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RADIATION AND NUCLEAR SAFETY ANALYSIS FOR THE SPENT FUEL TRANSPORTATION PACKAGE AFTER THE SHORT-TERM COOLING BY USING COMPUTER SIMULATION MODELING METHODS

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This work represents results of the second stage of a complex research to substantiate the principal possibility of managing spent fuel after short-term cooling and developing requirements for the design of the package. Calculations for the radiation protection have been completed, the nuclear safety has been justified, and the relevant accompanying neutron-physical processes have been studied. In the first stage of the research, calculations were performed to assess the level of heat generation and radiation characteristics of the spent nuclear fuel after a short-term cooling period, as well as thermal calculations of the package.
As a result, technical feasibility of using a transport container for spent nuclear fuel has been substantiated. The most acceptable option of a container for spent nuclear fuel is to be made of iron-concrete + uranium dioxide and cast iron + uranium dioxide with gas filling or with liquid filling of the absorber in a basket.
MCNP code was used to justify radiation protection and nuclear safety under normal and emergency operating conditions, and neutron-physical processes accompanying spent nuclear fuel were studied. The work examined several options for materials for radiation protection, depending on their thickness and the fill of the spent nuclear fuel container.

Keywords: *nuclear fuel, nuclear fuel cycle, nuclear safety, spent fuel container, nuclear reactor.*

INTRODUCTION

At the nationwide referendum held on October 6, 2024, the population of the Republic of Kazakhstan confidently supported the course towards the construction of a nuclear power plant in Kazakhstan, marking a significant milestone in the transition to a new practical phase of developing its own nuclear energy sector. Currently, a systematic concept for the development of the nuclear energy industry is being elaborated [1], which takes into account all aspects related to the project of construction, operation, and decommissioning of the future nuclear power plant without exception. The management of spent nuclear fuel and radioactive waste is one of the key aspects of this concept and generally corresponding to national strategy of the Republic of Kazakhstan.

The nuclear fuel cycle is a chain of interconnected technological processes, and one of its final stages is the management of spent nuclear fuel (SNF). At this stage, an important aspect of ensuring safety when handling spent nuclear fuel is reducing the risk of nuclear materials becoming uncontrolled and further spreading. One way to lower the risk level may be to shorten the list of processes in the operational chain and their duration. Most operations related to spent nuclear fuel cannot be technically bypassed. However, excluding the operation of prolonged storage of fuel in the spent fuel pool is quite promising for consideration in the management of SNF. With this approach, spent nuclear fuel (SNF) can be placed into specialized transport packaging units (TPUs) after a short holding period in the active zone of the stopped reactor or in a special storage pool, and then sent to a reprocessing plant or a temporary storage location [2].

The container for transporting spent nuclear fuel is key equipment in this scheme for handling spent fuel, and the conceptual design of the TPU must ensure safe thermal conditions, nuclear and radiation safety, protection from radiation, preservation of radioactive materials, and integrity and tightness even after serious accidents and incidents [3, 4]. The thermal analysis research was conducted in the first stage of the comprehensive work and it was the basis for farther investigations conducted in this article.

SOURCE DATA, APPROACHES, AND ASSUMPTIONS

The developed Transport Package (TP) in accordance with document [5] is classified as type B (U)F package, as well as according to the National Regulations [6–8] – as type B, class I for nuclear safety, category III for radiation hazard.

In this work, the calculations of nuclear safety and the calculation of the distribution of the effective dose rate of photon radiation were carried out by using the MCNP calculation code for the three-dimensional geometry of the TP.

Whereas the tank-type water-cooled nuclear reactor on thermal neutrons is a main item of worldwide nuclear generation, also considering prospects of possibly NPP construction with this type of reactors in the Republic of Kazakhstan, for further consideration shall be accepted that the TP will be use in aims of high powered ABWR, AR1000, VVER-1000 type reactor's fuel assemblies transportation issues.

As a working fluid, filling TP was considered to use water or gas (helium, argon, CO_2). On the table 1 are shown the total initial findings on structures of being modelling fuel assemblies for various type reactors and SNF radiation characteristics.

Table 1. Characteristics of Spent Nuclear Fuel Used in TP Modeling

Characteristics	Parameters		
	ABWR	AP1000	VVER-1000
Nuclear fuel material	Sintered uranium dioxide	Sintered uranium dioxide	Sintered uranium dioxide
Length of fuel assembly (full), mm	4470	4795	3837
Lattice type (geometry of fuel rod arrangement in fuel assembly)	10×10 (square)	17×17 (square)	triangle
Fuel assembly number in the core	872	157	163
Fuel elements number in the FA	92	264	311
Fuel elements shell material	Zircaloy-2	ZIRLO	Zr+Nb alloy
Shell material thickness, mm	0.66	0.57	0.67
Fuel element outer diameter, mm	10.3	9.5	9.1
One piece FA weight (cover including), kg	300.0	799.7	680.0
Intensity of gamma radiation sources of fission products, photon/seconds·kgU	$5.09 \cdot 10^{14}$	$7.79 \cdot 10^{14}$	$8.36 \cdot 10^{14}$
Intensity of neutron radiation sources, neutron/seconds·kgU	$3.07 \cdot 10^{6}$	$1.87 \cdot 10^{6}$	$2.34 \cdot 10^{6}$

