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THERMAL ANALYSIS FOR THE SPENT FUEL TRANSPORTATION PACKAGE AFTER SHORT-TERM COOLING BY USING COMPUTER SIMULATION MODELING METHODS

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The paper presents the results of studies to substantiate the possibility of using a container for transfer of spent nuclear fuel after the short-term cooling. Spent nuclear fuel source-terms and energy release calculations, as well as thermal hydraulic calculations have been made to serve as the basis for further studies.
The proposed transport package is designed for transportation of recently discharged spent nuclear fuel with a high level of radioactivity. It is assumed that the use of this package allow transportation of up to 5 fuel assemblies from high-power reactors such as ABWR, AP1000, and VVER-1000.
As a result of this study, the possibility of using a transport package for spent fuel assemblies after the short-term cooling is substantiated in terms of the thermal processes that occur within them. These processes impose specific restrictions on the container design.
This work is the first part of a comprehensive study aimed to substantiate the possibility of transporting spent fuel after the short-term cooling and develop requirements for the design of such transport packages.

Keywords: *spent nuclear fuel, thermal processes, nuclear reactor, transport package, container, transportation.*

Introduction

Spent nuclear fuel (SNF) management – is the final stage of the nuclear fuel cycle. A number of technological operations take place during this stage such as the discharge of spent fuel assemblies (SFAs) from the reactor, the placement of SFAs at the plant site, and the transportation of SFAs to the centralized long-term dry storage or the nuclear reprocessing center and etc. [1]. Under the current approach, SFAs are stored for at least a year on the NPP's site, before moving to the centralized storage or to the reprocessing. The storage of SFAs on the plant's site requires the implementation of many complex operations, the presence of special facilities and free space in them [2, 3].

The paper proposes to consider an approach to SNF management that includes a number of operations for the storage of SFA at the near station area. This approach includes placing the SNF, after the short-term cooling, into special transport packages (TP) and sending them to either the reprocessing plant or temporary storage areas.

The proposed approach to the SFA's management has a number of benefits such as a lower risk of nuclear materials proliferation within the NPP territory, reducing doses on personnel responsible for storing SFAs and many others.

In the scheme proposed for SFA's management, materials and design of the container play a crucial role. The design should ensure safe thermal conditions, nuclear and radiation safety, radiation protection, radioactive materials safety, integrity, and sealing even in the case of serious incidents and accidents [4, 5].

The objective of this study is to consider the possibility of developing and designing transport packages for spent fuel after the short-term cooling in terms of the acceptability of TP thermal properties. To reach this objective, the methods of computer simulation modelling were used to calculate and study radiation intensity, SNF energy release and the heat transfer processes within a package with spent fuel. The results of this study will be used in further calculations.

Initial Data, Approaches and Assumptions

The TP being considered in this study is categorized as the type B(U)F package according to IAEA regulations [6] and as the type B, nuclear safety class I, radiation hazard category III according to national regulations [7–9].

It is assumed in this study that TPs will be used to transport fuel assemblies of ABWR, AP1000 and VVER-1000 reactors.

In accordance with the requirements of regulatory documents of the Republic of Kazakhstan and IAEA recommendations it is accepted that the temperature of easily accessible surface of the TP being considered in the absence of insolation should not be more than 85 °C.

Based on the data from [10, 11] it has been determined that maximum working temperature limit for fuel element cladding from zirconium alloys should not be more than 350 °C, because at temperatures above 350 °C the strength properties deteriorate.

The insolation parameters are taken as follows from [12]: 942 W/m^2 for the horizontal surface over a 12-hour period, and 532 W/m^2 for the vertical surface.

The SNF loaded into the package has a high activity, and the integral energy release of the fuel in the package approaches 300 kW. Removing such a rather high energy release is usually achieved by organizing the forced movement of the coolant with the subsequent transfer of heat from the coolant to the environment. To transfer heat from the coolant to the environment, using a dry cooler (dry cooling tower) in the packaging set is optimal.

In order to facilitate the protection against the release of radioactive products, the package cooling system is made according to a two-circuit scheme: the coolant of the first circuit fills the cavity of the container and contacts with fuel elements of FAs, the coolant of the second circuit removes the heat from the TP into the environment.

Water or gas (helium, argon, CO_2) were considered as the working body filling the cavity of the package set.

