs from other containers already in operation, where either the maximum burnup level is less than 45 (MW·day)/kg U or the number of fuel assemblies is less than 20. Thus, the HI-STAR 190 UA container contains the maximum number of assemblies and nearly the maximum burnup level. Therefore, increasing the efficiency of shielding materials to ensure radiation safety is one of the priority tasks

According to the calculations of the company HOLTEC, the maximum dose rate level at the middle of the lateral surface of the TC HI-STAR 190 UA [4] is given in Table 1.

Table 1

Maximum dose rate on the surface of the HI-STAR 190 UA container

Neutrons (mSv/h)	⁶⁰ Co (mSv/h)	Gamma (mSv/h)	(n,γ) (mSv/h)
1.027	0.067	0.05	0.015

The data show that the main hazard is neutron radiation. Unlike gamma radiation, for which the dose contribution from the basket is mainly determined by the outer layer along its perimeter, neutron radiation involves the entire volume of fuel assemblies located in the basket. In addition, since most materials in the container consist of elements with large atomic numbers, multiple scattering phenomena play a significant role in neutron distribution. As a result, the geometry of the container has a substantial influence on the neutron dose distribution outside the container [5].

In accordance with the above, a three-dimensional model was developed to calculate neutron dose loads that closely approximates the real geometry without simplifications except for the fuel assembly active zone. The active zone was partially homogenized and modeled as a hexagonal prism with a density of 3.5 g/cm³ and the following composition: 52.3% ²³⁸U, 25.8% ⁴⁰Zr, 14.6% ²⁶Fe, and 7.3% ¹⁶O. Similar zones with corresponding densities will be built for the SFA head and SFA tail. A cross-section through the central axis of the HI-STAR 190 UA TC is shown in Fig. 1.

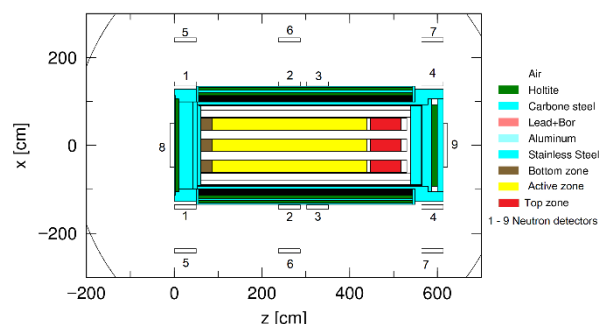


Fig. 1. TC scheme and location of neutron detectors outside the container (section through the central axis)

To enable a more detailed analysis of the distribution of neutron radiation in the model, 9 neutron detectors were located around the TC, 7 of which are located in the radial direction from the lateral side of the neutrons and are rings 50 cm high and 10 cm thick and

are located on the lateral surface of the TC and at a distance of 1 m from it. Two detectors located on the bottom and lid of the TC have the shape of cylinders with a radius of 50 cm and a thickness of 10 cm.

Dose loads are determined as the average value of detectors per unit volume. The detectors are marked with numbers 1–9, the source of neutron radiation is the active zone of the SFA, the neutron shield material is a special Holtite-B composite and plates made of the Metamic – HT (Al – B₄C) nanomaterial [6] which are located inside the MPC, as shown in Fig. 1. Note that for this configuration, detector 3, which is located in the middle of the side surface of the TC, corresponds to the location of the detector in the calculations by specialists of the HOLTECH corporation [4], and the results we obtained for this detector can be compared with the results given in Table 1.

The neutron radiation passage through the HI-STAR 190 TC was simulated when different quantities of SFAs were placed in the MPC. The dose distribution depending on the detector number is shown in Fig. 2. It was established that the neutron radiation dose near the middle of the TC side surface is not maximum, and therefore there are areas where the dose level exceeds the limit level that was established according to the calculations of the HOLTEC company [4].

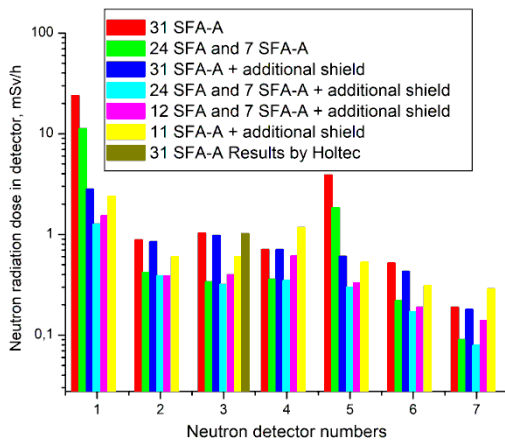


Fig. 2. Distribution of dose rate depending on the detector number

The obtained data show that ensuring the required level of radiation safety is possible only when installing an additional layer of neutron protection material of the Holtite-B type in the lower part of the TC, and a direct reduction in the number of HFFs is not advisable. It has been established that the most optimal option for transporting 31 HFFs is a TC with additional neutron protection in the lower part and with a BCC with a mixed arrangement of 24 HFFs and 7 HFFs-A

Taking into account world experience and the results of our previous research [7], a composite based on lead, aluminum and boron is an effective material for protecting the TC, so we proposed to replace lead with a composite of 95% lead and 5% boron and add aluminum to this composite in the form of a spiral structure similar to a roll as a heat conductor, to preserve the external geometric parameters, the thickness of the aluminum spiral was 1 mm, and that of lead with the addition of 5% boron was 13.1 mm, which corresponds to the spiral pitch for aluminum. Thus,

instead of a lead layer, we have a structure similar to a roll of 11 layers of a composite of 95% lead and 5% boron and 12 layers of aluminum. The scheme of the new TC and the location of the neutron detectors are shown in Fig. 3.