RADIATION PROTECTION CALCULATIONS

Analysis of the TP radiation protection is performed by usage of SNF radiation characteristics for various time of handling considering gamma radiation fission products, SNF neutron radiation, activate nuclides gamma-radiation in the FA structural steel.

The analysis of radiation safety was carried out in the following scenarios:

– For normal operating conditions: loading of SFAs from the reactors under gas and liquid medium conditions.

– For emergency conditions: SFA is damaged and compacted against one of the internal surfaces (side, bottom, lid); there are two proposed variants of concrete protection: 100 mm depth concrete dehydration (emergency situation in the case of TP appearance in fire zone).

The calculation of the shielding was based on the constraints of the container's mass (no more than 130 tons), dimensions (external diameter no more than 250 cm), and the equivalent dose rate of radiation on the surface of the TP (not exceeding 2.0 mSv/h).

The following shielding layers were considered in this calculation (consequently throughout the radius):

1. Aluminum basket (with a neutron-absorbing material or it may be absent);
2. Steel (for the structural integrity of the container);
3. Shielding layer based on reinforced concrete or high-strength cast iron (with the addition of depleted uranium oxide (30% by volume), lead (as a flat layer, 50% of the shielding volume), gadolinium oxide (10% by volume), boron carbide (10% by volume);
4. Outer shell made of steel.

Radiation protection calculations under normal operating conditions

Radiation protection calculations under normal operating conditions were conducted for various numbers of SFAs, with the container filled with either water or gas, and for different options of protective layers, where the thicknesses of the layers were selected based on the limitations for the equivalent dose rate of radiation at the surface of the container (not exceeding 2.0 mSv/h). Furthermore, the mass and dimensional characteristics of the container were evaluated to eliminate variants that did not meet the design restrictions in terms of mass (not more than 130 tons) and dimensions (outer diameter not more than 250 cm).

When evaluating the loading of the container filled with water, limitations arising from the analysis of neutron-physical calculation results were also taken into account, as for some options, the dose rate from neutron radiation could not be calculated due to criticality concerns.

Calculation results

The results of the calculations pertaining to the SFA container loading of different SFA of the reactor, along with the mass and dimensional characteristics of the container when using shielding made of different materials, are presented in Table 2..

Neutron protection

In order to reduce the dose from neutron radiation during gas filling of the container cavity, the influence of different variants of the neutron shielding arrangement in the container structure on the neutron flux attenuation was evaluated. The variants consider a cast iron canister with DUO_2 additive filled with spent fuel assembly from VVER-1000 reactor.

The initial variant the influence of the presence of a boron carbide spacer grid (1 cm thick) surrounding all SFAs of the container is considered. In the second and third variants, the influence of adding gadolinium oxide and DUO_2 to the cast iron shielding – 1% and 10% of the shielding volume, respectively – was considered. The results of the comparison of the design variants with such baskets and with additives are given in Table 3.

Conclusions for the assessment of radiation protection under normal operating conditions

The contribution of neutron and gamma radiation to the formation of the dose rate at the container surface depends significantly on the type of protection used. For example, if reinforced concrete-based protection is used, the main contribution is from gamma radiation (the dose rate from neutron radiation is 1–3 orders of magnitude lower). Moreover, when using protection based on cast iron, the gamma radiation dose rate is 3–10 times lower than the neutron dose rate.

Table 2. Results of calculating the mass and size characteristics of the TP for protection made of reinforced concrete

