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Experimental verification of computational models for evaluation of safety parameters during spent nuclear fuel transportation

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Abstract

Scientific and Engineering Centre for Nuclear and Radiation Safety (SECNRS) is technical support organization of the Federal Environmental, Industrial and Nuclear Supervision Service of Russia. SECNRS was established aiming to collect and apply new scientific knowledge for scientific and engineering support of nuclear and radiation safety regulation, including analysis and substantiation of criteria and requirements for nuclear and radiation safety. One of the main types of works that correspond with the goals of the SECNRS establishment is the safety assessment of nuclear facilities and activities in the field of nuclear energy. During safety assessment specialists of SECNRS often perform independent evaluation of safety parameters. Present paper focuses on computational models used in SECNRS for independent evaluation of safety parameters spent nuclear fuel (SNF) transportation and its verification.

Introduction

In the Russian Federation basic requirements for nuclear and radiation safety during SNF transportation are governed by the "Safety regulations for transport of radioactive material" (NP-053-04) [1], which generally correspond to the norms of the IAEA SSR-6 [2, 3]. According to NP-053-04 the main indicator of the nuclear safety is the effective neutron multiplication factor (K_{eff}), of an individual package shall not exceed 0.95 under the normal, ordinary and emergency transport conditions. The main indicator of the radiation safety during SNF transport is the maximum dose rate beyond package protection. Requirements of NP-053-04 set the following limits on the maximum dose rate beyond package protection:

- normal conditions:
  - 2 mSv/h on the external package surface (10 mSv/h during transportation under exclusive use);
  - 2 mSv/h on the outer vehicle surface.
  - 0,1 mSv/h in two meters from the outer vehicle surface;
- Accident conditions: 10 mSv/h in one meters from the outer vehicle surface.

According to NP-053-04, the main thermal characteristic during SNF transportation is the maximum temperature of any readily accessible surface of the package, which should not exceed 85 ºC. In addition, the technical specifications limit the maximum temperatures of the surface of fuel cladding and the surface of the gasket, which is located between the inner lid of the package and its body.

Justification of nuclear and radiation safety of SNF transport is generally prepared using the widely used and recognized software, most of which (SCALE [4] in particular) implements the Monte Carlo method. To estimate temperature during SNF transportation a widely used and recognised code ANSYS [5] is often used. ANSYS designed for solving a wide range of physical problems and is certified for calculation of the temperature state of elements of construction, equipment and pipelines of various purposes in the justification of nuclear power facilities safety. The conservative approach should be used during the development of computational models used to estimate of SNF transport safety parameters. But the specific approximations and assumptions accepted in the mathematical models used for evaluation of various safety parameters are fundamentally different.

When a computational model to analyze nuclear safety is being created, special attention is given to a detailed description of the spent fuel assembly (SFA) design, fissile nuclear materials and structural elements of SFA cover, while it's allowed to set a package shielding more roughly. When defining the nuclide composition, neutron absorbing elements, the initial enrichment, fuel mass, length and density of the fuel, all simplifying approximations and design assumptions of this computational model should lead to the maximum possible value of K_{eff}, thereby providing a conservative approach.

When creating a computational model for radiation safety parameters estimation, a conservative approach is the most detailed description of package shielding, especially the elements near which there is a sharp increase gamma- and neutron radiation. Thus all simplifications of a real package design should lead to an overestimation of dose rate levels beyond the package protection at the points (strictly speaking, a priori unknown), in which maximum dose rate values are achieved and normalized according to NP-053-04.

When creating a computational models for estimate temperature of constructional elements, conservative approach is the replacement of complex heterogeneous areas in which heat transfer is carried out by conduction, convection and radiation, by equivalent homogeneous areas, in which heat transfer is performed only by thermal conductivity. At the same time thermal parameters of these areas are selected so that boundary conditions of the original (not simplified) problem are preserved.

An example of used in SECNRS approaches some results of computational-experimental researches are presented in this paper. These researches included study of nuclear safety parameters and computational-experimental studies of radiation safety parameters and thermal characteristics of VVER-1000 SNF transportation in cask TUK-153 under normal and accident conditions.

Study of nuclear safety parameters

Authors developed a computational models for K_{eff} calculations for cask TUK-153 loaded with 18

VVER-1000 SFA using the functional module KENO [6], which is a part of SCALE. These calculations were carried out using the library of continuous energy cross sections based on the evaluated nuclear data files ENDF/B-VII [7]. All simplifications of the actual packaging design which were accepted during the development of the computational model, such as lack of spacing grids, use of fuels with a maximum initial enrichment, etc., are conservative and lead to an overestimation of the calculated value of K_{eff}. Cross sections of the computational model of TUK-153 with SNF for K_{eff} calculation in normal conditions of transport are shown in Figure 1.