Table 1 shows the fuel specification of ABWR, AP1000, and VVER-1000 reactor plants.

Table 1. Technical characteristics of fuel assemblies

Characteristic	Reactor		
	ABWR	AP1000	VVER-1000
Grid structure of a fuel assembly	10×10 (square)	17×17 (square)	triangular
Number of fuel rods in an assembly	92	264	312
Enrichment, % ^{235}U	3-4	4.8	4.0
Average fuel burnup, MW·day/tU	50000	60000	43000
Material of rod cladding	Zircaloy-2	ZIRLO	Zr+1%Nb
Outer diameter of fuel rod, mm	10.3	9.5	9.1
Fuel rod cladding thickness, mm	0.66	0.57	0.67
Overall Fuel Assembly length, mm	4470	4795	4570

ASSESSMENT OF ENERGY RELEASE AND RADIATION CHARACTERISTICS

To check all further calculations there were built verification models based on the experimental data for the same or similar types of reactors [13–16]:

- The data of the Takahama-3 reactor was selected to verify the AP1000 type reactor: JPNNT3PWR-4 (36.7 GW·d/tU) and JPNNT3PWR-14 (47.0 GW·d/tU). The initial uranium isotopic composition was 0.04% ^{234}U, 4.11% ^{235}U and 95.85% ^{238}U.
- The data of the Fukushima-Daini-2 reactor was selected to verify the ABWR type reactor: JPN2F2BWR-5 (43.99 GW·day/tU) and JPN2F2BWR-13 (37.41 GW·day/tU). The initial uranium isotopic composition was 0.03% ^{234}U, 3.41% ^{235}U and 96.56% ^{238}U.
- For the VVER-1000 type reactor, data were selected for two samples of two different assemblies with maximum burnup: No. 195-720 (51.7 GW·day/tU; FA number 4433001114) and No. 581 (47.9 GW·day/tU; FA E-1591, fuel element 23).

After generation of models in the TRITON module of the SCALE 5 software complex [17] with the reactor parameters and performing calculations on them, it was found that in the well-studied range of burnup levels (from 10 to 50 GW·day/tU) and conditioning (from 2 to 10 years), the results of calculation are well consistent with the experimental data. For example, it was found that the average ratio of the calculated and experimental values of the energy release level for BWR was 1.005 with a standard deviation of 2.4%.

After verification calculations, the models of considered reactors ABWR, AP1000 and VVER-1000 types were built.

The results of assessment of the energy release rate within the cooling time range of 3–30 days for considered reactors in dependence to their burnup are shown in Table 2.

Table 2. Energy release, W/kgU

Burn-up, MW·day/tU	Cooling time, days								
	ABWR			AP1000			VVER-1000		
	3	10	30	3	10	30	3	10	30
18	94	57	34	144	87	51	155	92	53
36	102	62	38	154	93	56	168	100	60
54	110	67	42	165	99	61	179	106	65

The results of assessment of neutron radiation source intensity for 54 MW·day/tU burnup after 10 days cooling are shown in Table 3.

Table 3. Neutron Radiation Intensity, neutron/(s·kgU)