Cooling/Reactor		SFA Q-ty	Inner radius, cm	Containment thickness	TP mass, t,	Surface dose rate, mSv/h		
						γ-quants	neutrons	total
Reinforced concrete protection (ρ=4 g/cm³)								
Gas	ABWR	9	40	80	100	2.02	$4.40 \cdot 10^{-3}$	2.02
	AP1000	4	40	85	104	1.32	$1.67 \cdot 10^{-3}$	1.32
	VVER-1000	3	40	85	103	2.04	$2.06 \cdot 10^{-3}$	2.04
Water	ABWR	8	40	80	103	1.04	$3.08 \cdot 10^{-2}$	1.07
	AP1000	4	40	80	97	1.68	$1.34 \cdot 10^{-2}$	1.69
	VVER-1000	4	45	80	103	1.37	$1.53 \cdot 10^{-2}$	1.39
Reinforced concrete protection with the addition of DUO_2 (50% by volume, ρ=7.2 g/cm³)								
Gas	ABWR	37	70	45	112	0.90	0.17	1.07
	AP1000	21	80	45	129	1.47	0.13	1.60
	VVER-1000	9	70	45	107	1.90	1.11	2.01
Water	ABWR	25	70	45	116	0.52	0.20	0.72
	AP1000	12	75	45	125	0.77	0.21	0.98
	VVER-1000	8	70	45	114	0.45	0.13	0.58
Cast iron containment (ρ=7.2 g/cm³)								
Gas	ABWR	21	55	60	130	$5.25 \cdot 10^{-2}$	1.68	1.74
	AP1000	9	60	55	124	0.33	1.67	2.00
	VVER-1000	7	60	55	122	0.42	1.61	2.02
Water	ABWR	21	70	50	130	0.43	0.62	1.05
	AP1000	5	55	50	106	0.55	0.97	1.52
	VVER-1000	8	70	50	130	0.50	0.60	1.10
Cast iron cladding with the addition of DUO_2 (50% by volume, ρ=8.8 g/cm³)								
Gas	ABWR	37	70	45	130	$5.84 \cdot 10^{-2}$	1.65	1.71
	AP1000	13	75	40	121	0.59	1.59	2.18
	VVER-1000	9	70	45	128	$9.96 \cdot 10^{-2}$	0.99	1.09
Water	ABWR	21	65	35	97	1.50	0.37	1.87
	AP1000	9	65	45	129	$4.93 \cdot 10^{-2}$	1.29	1.34
	VVER-1000	8	70	40	118	0.22	0.86	1.08

Table 3. Results of neutron shielding evaluation - basket containing boron carbide

Additive	SFA q-ty	Inner radius, cm	Containment thickness	TP mass, ton	Surface dose rate, mSv/h		
					γ-quants	neutrons	total
Without spacer grid	3	50	40	83	0.59	0.98	1.57
With spacer grid	3	50	40	83	0.59	0.97	1.56
Without Gd_2O_3	7	60	40	98	0.85	1.80	2.65
1% Gd_2O_3	7	60	40	98	0.85	1.70	2.54
10% Gd_2O_3	7	60	40	98	0.85	1.51	2.36

Analysis of the results of the evaluation of the applicability of different variants of neutron protection for gas filling of the container cavity shows that their influence on the dose rate of neutron radiation on the container surface is negligible. This is explained by the large proportion of high energy neutrons in the radiation spectrum of the sources, while the considered neutron shielding materials are capable of absorbing low-energy neutrons well.

Calculation of radiation protection in case of emergency

Calculations of the radiation protection of TP under emergency conditions were carried out for the following situations:

1. The SFAs are destroyed and compacted on one of the inner surfaces (side, bottom, or top);
2. Dehydration of the concrete shielding due to the effect of high temperature.

In the first situation, it is assumed that 10% of the fuel will escape from the SFA cladding and would be compacted in the lower part of the container. The broken fuel is modelled as a homogeneous mixture of uranium dioxide and air (or water) with a porosity ranging from 40% (sand porosity) to 70% (undamaged fuel porosity).

In the second situation, it is assumed that if the concrete protected TP is placed in a fire zone with a flame temperature of 800 °C for 30 minutes, it is possible to dehydrate the concrete to a depth of 100 mm.

Fuel destruction dose rate estimation

The results of the dose rate estimation on the surface of the cask at points located in the plane passing through the center of the destroyed fuel volume at different porosities of the compacted fuel are given in Table 4.

For the example, a reinforced concrete gas-cooled container with 9 SFAs of ABWR and a wall thickness of 80 cm is considered.

Table 4. Results of the dose rate calculations for the case of the destruction of the fuel element

Porosity %	Surface dose rate, mSv/h		
	γ-quants	neutrons	total
Intact	2.02	$4.40 \cdot 10^{-3}$	2.02
70	1.41	$3.08 \cdot 10^{-3}$	1.42
60	1.88	$4.11 \cdot 10^{-3}$	1.89
50	2.35	$5.14 \cdot 10^{-3}$	2.36
40	2.83	$6.17 \cdot 10^{-3}$	2.83

Concrete dewatering dose rate estimation

The results of the dose rate estimation on the container surface at points located at the center of its height during wall dehydration are given in Table 5. For the example, a reinforced concrete gas-cooled container with 3 SFAs of the VVER-1000 reactor with a wall thickness of 85 cm is considered.