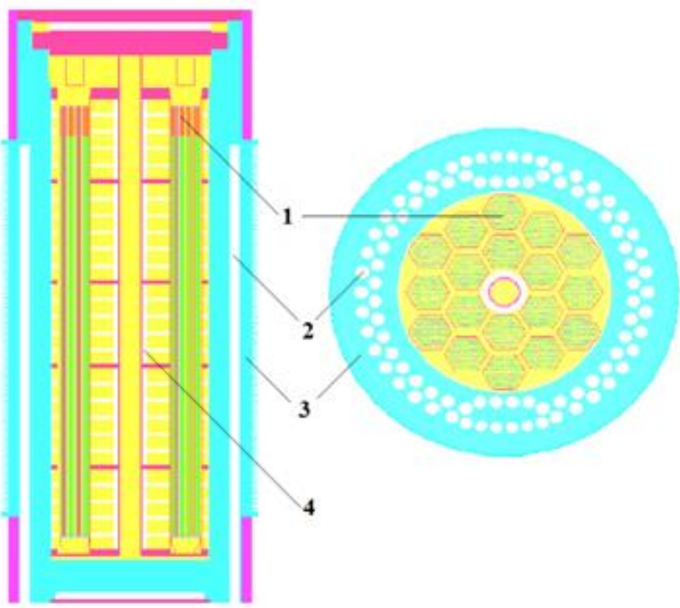
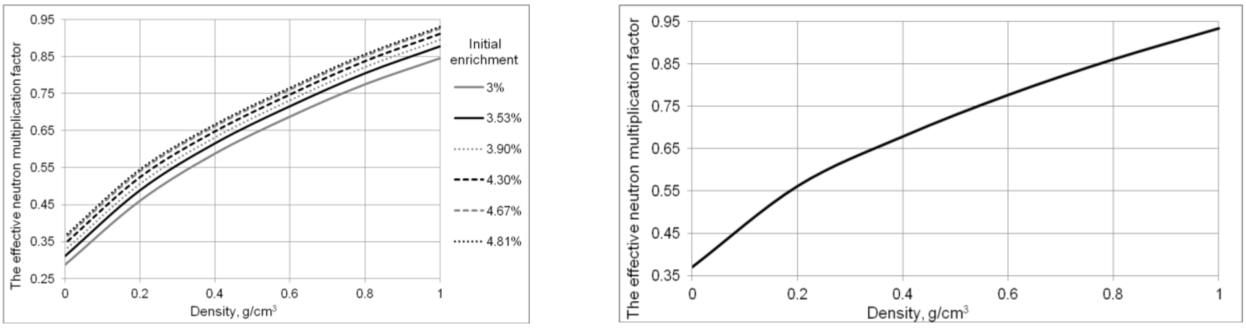


Figure 1. Cross sections of the computational model of TUK-153 with SNF for K_{eff} calculation in normal conditions of transport (1 – SFA, 2 – high molecular weight polyethylene, 3 – ductile cast iron, 4 - basket)

By using the created computational model, the K_{eff} values were calculated for normal conditions of SNF transportation and under accident, which are characterized by variation of the level, distribution and density of water in the cask. According to the results of K_{eff} calculations using conservative approach of "fresh fuel", for the maximum enrichment of ^{235}U under normal conditions K_{eff} value is 0.365 for a separate package and 0.371 for a group of packages.

Currently, during justification of nuclear safety there is a widely used approach, which is just a change of steam-water mixture density in the packaging. Dependences of K_{eff} values of a single package (depends on initial enrichment) and groups of packages on the steam-water mixture density in the packing are shown in Figures 2 (a) and 2 (b), respectively.

As can be seen from figure 2 (a, b), the nuclear safety ensuring during transportation of TUK-153 with VVER-1000 SFA under normal conditions of transport and for accident with a uniform filling of TUK-153 with different density of steam-water mixture is confirmed by the results of K_{eff} calculations carried out using very conservative "fresh fuel" approach (without more complex «Burn-up-credit» approaches [8]).



(a) (b)

Figure 2. Dependence of K_{eff} on the steam-water mixture density in the TUK-153 for a single package (depends on initial enrichment) – (a) and for an infinite number of packages (^{235}U initial enrichment 4,925 %) – (b)

However, NP-053-04 require the need to consider not only the change of water density, but also the change of its distribution in the packing. Hypothetical scenario of water penetration into TUK-153 can be implemented in the fall of TUK-153 from the bridge to the rock. As a result of such fall there may occur depressurization of the package with subsequent filling it with water. At the same time the speed of filling the various areas of the package will depend on their effective geometric cross section.

Based on the analysis of packing constructional features there is developed a computational model of TUK-153 loaded with 18 VVER-1000 SFA designed for estimation of K_{eff} in case of an accident with simultaneous but non-uniformly change of water level in the duct, the central pipe and the space between the basket and the body of TUK 153 (see Figure 3).