Range, MeV	ABWR	AP1000	VVER-1000
$1.00 \cdot 10^{-11}$–$1.00 \cdot 10^{-8}$	$2.35 \cdot 10^{-6}$	$1.70 \cdot 10^{-6}$	$1.94 \cdot 10^{-6}$
$1.00 \cdot 10^{-8}$–$3.00 \cdot 10^{-8}$	$5.58 \cdot 10^{-6}$	$3.58 \cdot 10^{-6}$	$4.36 \cdot 10^{-6}$
$3.00 \cdot 10^{-8}$–$5.00 \cdot 10^{-8}$	$7.32 \cdot 10^{-6}$	$4.53 \cdot 10^{-6}$	$5.62 \cdot 10^{-6}$
$5.00 \cdot 10^{-8}$–$1.00 \cdot 10^{-7}$	$2.41 \cdot 10^{-5}$	$1.47 \cdot 10^{-5}$	$1.83 \cdot 10^{-5}$
$1.00 \cdot 10^{-7}$–$2.25 \cdot 10^{-7}$	$8.63 \cdot 10^{-5}$	$5.19 \cdot 10^{-5}$	$6.52 \cdot 10^{-5}$
$2.25 \cdot 10^{-7}$–$3.25 \cdot 10^{-7}$	$8.94 \cdot 10^{-5}$	$5.36 \cdot 10^{-5}$	$6.75 \cdot 10^{-5}$
$3.25 \cdot 10^{-7}$–$4.00 \cdot 10^{-7}$	$7.72 \cdot 10^{-5}$	$4.63 \cdot 10^{-5}$	$5.83 \cdot 10^{-5}$
$4.00 \cdot 10^{-7}$–$8.00 \cdot 10^{-7}$	$5.29 \cdot 10^{-4}$	$3.18 \cdot 10^{-4}$	$4.00 \cdot 10^{-4}$
$8.00 \cdot 10^{-7}$–$1.00 \cdot 10^{-6}$	$3.24 \cdot 10^{-4}$	$1.94 \cdot 10^{-4}$	$2.44 \cdot 10^{-4}$
$1.00 \cdot 10^{-6}$–$1.13 \cdot 10^{-6}$	$2.29 \cdot 10^{-4}$	$1.37 \cdot 10^{-4}$	$1.73 \cdot 10^{-4}$
$1.13 \cdot 10^{-6}$–$1.30 \cdot 10^{-6}$	$3.19 \cdot 10^{-4}$	$1.91 \cdot 10^{-4}$	$2.41 \cdot 10^{-4}$
$1.30 \cdot 10^{-6}$–$1.77 \cdot 10^{-6}$	$9.90 \cdot 10^{-4}$	$5.92 \cdot 10^{-4}$	$7.46 \cdot 10^{-4}$
$1.77 \cdot 10^{-6}$–$3.05 \cdot 10^{-6}$	$3.36 \cdot 10^{-3}$	$2.01 \cdot 10^{-3}$	$2.53 \cdot 10^{-3}$
$3.05 \cdot 10^{-6}$–$1.00 \cdot 10^{-5}$	$2.97 \cdot 10^{-2}$	$1.77 \cdot 10^{-2}$	$2.23 \cdot 10^{-2}$
$1.00 \cdot 10^{-5}$–$3.00 \cdot 10^{-5}$	$1.50 \cdot 10^{-1}$	$8.96 \cdot 10^{-2}$	$1.13 \cdot 10^{-1}$
$3.00 \cdot 10^{-5}$–$1.00 \cdot 10^{-4}$	$9.53 \cdot 10^{-1}$	$5.71 \cdot 10^{-1}$	$7.18 \cdot 10^{-1}$
$1.00 \cdot 10^{-4}$–$5.50 \cdot 10^{-4}$	$1.35 \cdot 10^{1}$	8.11	10.20
$5.50 \cdot 10^{-4}$–$3.00 \cdot 10^{-3}$	$1.72 \cdot 10^{2}$	$1.03 \cdot 10^{2}$	$1.30 \cdot 10^{2}$
$3.00 \cdot 10^{-3}$–$1.70 \cdot 10^{-2}$	$2.32 \cdot 10^{3}$	$1.39 \cdot 10^{3}$	$1.75 \cdot 10^{3}$
$1.70 \cdot 10^{-2}$–$1.00 \cdot 10^{-1}$	$3.25 \cdot 10^{4}$	$1.95 \cdot 10^{4}$	$2.45 \cdot 10^{4}$
$1.00 \cdot 10^{-1}$–$4.00 \cdot 10^{-1}$	$2.20 \cdot 10^{5}$	$1.32 \cdot 10^{5}$	$1.66 \cdot 10^{5}$
$4.00 \cdot 10^{-1}$–$9.00 \cdot 10^{-1}$	$4.79 \cdot 10^{5}$	$2.88 \cdot 10^{5}$	$3.62 \cdot 10^{5}$
$9.00 \cdot 10^{-1}$–1.40	$4.79 \cdot 10^{5}$	$2.89 \cdot 10^{5}$	$3.64 \cdot 10^{5}$
1.40–1.85	$3.86 \cdot 10^{5}$	$2.34 \cdot 10^{5}$	$2.94 \cdot 10^{5}$
1.85–3.00	$7.37 \cdot 10^{5}$	$4.54 \cdot 10^{5}$	$5.66 \cdot 10^{5}$
3.00–6.43	$6.72 \cdot 10^{5}$	$4.13 \cdot 10^{5}$	$5.15 \cdot 10^{5}$
6.43–20.00	$6.11 \cdot 10^{4}$	$3.61 \cdot 10^{4}$	$4.57 \cdot 10^{4}$
Total	$3.07 \cdot 10^{6}$	$1.87 \cdot 10^{6}$	$2.34 \cdot 10^{6}$

The results of assessment of gamma radiation source intensity for 54 MW·day/tU burnup after 10 days cooling are shown in Table 4.