Table 5. Results of the dose rate calculations for the case of concrete dehydration

Dehydration depth, cm	Surface dose rate, mSv/h		
	γ-quants	neutrons	total
0	2.04	$2.06 \cdot 10^{-3}$	2.05
5	2.05	$4.81 \cdot 10^{-3}$	2.05
10	2.05	$1.12 \cdot 10^{-2}$	2.07
20	2.06	$6.08 \cdot 10^{-2}$	2.13

Conclusions on the assessment of radiation protection in emergencies

From the results of the evaluation of the dependence of the dose rate on the surface of the container on the degree of fuel compaction, it can be seen that the dose rate increases with increasing density of the destroyed fuel. However, even for the variant with the highest fuel density, the excess of the design value of the dose rate on the surface of the container will not exceed 41%. Which, taking into account the application of the reserve factor 2 in the assessment of the thickness of the radiation shielding, indicates that the severity of such an accident for the personnel and the public is insignificant and that there is no need to take any protective measures.

From the results of the evaluation of the dependence of the dose rate at the surface of the container on the thickness of the dehydrated layer: dewatering has the greatest effect on the dose rate from neutron radiation, which increases by a factor of 6 when the shielding is dewatered to a depth of 10 cm. However, as the neutron dose rate is several orders of magnitude lower than the photon dose rate. There is no significant increase in the total dose rate – even if the shielding is dewatered to a depth of 20 cm, the excess of the design value of the dose rate at the surface of the canister will be no more than 7%, which is less than the calculation error.

RESTRICTIONS ON CONTAINER DESIGN

The results of the assessments of the mass and dimensional characteristics of casks with different shielding options and their SFA loading under the assumed constraints on container mass, outside diameter and surface dose rate are summarized in Table 6.

NUCLEAR CALCULATIONS

The nuclear safety of TP is analyzed taking into account the regulatory requirements specified in [6–11], using the initial data on the spent fuel of the reactors under consideration, collected or evaluated in the previous stage of the topic.

The main task of the calculations for normal operating conditions is to determine the maximum load on the spent fuel of the reactors under consideration under different cooling options. The main constraints are the size of the inner cavity of the package (assumed diameter not exceeding 200 cm) and the subcriticality of the system (K_{eff} <0.95).

The objectives of the accident calculations are to analyze the cask designs in terms of their suitability to maintain subcriticality in the event of an emergency and, based on this analysis, to develop design recommendations.

The following are considered as emergency situations the inner cavity of the vessel contains residual water after dehydration (or water enters and replaces the gas), the SFAs are not destroyed; residual water in the inner cavity of the container after dehydration (or water ingress replacing the gas), the SFAs are destroyed.

Figures 1–3 show the calculation diagrams of the VVER-1000, ABWR and AP1000 reactor assemblies used in the neutron physics calculations.

Table 6. Limitations on SFA loading in TPs of different reactors for different shielding options

Option	SFA q-ty			Inner radius, cm	Shielding thickness	TP mass, ton
	ABWR	AP1000	VVER-1000			
Gas cooling						
R/C	9	4	3	40	85	104
R/C+DUO_2	37	21	9	80	45	129
Cast iron	21	9	7	60	60	130
Cast iron+DUO_2	37	13	9	75	45	130
Liquid cooling						
R/C	8	4	4	45	80	103
R/C+DUO_2	25	12	8	75	45	125
Cast iron	21	5	8	70	50	130
Cast iron+DUO_2	21	9	8	65	45	129

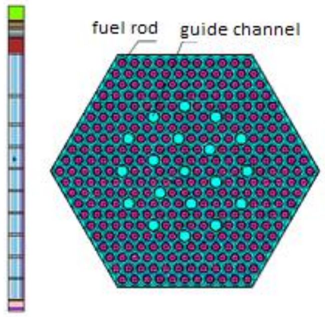

Figure 1.5 Computational model of Fuel assembly for VVER-1000

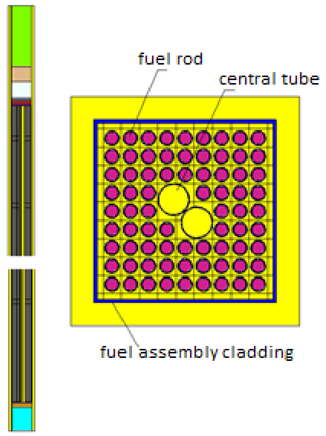

Figure 2. Computational model of Fuel assembly for ABWR

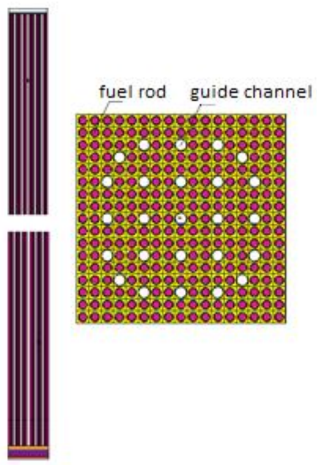

Figure 3. Computational model of Fuel assembly for AP1000

Each fuel assembly was placed in a separate basket cover made of aluminium (or MBL05 alloy in some designs) with a wall thickness of 1 cm. A continuous layer of steel was used as a reflector.