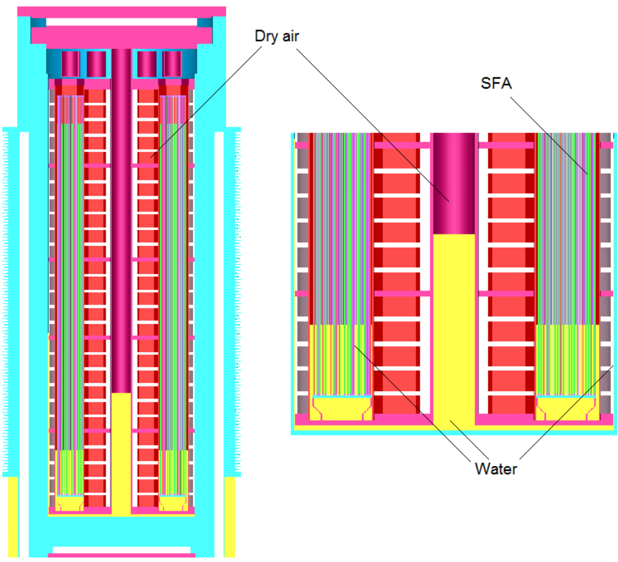


Figure 3. Cross section of the computational model of TUK-153 with SNF for K_{eff} calculation in accident conditions of transport

Using a very rough estimate of relations of volume rates of water flow to the duct, the central pipe and the space between the basket and the body of TUK 153, which is based on the various geometric sections of the air voids of the structural elements, it can be concluded that water will arrive to the duct approximately two times slower than to the central pipe and three times slower than to the space between the basket and the body of TUK 153. Obtained by this approximation, the dependence of K_{eff} on the water level in the duct is shown in Figure 4.

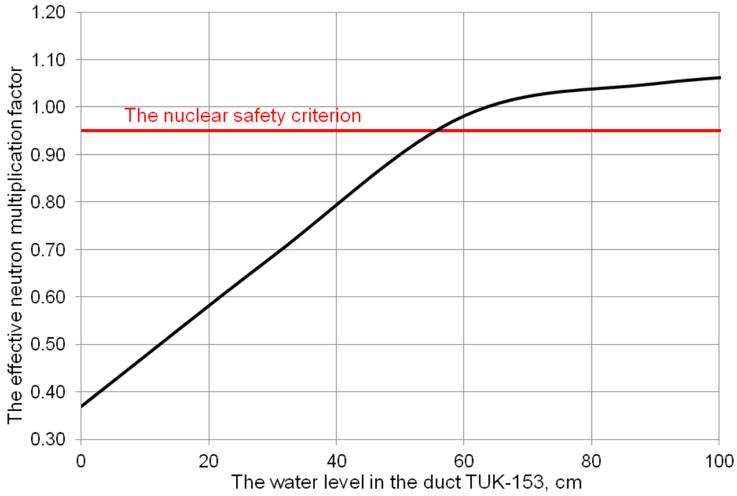


Figure 4. Dependence of K_{eff} on the water level in the duct for SFA

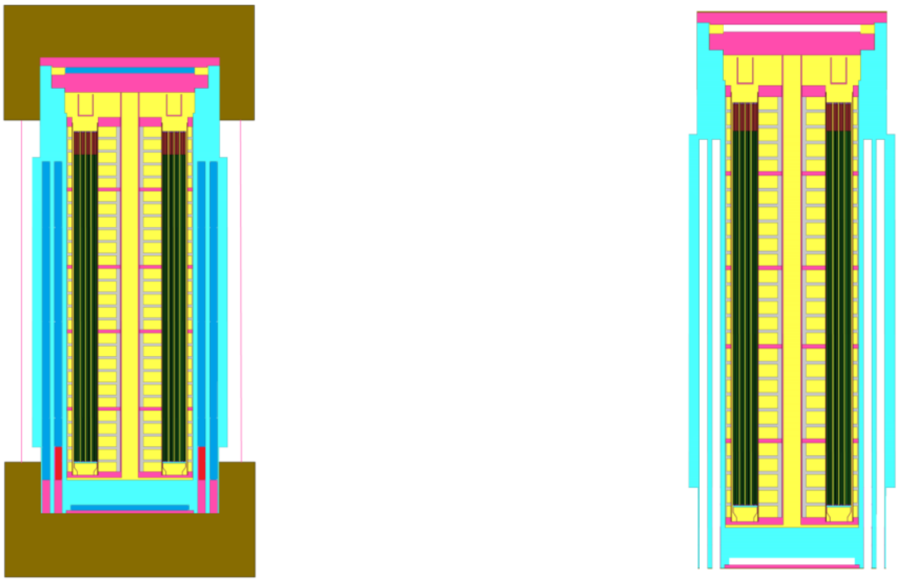
As seen in Figure 4, the nuclear safety criterion in case of an accident with a water distribution change in TUK-153 is not provided. It should be noted that the considered accident scenario may be aggravated by the possible water boiling into the packing because of the SNF residual heat, which make the fuel cladding temperature reach 300 ºC. Therefore, accidents defined above require a separate detailed study, including using «Burn-up-credit» approach. Such studies have been carried out for two levels of «Burn-up-credit» approach: «actinides only» and «actinides + fission products». According to the results, the nuclear safety criterion can be ensured if cask TUK-153 loaded by VVER-1000 SNF with burnup at least 31 GWd/tU for «actinides only» level of «Burn-up-credit» approach, and at least 22 GWd/tU for «actinides + fission products» level of «Burn-up-credit» approach.