Table 4. Gamma Radiation Intensity, photon/s·kgU

Range, MeV	ABWR	AP1000	VVER-1000
$1.00\cdot10^{-2}$–$5.00\cdot10^{-2}$	$1.16\cdot10^{14}$	$1.79\cdot10^{14}$	$1.92\cdot10^{14}$
$5.00\cdot10^{-2}$–$1.00\cdot10^{-1}$	$4.29\cdot10^{13}$	$6.69\cdot10^{13}$	$7.11\cdot10^{13}$
$1.00\cdot10^{-1}$–$2.00\cdot10^{-1}$	$5.68\cdot10^{13}$	$8.75\cdot10^{13}$	$9.53\cdot10^{13}$
$2.00\cdot10^{-1}$–$3.00\cdot10^{-1}$	$2.20\cdot10^{13}$	$3.30\cdot10^{13}$	$3.67\cdot10^{13}$
$3.00\cdot10^{-1}$–$4.00\cdot10^{-1}$	$2.18\cdot10^{13}$	$3.41\cdot10^{13}$	$3.67\cdot10^{13}$
$4.00\cdot10^{-1}$–$6.00\cdot10^{-1}$	$7.62\cdot10^{13}$	$1.11\cdot10^{14}$	$1.22\cdot10^{14}$
$6.00\cdot10^{-1}$–$8.00\cdot10^{-1}$	$1.11\cdot10^{14}$	$1.75\cdot10^{14}$	$1.83\cdot10^{14}$
$8.00\cdot10^{-1}$–1.00	$2.04\cdot10^{13}$	$2.92\cdot10^{13}$	$3.12\cdot10^{13}$
1.00–1.33	$7.17\cdot10^{12}$	$8.81\cdot10^{12}$	$9.87\cdot10^{12}$
1.33–1.66	$3.06\cdot10^{13}$	$4.98\cdot10^{13}$	$5.29\cdot10^{13}$
1.66–2.00	$9.25\cdot10^{11}$	$1.13\cdot10^{12}$	$1.35\cdot10^{12}$
2.00–2.50	$1.76\cdot10^{12}$	$2.29\cdot10^{12}$	$2.65\cdot10^{12}$
2.50–3.00	$9.50\cdot10^{11}$	$1.56\cdot10^{12}$	$1.65\cdot10^{12}$
3.00–4.00	$8.13\cdot10^{9}$	$1.31\cdot10^{10}$	$1.40\cdot10^{10}$
4.00–5.00	$1.04\cdot10^{5}$	$6.33\cdot10^{4}$	$7.99\cdot10^{4}$
5.00–6.50	$4.17\cdot10^{4}$	$2.54\cdot10^{4}$	$3.21\cdot10^{4}$
6.50–8.00	$8.18\cdot10^{3}$	$4.98\cdot10^{3}$	$6.29\cdot10^{3}$
8.00–10.00	$1.74\cdot10^{3}$	$1.06\cdot10^{3}$	$1.33\cdot10^{3}$
Total	$5.09\cdot10^{14}$	$7.79\cdot10^{14}$	$8.36\cdot10^{14}$

THERMAL ANALYSIS

During the calculations, it was assumed that the heat exchange inside the TP occurs through the free gas and water convection in the cavity of package. The heat is removed from the working body, which fills the package, and transferred to the package's walls which have cooling paths with forced water circulation. Alternatively, the heat can be transferred to a system of pipes that pass through a cavity where cooling water is also circulating.

To simplify the thermal-physical modeling:

− AP1000 assembly was chosen for further calculation;

− the external water jacket with forced convective heat removal was not considered, but to assess the impact convective heat exchange with the heat transfer coefficient was set 2000 W/(m^{2}K) [18].