Neutron-physical calculations under normal operating conditions

Calculations of the nuclear safety of the TPs under normal operating conditions were carried out with the container filled with gas or liquid medium. The temperature of all areas of the TPs was set to 20 °C

Calculation results

The results of the K_{eff} calculations for a container with 5 SFAs under normal operating conditions using different materials to fill the inner cavity are shown in Table 7.

Table 7. Results of K_{eff} calculations for a container with five SFAs under normal operating conditions with different moderators

Moderator	K_{eff}		
	VVER-1000	ABWR	AP1000
Water	0.92431±0.0005	0.79901±0.0005	0.96055±0.0006
Argon	0.20452±0.0005	0.48602±0.0006	0.22303±0.0006
Helium	0.20292±0.0005	0.48573±0.0006	0.23360±0.0004
Carbon dioxide	0.20456±0.0005	0.48641±0.0005	0.23327±0.0004

In the case of gas cooling the influence of the inner medium material on the criticality is insignificant and therefore only helium was used in further calculations.

The SFA container loading was then evaluated for different coolant types and SFA locations. The loading was calculated on the basis of restrictions on container size (the inner diameter was assumed not to exceed 200 cm) and criticality (K_{eff}<0.95).

When calculating the container loading of SFAs from different reactors under gas cooling, the assemblies were arranged in a square, densely packed grid. Examples of calculation schemes for the cases of maximum cask loading (¼ species) are shown in Figures 4–6.

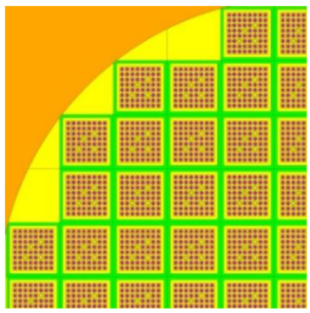

Figure 4. Computational model of gas-cooled fuel assembly container for ABWR

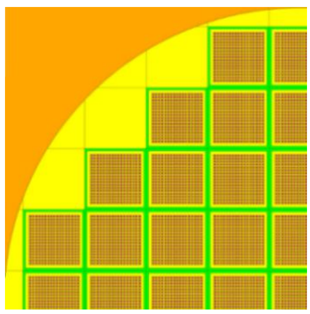

Figure 5. Computational model of gas-cooled fuel assembly container for AP1000

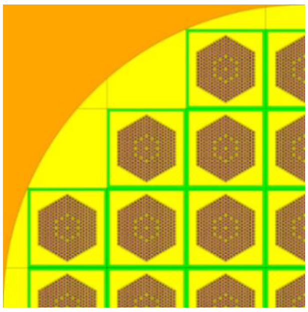

Figure 6.6 Computational model of gas-cooled fuel assembly container for VVER-1000

The results of the calculation of the SFA container loading of the different reactors with gas cooling are shown in Figure 7.

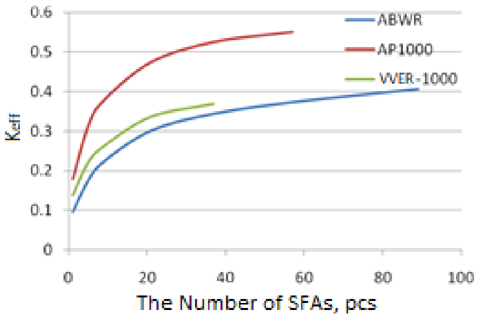

Figure 7: Criticality as a function of SFA loading during gas cooling

In the case of filling the container with water in addition to the above densely packed grid of SFA container loading, we also considered the possibility of arranging the assemblies in a square sparse grid (staggered). An example of the calculation scheme for the case of maximum container loading (¼ view) with AP1000 assemblies is shown in Figure 8.

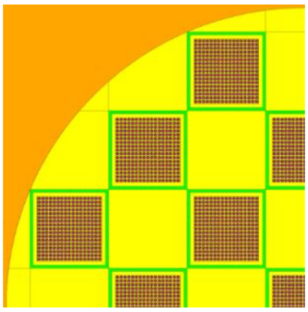

Figure 8. Computational model of a container for AP1000 FA arranged in a sparse grid with water cooling.

The results of the water-cooled container loading calculation are shown in Figure 9 and 10.

In addition, for the case of water cooling, a variant of assembly arrangement on a square densely packed grid in a basket made of MBL05 alloy was studied. This material is an alloy of aluminum with boron carbide (5%) with natural ^{10}B enrichment. The results of the calculation of the SFA loading of different reactor for this variant are shown in Figure 11.