Study of radiation safety parameters

Calculational study of radiation safety parameters

Computational models for dose rates calculations are prepared using the functional module MAVRIC [9], which is part of the SCALE 6. Calculations were performed using multigroup cross sections library based on the nuclear data files ENDF/B-VII. Assessment of radiation source is made using control modules TRITON [10] and ORIGEN-ARP [11], which are a part of the SCALE 6 and are designed to simulate the process of nuclear fuel burning in the reactor and to determine the nuclide composition of SNF.

Figure 5 shows the computational models of TUK-153 loaded with 18 VVER-1000 SFA, which takes into account design and material features of biological protection of package and fuel assembly and which was designed to assess the dose rates outside the package protection under normal and accident conditions of transport, including and the worst (in terms of the radiation consequences) design accident (placing the loaded TUK-153 in the in the seat of fire with temperature 800 °C during 0.5 hours, which leads to loss of the neutron shielding of TUK-153). During calculations it is assumed that the surface of the vehicle coincides with the surface of TUK-153 which is a conservative approach. It should also be noted that for the calculations all 18 SFA were assumed identical and characterized by a maximum radiation source.



(a) (b)

Figure 5. Cross sections of the computational model of TUK-153 with SNF for dose rates calculation: under normal conditions – (a); b - under accident conditions – (b)

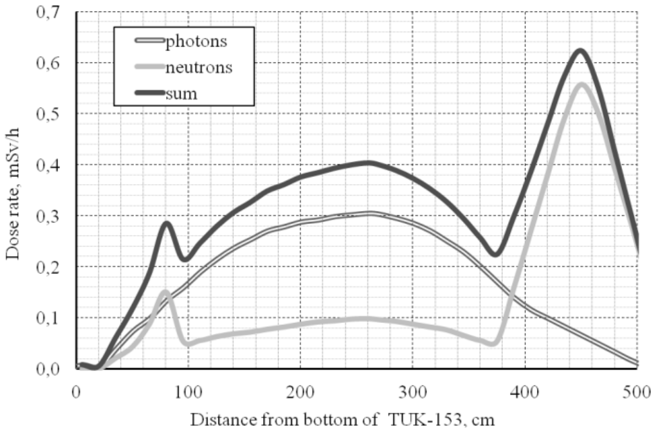
All simplifications of the actual packaging design, such as construction tolerances as well as the absence of some structural elements and freight pins, are conservative and lead to an overestimation of dose rate beyond the TUK-153 protection.

The results of calculations of the dose rate beyond the TUK-153 protection are shown in Figure 6 (a, b, c). Table 1 shows the maximum dose rate values beyond the TUK-153 protection under normal and accident conditions of transport, as well as there is given a dose rates limits according to NP-053-04.

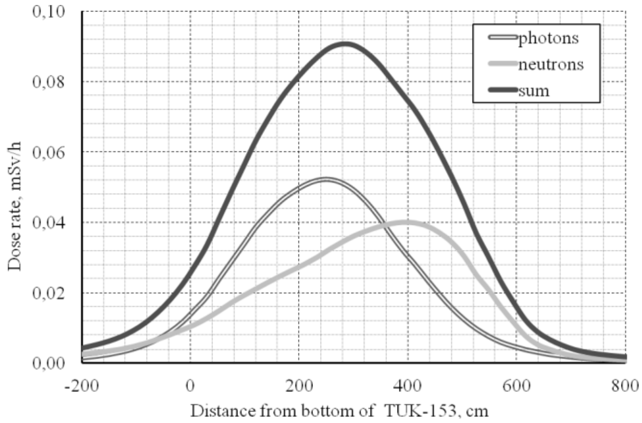
Table 1. The maximum dose rate values beyond the TUK-153 protection under normal and accident conditions of transport

Area of detection	Operating conditions	Dose rate, mSv/h	
		Calculated values	Limits according to NP-053-04
The surface of the TUK-153	Normal	0,625	2
2 meters from the surface of the vehicle		0,091	0,1
1 meter from the	Accident	5,4	10

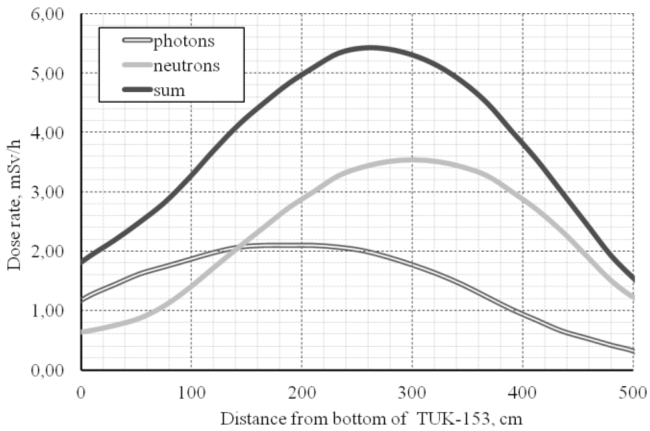
surface of the vehicle			



(a)