The water movement within the cavity of package caused by natural liquid convection, was described using the Boussinesq Model [19] during the calculations.

The thermophysical properties of water in the calculations have been set as follows:

− density at the temperature of 290 K was set equal to 990 kg/m3;

− thermal conductivity – 0.6 W/(m·K);

− heat capacity – 4182 J/(kg·K);

− viscosity – 0.00103 kg/(m·s);

− coefficient of thermal expansion – 0.00035 1/K.

The gas movement caused by natural convection was described using the model created on ideal gas ratio (Clayperon-Mendeleev equation).

Thermophysical properties of gas used during the calculations, are shown below (Tables 5–7) [20, 21].

Table 5. Heat capacity and molecular weight of gas

Gas	Helium	CO_2	Argon
Heat capacity, J/(kg·K)	5204	1050	519
Molecular mass, kmol/kg	4	28	40

Table 6. Dependence of thermal conductivity of gases [W/(m·K)] on temperature

Temperature, K	273	373	473	573	673	773	873
Helium	0.143	0.179	0.212	0.245	0.275	0.305	0.333
CO_2	0.023	0.030	0.036	0.043	0.049	0.054	0.060
Argon	0.0165	0.0212	0.0256	0.0299	0.0339	0.0379	0.0394

Table 7. Dependence of gas viscosity [kg/(m·s)]·10^5 on temperature

Temperature, K	273	373	473	573	673	773	873
Helium	1.86	2.29	2.67	3.05	3.40	3.74	4.07
CO_2	1.66	2.07	2.44	2.80	3.12	3.44	3.74
Argon	2.11	2.70	3.22	3.69	4.11	4.52	4.85

In the case of filling TP container cavity with gas, radiant heat exchange had been calculated using the spherical harmonics method in P-1 approximation [19].

In calculations, it was assumed that the internal containment of package was made of steel. The emissivity factor of package steel containment and zirconium cladding of fuel elements equaled to 0.5.

Considering that 5 fuel assemblies are placed in the package in a “cross” shape, the diameter of the package cavity should be about 0.8 m. Since the length of package is significantly greater than its diameter, the main heat outflows through the side walls, therefore, heat removal through the end walls of the container was disregarded in the calculations. Thermal calculations were carried out in the ANSYS software package using a two-dimensional P1 heat exchange calculation model in the horizontal orientation and the finite element method.

Calculations of TP with Water Filling the Container Cavity

The calculation scheme of TP loaded with 5 SFAs of AP1000 reactor in it as shown in Figure 1. To provide the symmetry in the calculations, ½ (one half) of the package structure had been used. Outside temperature of 300K were set as boundary conditions at the outer wall of the package containment.

Boiling water in the cavity of the TP can lead to a significant increase in pressure in its cavity and damage the integrity of its design construction, so the use of water as a coolant implies that its temperature should not exceed 373 K. In order to comply with this condition, at least, it is necessary that the temperature difference at “fuel elements – water” border and “water – wall of the package” border does not exceed 10…20 degrees.

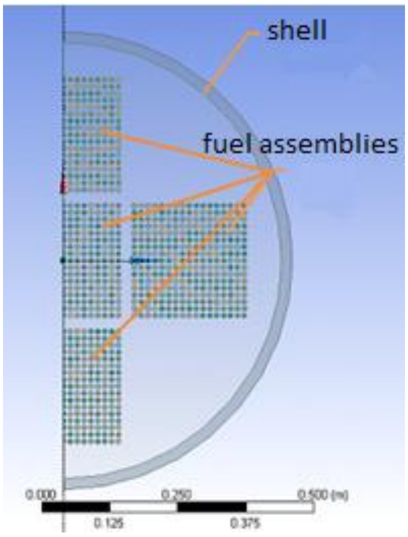

Figure 1. TP Calculation Scheme

The fuel elements of SFA have well developed heat exchange surface so even in minimal value of heat transfer coefficient for water under the free convection [18] equal to 100 W/(m^2·K). The temperature drop among the fuel elements and water within the package cavity doesn't exceed several degrees.

To find the required surface area of heat exchange between the water and the package wall, the coefficient of heat transfer at the boundary "water – wall of the package containment" at the initial stage of the calculations was determined. Calculations showed (Figure 2) that this heat transfer coefficient is ~450 W/(m^2·K).