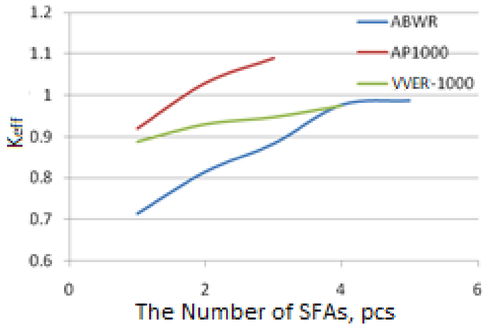

Figure 97. Criticality as a function of SFA loading during water cooling (dense grid)

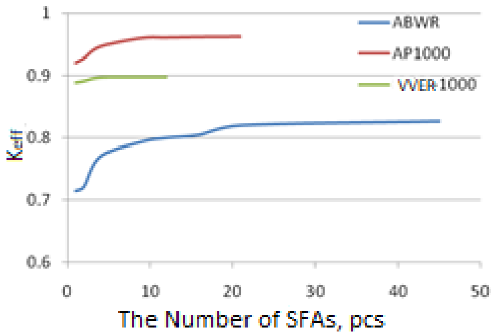

Figure 10. Criticality as a function of SFA loading during water cooling (sparse checkered grid)

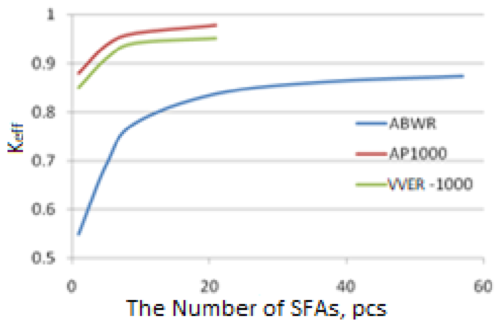

Figure 11. Criticality as a function of SFA load with water cooling (densely packed grid, MBL05 alloy basket)

Conclusions on nuclear safety in normal operation

The lowest criticality is achieved by filling the inner cavity of the TP with gases, and the difference in criticality for different gases is minimal.

When the container is filled with gas, the main limiting factor for SFA loading is the size of the inner cavity, and when the container is filled with liquid, it is the location of the SFAs and the material of the basket.

With liquid filling of the container cavity, the lowest loading is achieved by arranging the SFAs in a dense lattice without neutron-absorbing basket material.

For both gas and liquid coolings, the lowest criticality level and consequently the highest loading is observed for ABWR SFAs, while the highest is observed for AP1000 SFAs. This is due to both the lower fuel mass in the ABWR SFA and the lower enrichment.

Neutron Physics Calculations under Emergency Conditions

Calculations of the nuclear safety of fuel assemblies under emergency conditions have been performed for the following situations:

1. Filling the container with a steam-water mixture (insufficient drying of the SFAs during loading into a gas-cooled container; leakage in a container and partial replacement of the gas medium by water; leakage of a liquid-cooled container and partial release of liquid to the outside);
2. Destruction of the SFAs and fuel compaction in water or gas medium.

In the first situation, the case of filling the tank with a vapour-water mixture is considered and the influence of the water content in the mixture on the K_{eff} value is evaluated. As an example, containers are considered with nine SFAs of each fuel type arranged in a tightly packed grid in an aluminum or MBL05 basket. The vapour-water mixture is modelled with water of different densities: 25%, 50% and 75%.

In the second situation, it is assumed that in the event of a container collapse accident (when the SFAs are subjected to shock overloads), 10% of the fuel will escape from SFAs cladding and clump at the bottom of the container. This assumption is very conservative as the fuel yield estimate given in [12] showed that although up to 5% of the fuel cladding may be destroyed at this load, only 0.04% of the SFA fuel will leave the destroyed cladding. In the modelling the destroyed fuel is assumed to be a homogeneous mixture of uranium dioxide and air (or water), with the porosity of the mixture varying from 40% (sand porosity) to 70% (undestroyed fuel porosity).

Calculation results

The results of the criticality calculation of a container filled with a steam-water mixture with nine SFAs of different reactors arranged in a square densely packed grid in an aluminum or MBL05 basket are shown in Figures 12 and 13.

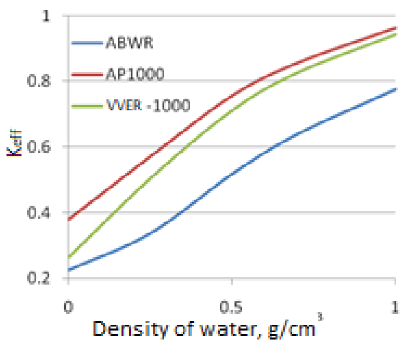

Figure 12. Criticality as function of water density (aluminum basket)

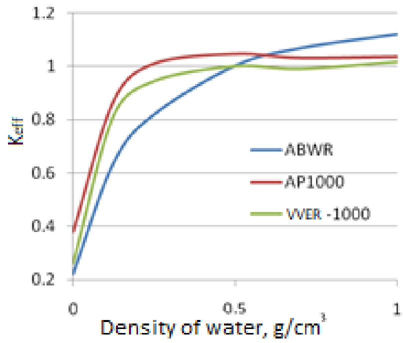

Figure 13. Criticality as function of water density (MBL05 alloy basket)

Evaluation of fuel destruction-criticality dependency

The results of the dependence of the criticality of a cask with SFAs from different reactors at 10% fuel destruction on the porosity of the packed fuel are given in Table 8.