(b)



(c)

Figure 6. The distribution of dose rate at the surface of TUK-153 under normal conditions – (a), at 2 m from the surface of TUK-153 under normal conditions – (b), at 1 m from the surface of TUK-153 under accident conditions – (c)

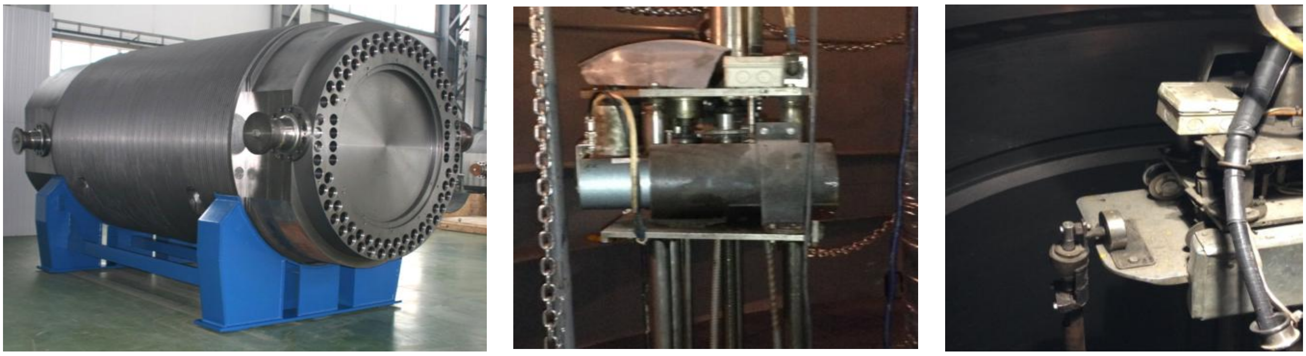
As can be seen from Table 1, the calculated values of dose rate meet the requirements NP-053-04 with a large margin. However, the resulting value of dose rate at 2 meters from the vehicle surface was 0.091 mSv/h, which is only 10% less than the limit value of 0.1 mSv/h according to NP-053-04, and at the same time the contribution to the total dose rate by gamma radiation, as seen in Figure 8, was 57%.

Experimental study of radiation safety parameters

The results presented above are given without the error of radiation transfer calculation. Conservative accounting of this error may exceed the dose rates limits according NP-053-04. With the authors participation there was carried out an experimental study of shielding of TUK-153 biological protection in order to confirm the conservatism of the created computational model.

The basis of the experimental setup which was designed to determine the shielding characteristics of of TUK-153 biological protection from the gamma radiation, is TUK-153 body, which is casted together with the bottom of ductile cast iron. General view of TUK-153 body is shown in

Figure 7 (a) [12]. Furthermore, the experimental setup includes a stand of gamma control, CsI(Tl) scintillation detector and closed radionuclide gamma ray source ^{60}Co, as seen in Figure 7 (b, c).



(a) (b) (c)

Figure 7. General view of TUK-153 body

Stand of gamma control consists of two parallel columns, kinematic device and electric motors, and it's mounted directly to the TUK-153 body. This stand allows to put a source-detector pair at any point of the control zone. ^{60}Co gamma-ray source is mounted in the collimator on the inside column, and the detector is fixed in a lead protective container which is placed on the outside column. The experimental stand is mounted on the TUK-153 body, which is previously placed in the shaft of gamma control (see. Figure 8).

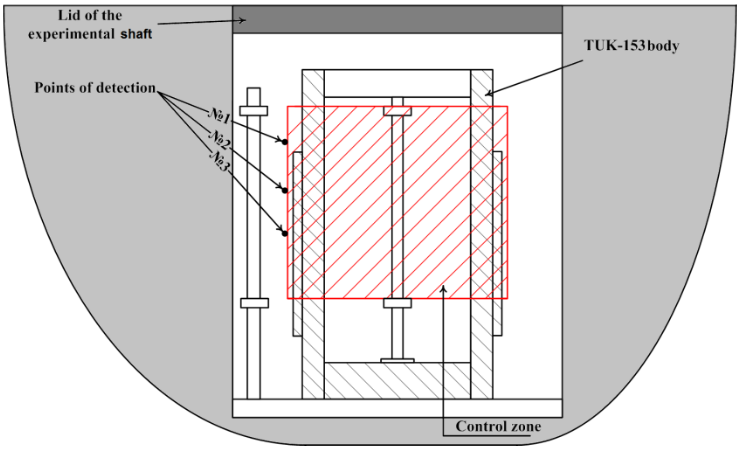


Figure 8. Scheme of experimental stand

The control zone shown in Figure 8 includes all «problem places» on the TUK-153 body, characterized by the smallest value of «optical» thickness of biological protection:

- detector № 1 is located near the chamfer for truck pins;
- detector № 2 is located near the chamfer for installation of TUK-153 to the vehicle;
- detector № 3 is located near the heat-removing ribs.