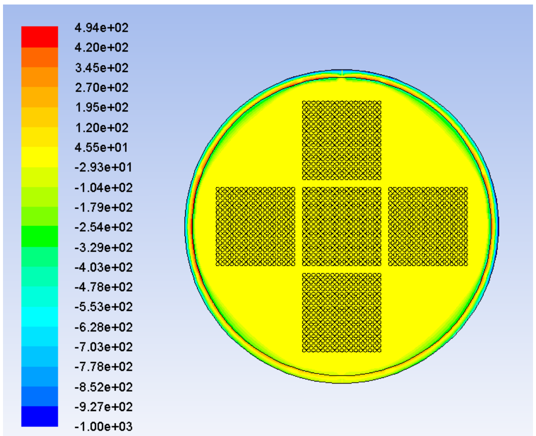

Figure 2. Heat Transfer Coefficient at the Boundary "Water – Package Containment"

The running rate of energy release per unit of TP length is 67.7 kW/m. Considering that the temperature drop between the water and package wall is about 15 °C, we can calculate that the perimeter of heat exchange surface of the package containment internal wall shall be 10 m.

The required value of the heat exchange surface area can be reached by ribbing of the containment internal wall. Estimated temperature field in the cavity of TP with ribbed internal wall of the containment is shown in the Figure 3. In calculations, it was assumed that the rib height is 40 mm, the rib thickness is 5 mm, and 80 ribs are located along the perimeter of the containment.

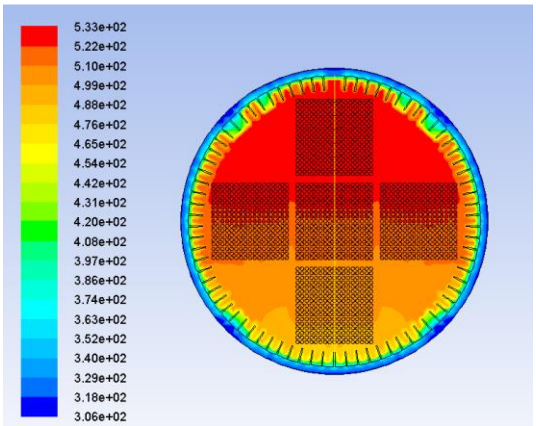

Figure 3. Temperature Field inside the TP [K] Cavity with Ribbed Internal Wall of the Containment

The analysis of calculation results showed that the ribbing of containment wall doesn't bring the expected result: stagnant zones formed in spaces between the ribs, convective water flows were low, the decisive role in the heat exchange process played the conductive component, and water temperature in the cavity is significantly above the boiling temperature.

Heat removal from water, which placed in the package cavity, cannot be implemented to the wall of package, but to the pipes with circulating coolant that pass through the cavity. The calculated temperature field of such system is shown in Figure 4.

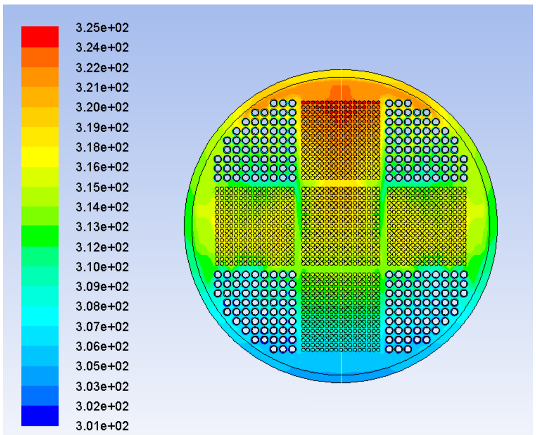

Figure 4. Temperature Field [K] in Package Filled with Water with Array of Pipes Passed through the Cavity

In carrying out this calculation it was assumed that 256 pipes with an outer diameter of 20 mm and wall thickness of 2 mm were passed through the cavity. Convective heat exchange with heat transfer coefficient equal to 2000 W/(m^2·K) and coolant temperature equal to 300 K was applied for the inner surface of pipes.

Calculations of TP with Gas Filling the Package Cavity

Calculations of TP with gas filled package cavity were performed using calculation schemes, which describes the design of the set, similar to the schemes used for calculations of TP with package filled with water.