Table 8. Dependence of criticality on the porosity of the packed fuel

Porosity, %	ABWR (37 SFAs)	AP1000 (12 SFAs)	VVER-1000 (8 SFAs)
Intact	0.86262	0.95173	0.94175
60	0.84752	0.94946	0.93658
50	0.84417	0.94503	0.93987
40	0.84636	0.94379	0.93701

Conclusions on nuclear safety in emergency situations

The results of the evaluation of the influence of the filling of the container cavity with the steam-water mixture on the criticality for different basket materials are different. For example, the K_{eff} value for SFAs in the MBL05 alloy basket increases monotonically with increasing density of the vapour-water mixture. Whereas for the aluminum basket, in addition to large K_{eff} values, a small maximum is observed for SFAs of VVER-1000 and AP1000 reactors at a density of the steam-water mixture of about 0.5 g/cm^3. All this indicates a lower level of nuclear safety of vessels with aluminum baskets and a preference for the use of baskets made of neutron-absorbing materials.

The results of the evaluation of the dependence of the criticality on the degree of compaction of the destroyed fuel show that there is no influence of the destroyed fuel on the criticality. Moreover, a slight decrease in the K_{eff} values for cases with destroyed fuel compared with intact fuel is explained by the loss of 10% of the uranium dioxide mass in the fuel elements.

The results of the assessments of SFA loading with restriction to criticality level are given in Table 9.

Table 9. Restrictions on the loading of SFAs from different reactors into TPs from a nuclear safety point of view

Options	SFAs q-ty		
	ABWR	AP1000	VVER-1000
Gas cooling	89	57	37
Water cooling (dense grid without neutron absorbers)	3	1	3
Water cooling (sparse grid without neutron absorbers)	45	12	21
Water cooling (dense grid with neutron absorbers)	57	21	15

Table 10. Recommended parameters of containers loaded with spent fuel assemblies from different type of reactors for different protection and cooling methods

Options	SFAs q-ty			Inner radius, cm	Shielding thickness	TPs mass, ton
	ABWR	AP1000	VVER-1000			
Gas cooling (or water cooling on dense grid with neutron absorbers)						
Reinforced concrete	9	4	3	40	85	104
R/c+DUO_2	37	8	9	80	45	129
Cast iron	21	6	7	60	60	130
R/c+DUO_2	36	8	9	75	45	130
Water cooling (on sparse grid without neutron absorbers)						
Reinforced concrete	8	4	4	45	80	103
R/c+DUO_2	25	8	8	75	45	125
Cast iron	21	5	8	70	50	130
R/c+DUO_2	21	6	8	65	45	129

Recommendations for the Selection of the Container Design

Neutron-physical calculations, radiation protection calculations and safety analysis are summarized in Table 10.

The container with depleted uranium oxide added to the shielding provides the highest loading; the container with cast iron-based shielding provides the best results. The lowest loading is provided by using the container with a pure reinforced concrete shield.

The influence of mass and dimensional constraints on the container design differs slightly when using different shielding variants. For example, the geometrical dimensions of the container are the main limiting parameter for reinforced concrete-based shielding and the weight of the container for cast-iron-based shielding.

If the container is filled with liquid, the dimensions of the container are reduced for most variants compared with a gas-filled container design. However, in the absence of an absorber in the basket, liquid-filled variants are more likely to have a lesser SFA load due to the need for greater space between SFAs to ensure that criticality limits are not exceeded.

Conclusion

The presented work is the final part of a two-stage complex research aimed at substantiating the feasibility to construct transport container design for nuclear spent fuel after short-term storage.

Computer simulation modelling methods and calculations were used to substantiate the radiation protection and nuclear safety solutions under normal and emergency operating conditions of the proposed TP (transport and packaging unit) designs for SFAs and to study the associated neutron-physical processes.

In the process of modelling, water and gas filling of containers was assumed in design solutions for heat removal inside the container in all considered operating modes and conditions of the TPs were proposed.

As a result of the work, the basic feasibility of containers for safe SNF management after short-term storage in the reactor was proposed, and the main technological and design requirements were determined for several optional TPU designs made of different materials, with different capacity and spatial arrangement of SFAs meeting all the necessary criteria for safe SNF transport. The most promising TP design options with maximum loading are identified as gas and liquid filled containers with an absorber in the basket that made of the following materials: reinforced concrete+DUO_2 and cast iron+DUO_2.