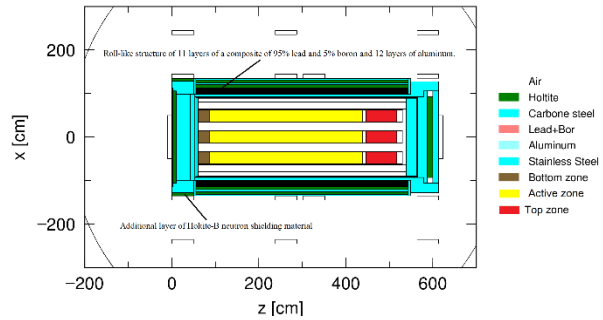


Fig. 3. Scheme of new TC with additional shield (section through the central axis)

Calculations of neutron radiation doses in detectors outside the TC with additional neutron shield when placing 31 SFAs with spent fuel with an initial fuel enrichment of 4.4%, burnup of 55 (MW·day)/kg U in the multi-purpose cask (MPC) after a three-year campaign in the WWER-1000 reactor and a 7-year aging in the reactor pools are given in Table 2.

Table 2
Neutron radiation doses outside the TC with additional shield when placed in the MPC 31 SFA-A with spent fuel with initial fuel enrichment of 4.4%, burnup of 55 (MW·day)/kg U

Neutron detector numbers	Neutron radiation dose in detector, mSv/h
1	2.21
2	0.39
3	0.41
4	0.57
5	0.47
6	0.22
7	0.13
8	0.39
9	0.1

A comparison of the dose rate relative to the original TC is given in Table 3.

Since the geometric parameters were the same for the old and new TC. Therefore, change in dose level clearly shows the effectiveness of shield in the form of a structure of 11 layers of a composite of 95% lead and 5% boron and 12 layers of aluminum that we proposed for the TC, the use of a new type of shield reduced the total weight of the TC by 4.6 t. As we can see, when 31 SFA-A with spent fuel with a burnup of 55 (MW·day)/kg U was placed in the MPC, the dose level on detector 1 decreased by 21% despite the fact that it is not located directly behind the protective layer but is shifted to the lower part of the TC. This is due to the increase in the efficiency of absorption of scattered neutrons, which is 2.21 mSv/h, which brings us closer to the Maximum Permissible Dose (MPD) of 2 mSv/day, which is a very good indicator, taking into account that in other TCs the maximum burnup level is less than

45 MW or the number of SFA is less than 20. The dose level on detectors 2 and 3 decreases by 55 and 58%, respectively, and is 0.39 and 0.41 mSv/h.

Table 3

Comparison of dose rate for the new TC with additional protection when placed in the MPC 31 SFA-A with spent fuel with initial fuel enrichment of 4.4%, burnup of 55 (MW-day)/kg U relative to the original TC with additional shield and pure lead

Neutron detector numbers	Relative dose level
1	0.79
2	0.45
3	0.42
4	0.81
5	0.77
6	0.51
7	0.7
8	0.96
9	1.16

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ОПТИМІЗАЦІЯ РАДІАЦІЙНОГО ЗАХИСТУ ТА РОЗМІЩЕННЯ ВТВЗ ТРАНСПОРТНОГО КОНТЕЙНЕРА HI-STAR 190 ДЛЯ МІНІМІЗАЦІЇ НЕЙТРОННОГО ВИПРОМІНЮВАННЯ

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На відміну від інших типів транспортних контейнерів, які вже експлуатуються у світі, та в яких або максимальний рівень вигорання менше ніж 45 (МВт·добу)/кг U, або кількість ВТВЗ менше 20, у ТК HI-STAR 190 компанії Holtec планується перевозити 31 ВТВЗ із ВЯП із вигоранням 55 (МВт·добу)/кг U. Отже, у ТК HI-STAR 190 є максимальна кількість ВТВЗ та майже максимальний рівень вигорання, тому підвищення ефективності матеріалів захисту ТК для забезпечення радіаційної безпеки є одним із пріоритетних завдань. Проведено моделювання проходження нейтронного випромінювання крізь ТК HI-STAR 190. Встановлено, що існують області, де рівень дозового навантаження перевищує граничний рівень, який було встановлено згідно з розрахунками компанії HOLTEK. З урахуванням світового досвіду, а також за результатами проведеного нами раніше дослідження композит на базі свинцю алюмінію та бору є ефективним матеріалом захисту ТК, тому нами запропоновано замінити шар свинцю в радіаційному захисту ТК композитом 95% свинцю і 5% бору, та додати у цей композит алюміній у вигляді спіральної структури, подібної рулету в якості теплопроводу. Проведено моделювання для нового ТК HI-STAR 190 при розміщенні в БЦК 31 ВТВЗ-А з ВЯП із вигоранням 55 (МВт·добу)/кг U. Встановлено, що середнє дозове навантаження зменшується на 25% у порівнянні від ТК, де в цьому блоці шарів застосовано чистий свинець. Для забезпечення радіаційної безпеки проведено дослідження з оптимізації розміщення у ТК збіроч із різним рівнем вигорання та часом витримки.