Computational models of TUK-153 body, the experimental stand, the source, the detector and shaft of gamma control are developed by MAVRIC and they are as close as possible in their geometrical and material parameters to the experimental conditions. General view of the computational model of the experimental setup located in the gamma control shaft is shown in Figure 9. The computational model of detectors №1 - № 3 are shown in Figure 10.

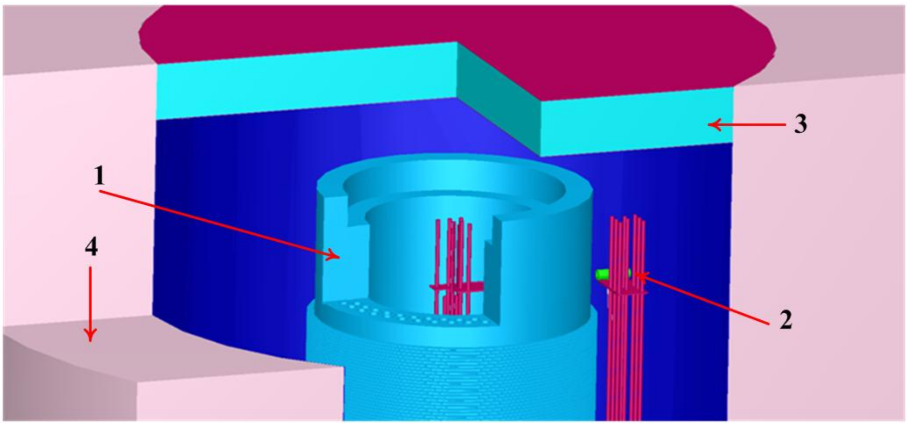


Figure 9. Computational model of the gamma control shaft and experimental setup (1 – TUK-153 body, 2 – stand of gamma control, 3 – lid of the shaft, 4 – gamma control shaft)

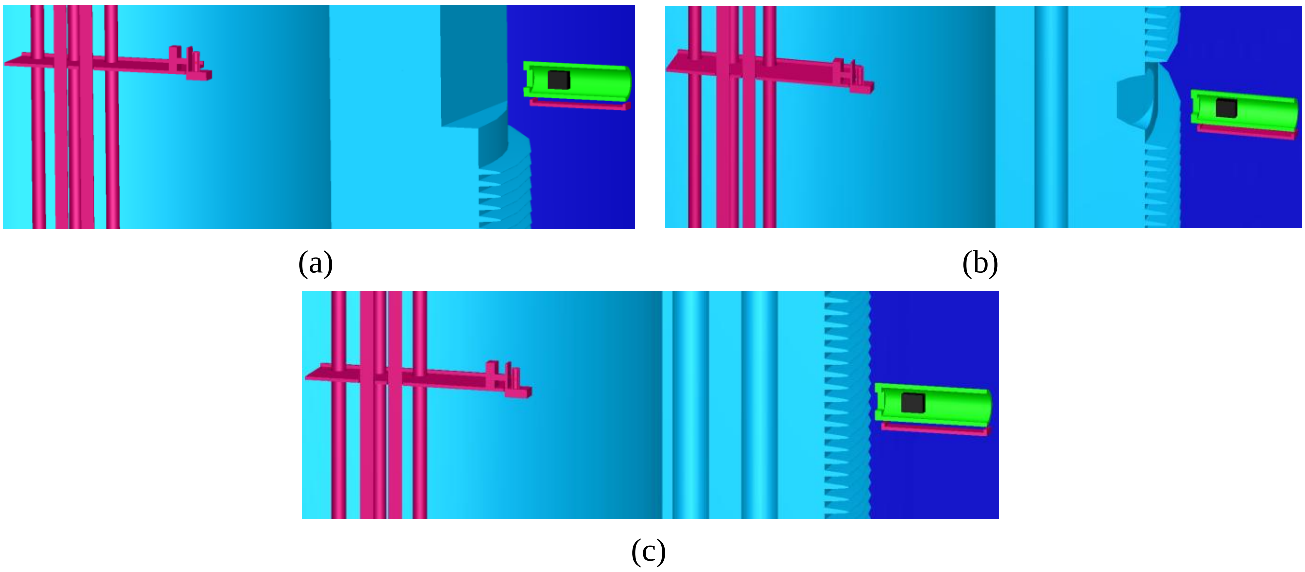


Figure 10. Locations of the detector number 1 – (a), number 2 – (b) and number 3 – (c) in computational model

Table 2. Results of experimental and the calculated gamma dose rates

Detector	№ 1	№ 2	№ 3
Average experimental value of dose rate, μGy/h	6,05	23,32	12,44
Calculated value of dose rate, μGy/h	6,71	27,45	13,91
Deviation, %	+10,9	+17,7	+11,8