Two ribbing options were considered: one with 80 ribs и another with 160 ribs placed along the perimeter of the containment wall. The ribs were 40 mm height and 5 mm thick.

Maximum temperature of the package materials during the filling the TP cavity with different gases (helium, CO_2, argon) obtained as a result of calculations and the above described methods of heat removal from the fuel is shown in Table 8.

To demonstrate the results of the calculations, Figures 5–7 show the distribution of the temperature field in the helium-filled TP cavity without ribbed inner wall of the package containment, with ribbed inner wall of the containment and with tubes passing through the package cavity.

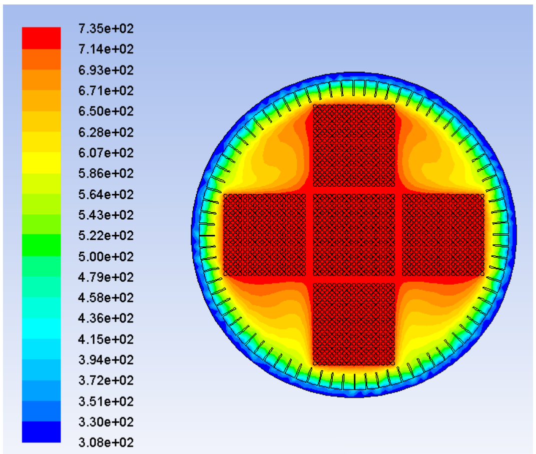

Figure 5. Distribution of Temperature Field in the Helium-Filled Cavity of TP with Ribbed Inner Wall of the Package Containment (80 Ribs around the Containment Perimeter)

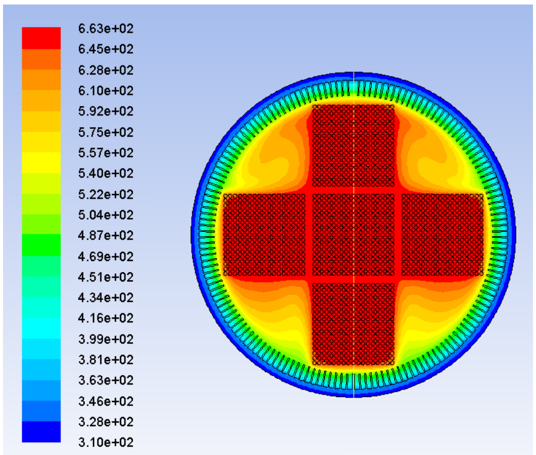

Figure 6. Distribution of Temperature Field in the Helium-Filled Cavity of TP with Ribbed Inner Wall of the Package Containment (160 Ribs around the Containment Perimeter)

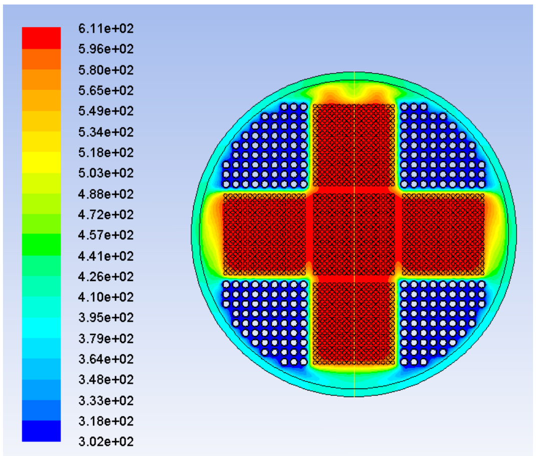

Figure 7. Distribution of Temperature Field in the Helium-Filled Cavity of TP with Pipes Passed through the Package

Table 8. Maximum temperature of the package materials (K) calculated using different methods of heat removal from fuel

Gas	Helium	CO_2	Argon
Without ribbing of containment	906	918	924
Rare ribbing of containment	735	742	746
Frequent ribbing of containment	663	666	667
Pipes passed through the cavity	611	612	615

CONCLUSION

This work is the first part of a complex research aimed at justification of principal feasibility for creation of a transport package designed to transport SNF after the short-term cooling.

An array of thermophysical data has been obtained through computer simulation modeling and calculation methods. Additionally, conceptual design and technological solutions have been proposed to ensure the operability and safety of such TP in terms of thermophysical processes occurring within it.