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КОМПЬЮТЕРДЕ ИМИТАЦИЯЛЫҚ МОДЕЛЬДЕУ ӘДІСТЕРІН ПАЙДАЛАНА ОТЫРЫП, ҚЫСҚА УАҚЫТ ҰСТАЛҒАН ПАЙДАЛАНЫЛҒАН ЯДРОЛЫҚ ОТЫНҒА АРНАЛҒАН ТАСЫМАЛДАУ КОНТЕЙНЕРІН ҚОЛДАНУДЫҢ ТЕХНИКАЛЫҚ МҮМКІНДІГІН НЕГІЗДЕУГЕ АРНАЛҒАН НЕЙТРОНДЫҚ-ФИЗИКАЛЫҚ ЕСЕПТЕУЛЕР

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Бұл жұмыс екі сатылы кешенді зерттеудің соңғы сатысы болып саналады, ол ПЯО-ны қысқа уақыт ұстап барып тасымалдау мүмкіндігін негіздеуге және ТҚК конструкциясына қойылатын талаптарды әзірлеуге бағытталған. Жұмыста қысқа уақыт ұсталған пайдаланылған ядролық отынды тасымалдау үшін тасымалдау контейнерін қолданудың техникалық мүмкіндігін негіздеу бойынша зерттеу нәтижесі ұсынылған. Радиоактивті сәулеленуден қорғау бойынша есептеу жүргізілді, ядролық қауіпсіздік негізделді, нейтрондық-физикалық ілеспе процестер зерделенді. Жүргізілген жұмыстардың нәтижесінде пайдаланылған ядролық отынға арналған тасымалдау контейнерін қолданудың техникалық мүмкіндігі негізделді, конструкциясы темір-бетон + уран диоксиді және шойын + уран диоксиді материалынан жасалған, газбен толтырылған немесе сіңіргіш сұйықтықпен толтырылған себеттегі пайдаланылған ядролық отынға арналған контейнердің ең қолайлы нұсқасы ұсынылған.
Мақсатқа жету үшін жұмыста компьютерлік модельдеу әдістері қолданылды, қалыпты және авариялық пайдалану жағдайында радиациядан қорғану мен ядролық қауіпсіздікті негіздеу үшін есептеулер жүргізілді, ПЯО салынған контейнерімен ілесе жүретін нейтрондық-физикалық процестер зерттелді. Материалының қалыңдығына және ПЖБЖ контейнерінің толтырылуына байланысты радиациядан қорғануға арналған материалдардың бірнеше нұсқасы қарастырылды.

Түйін сөздер: *ядролық отын, ядролық отын циклі, ядролық қауіпсіздік, пайдаланылған ядролық отын контейнері, ядролық реактор.*

НЕЙТРОННО-ФИЗИЧЕСКИЕ РАСЧЁТЫ В ОБОСНОВАНИЕ ТЕХНИЧЕСКОЙ ВОЗМОЖНОСТИ ПРИМЕНЕНИЯ ТРАНСПОРТИРОВОЧНОГО КОНТЕЙНЕРА ДЛЯ ОТРАБОТАННОГО ЯДЕРНОГО ТОПЛИВА ПОСЛЕ КРАТКОВРЕМЕННОЙ ВЫДЕРЖКИ С ИСПОЛЬЗОВАНИЕМ МЕТОДОВ КОМПЬЮТЕРНОГО ИМИТАЦИОННОГО МОДЕЛИРОВАНИЯ

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Данная работа является завершающим этапом двухстадийных комплексных исследований, направленных на обоснование возможности перемещения ОЯТ после кратковременной выдержки и выработку требований к конструкции ТУК.
В работе представлены результаты исследований по обоснованию технической возможности применения транспортировочного контейнера для транспортировки отработанного ядерного топлива после кратковременной выдержки. Выполнены расчеты защиты от радиоактивного излучения, обоснована ядерная безопасность, изучены соответствующие сопровождающие нейтронно-физические процессы. В результате проведенных работ обоснована техническая возможность применения транспортировочного контейнера для отработанного ядерного топлива, предложен наиболее приемлемый вариант контейнера для отработанного ядерного топлива с газовым заполнением или с жидкостным заполнением поглотителя в корзине с конструкцией из материалов: железобетон + диоксид урана и чугун + диоксид урана.
Для достижения поставленной цели в работе использовались методы компьютерного имитационного моделирования, проводились расчеты в обоснование радиационной защиты и ядерной безопасности в нормальных и аварийных условиях эксплуатации, изучались нейтронно-физические процессы, сопровождающие ТУК с ОЯТ. В работе рассмотрены несколько вариантов материалов для радиационной защиты, в зависимости от их толщины и наполняемости контейнера ОТВС.

Ключевые слова: *ядерное топливо, ядерный топливный цикл, ядерная безопасность, контейнер для отработанного ядерного топлива, ядерный реактор.*