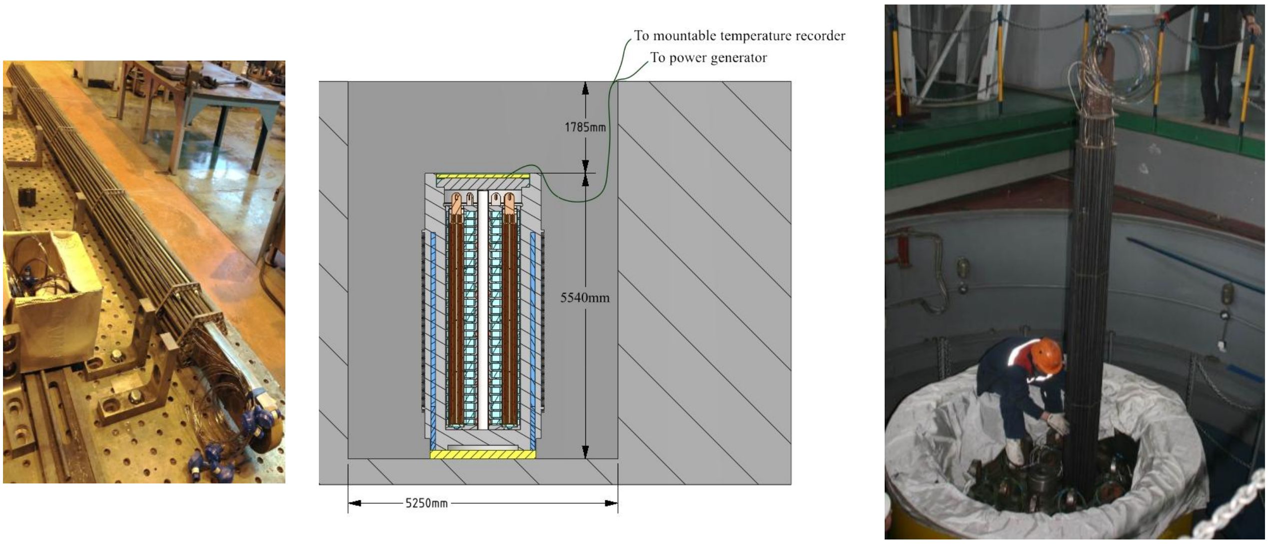
As can be seen from Table 2, the calculated value of dose rate exceeds the experimental value of 10 - 18% depending on the point of detection. This, at first, may be due to the fact that in computational models values of all the material and geometric characteristics of the TUK-153 body were taken conservatively, and secondly it can be caused by multigroup approximation of dependence of the photon cross sections on energy, implemented in SCALE 6.

Nevertheless, the experiment confirmed the conservatism of a created computational model for the calculation of gamma radiation dose rate beyond the protection of TUK-153 using SCALE 6. Completed computational and experimental studies have shown that the deviation of the calculated values of gamma dose rates beyond the TUK-153 body can exceed the experimental data on 10 – 18%.

Study of thermal characteristics

To evaluate the thermal characteristics of TUK-153 loaded VVER-1000 SNF there was performed a computational and experimental study of construction elements temperature of TUK-153, loaded with VVER-1000 SFA imitators.

The experimental part of the study was carried out using the developed by the authors together with the specialists of JSC "Energotex" and manufactured in JSC "Energotex" VVER-1000 SFA imitators and experimental setup. The design of VVER-1000 SFA imitator is a bundle of 84 metal rods and six tubular electric heaters (TEH) arranged in a triangular lattice, and the dimensions of SFA imitator and the working length of TEH are identical to overall dimensions of the SFA and the active part of the fuel elements, respectively. Figure 11 (a) shows an imitator with connected thermocouples. The experimental setup is TUK-153, loaded with 18 SFA imitators with established thermocouples and located on the thermally insulated support in the shaft. Scheme of the experimental setup is shown in Figure 11 (b), the SNF imitator installation process in TUK-153 is shown in Figure 11 (c).



(a) (b) (c)

Figure 11. VVER-1000 SFA imitator – (a), scheme of experimental setup – (b) and SNF imitator installation process – (c)

When creating computational models the following conservative assumptions and approximations were used:

- along the length of SFA imitators there are choosen sections with the same thermal characteristics in the radial direction can be assumed to be constant;

- neutron shielding layer in the TUK-153 body is replaced by a homogenized layer with equivalent thermo-physical characteristics;
- heat exchange between the outer surface of the TUK-153 and environment is carried out by convection and radiation;
- on the outer surface of the TUK-153 shell in the shock absorbers area adiabatic boundary conditions were assumed.

Because of the complexity the TUK-153 geometry, a computational study using ANSYS without assumptions and approximations would require a lot of processing and time. Therefore, for the purposes of this study there was performed a replacing of heterogeneous structure with an equivalent homogeneous structure with average thermal characteristics. Solution of the problem in this approach allows to estimate the maximum temperature of homogenized zone and to determine the temperature distribution on the boundary. The subsequent solution of the problem in reverse substitution (from the equivalent section to the original one) allows to determine the temperature field of the all model. According to this principle there were developed some finite element models (see Figure 12). Using the above models the effective values of thermal conductivity coefficients of each of the sites were obtained. As a result, the geometry of the equivalent model of TUK-153 can be represented by two-dimensional axial-symmetric model (see Figure 13).