Analyzing the results of calculations, it can be stated that the type and composition of the gas filling the container cavity has minimal impact on the temperature of materials in the TP. This can be explained by the fact that the heat exchange by radiation is predominant in the heat exchange between the fuel and the cooling wall.

The obtained results show that the best solution of SFA cooling conditions in the case of water and gas coolant is the use of an array of tubes passing through the container cavity. This enables to keep the temperature of the materials within the package, not exceeding the design temperature, and the TP of this design enables the provision of the necessary thermal modes for the device's operation.

Declaration of Competing Interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

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КОМПЬЮТЕРЛІК ИМИТАЦИЯЛЫҚ МОДЕЛЬДЕУ ӘДІСТЕРІН ҚОЛДАНА ОТЫРЫП, ҚЫСҚА УАҚЫТ ҰСТАЛҒАН ПАЙДАЛАНЫЛҒАН ЯДРОЛЫҚ ОТЫНҒА АРНАЛҒАН ТАСЫМАЛДАУ КОНТЕЙНЕРІН ЖЫЛУФИЗИКАЛЫҚ ТАЛДАУ

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Бұл жұмыста реактордан жаңа түсірілген пайдаланылған ядролық отынды тасу үшін контейнерді қолданудың техникалық мүмкіндігін негіздеу бойынша зерттеу нәтижесі ұсынылған. Пайдаланылған ядролық отынның энергия бөлу деңгейін және радиациялық сипаттамасын бағалау бойынша есептеу жүргізілді, бұл контейнердің жылуфизикалық есептеуіне негіз болды.
Контейнерді пайдаланып, ABWR, AP1000 және ВВЭР-1000 типті жоғары қуатты реакторлардың 5-ке дейінгі жылу бөлгіш жинағын тасымалдауға болады деген болжам бар.
Жұмыс нәтижесінде пайдаланылған жылу бөлгіш жинақтарды қысқа уақыт ұстау кезінде болатын жылу-физикалық процестердің контейнер конструкциясына белгілі бір шектеу қоятындығы тұрғысынан тасымалдау контейнерін қолданудың техникалық мүмкіндігі негізделеді.
Бұл жұмыс қысқа уақыт ұсталған пайдаланылған отынды тасу мүмкіндігін негіздеуге және контейнер конструкциясына қойылатын талаптарды әзірлеуге бағытталған кешенді зерттеулердің бірінші бөлігі болып саналады.

Түйін сөздер: *пайдаланылған ядролық отын, ядролық реактор, көлік-қаптау контейнері, тасымалдау.*

ТЕПЛОФИЗИЧЕСКИЙ АНАЛИЗ ТРАНСПОРТИРОВОЧНОГО КОНТЕЙНЕРА ДЛЯ ОТРАБОТАННОГО ЯДЕРНОГО ТОПЛИВА ПОСЛЕ КРАТКОВРЕМЕННОЙ ВЫДЕРЖКИ С ИСПОЛЬЗОВАНИЕМ МЕТОДОВ КОМПЬЮТЕРНОГО ИМИТАЦИОННОГО МОДЕЛИРОВАНИЯ

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В работе представлены результаты исследований по обоснованию технической возможности применения контейнера для перемещения свежевыгруженного из реактора отработанного ядерного топлива. Выполнены расчеты по оценке уровня энерговыделения и радиационных характеристик отработанного ядерного топлива, которые в свою очередь послужили основанием для дальнейших теплофизических расчётов контейнера.
Предполагается, что с использованием контейнера становится возможным перевозка от 5 тепловыделяющих сборок реакторов большой мощности типа ABWR, AP1000 и ВВЭР-1000.
В результате работы обоснована техническая возможность применения транспортировочного контейнера для отработанных тепловыделяющих сборок после кратковременной выдержки с точки зрения протекающих в них теплофизических процессов, накладывающих определенные ограничения на конструкцию контейнера.
Данная работа является первой частью комплексных исследований, направленных на обоснование возможности перемещения отработавшего топлива после кратковременной выдержки и выработки требований к конструкции контейнера.

Ключевые слова: *отработанное ядерное топливо, ядерный реактор, транспортный упаковочный контейнер, перевозка.*