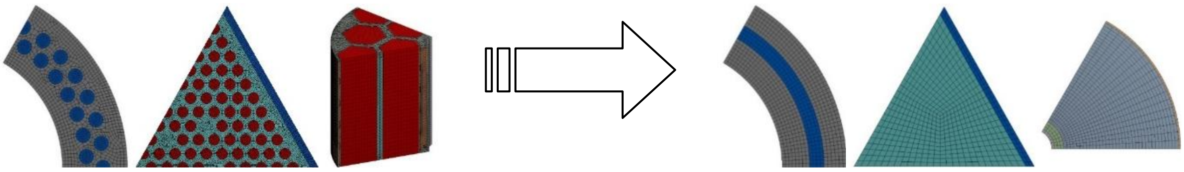


Figure 12. Computational finite element models to calculation of thermal conductivity coefficients

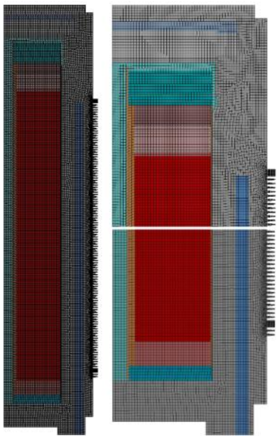


Figure 13. Two-dimensional axial-symmetric computational model of TUK-153

The experimental time dependence of the TUK-153 surface temperature, together with the relevant calculations results, are shown in Figure 14.

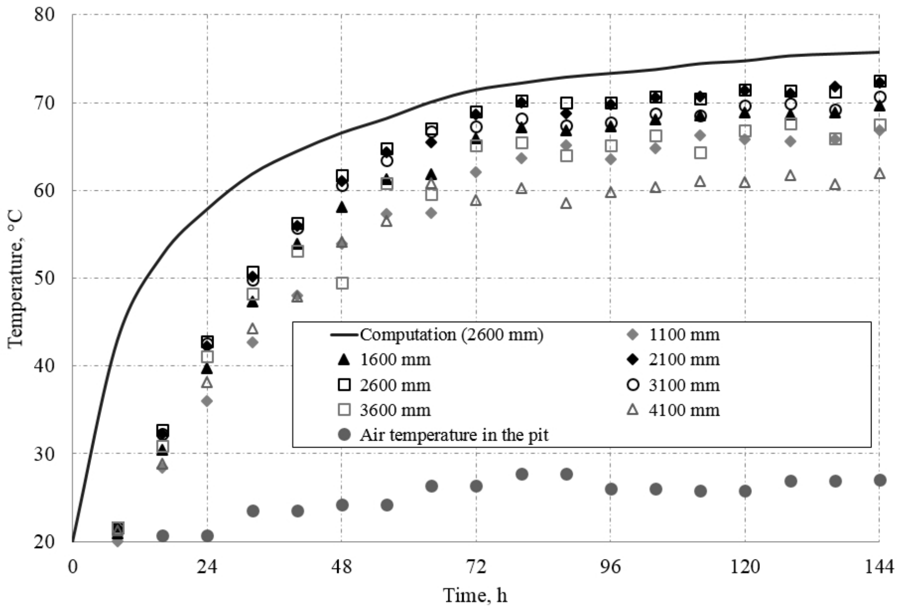


Figure 14. Computational and experimental values of TUK-153 (with SNF imitators) surface temperature

According to Figure 16 the TUK-153 surface temperature is approximately 70 ºC when stationary thermal regime was achieved. Completed computational-experimental researches of temperature modes of TUK-153 structural elements show conservatism of used approaches in creating the computational model using ANSYS. The calculated values of the surface temperatures of TUK-153 loaded with SFA imitators can conservatively exceed the experimental values by 10%.

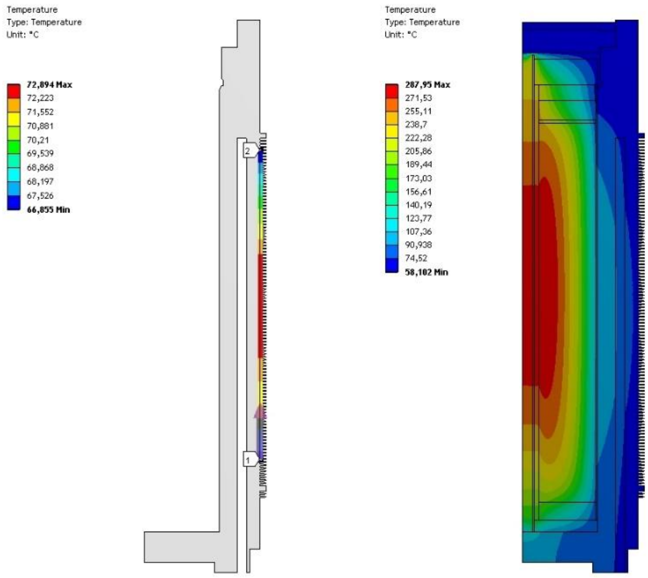


Figure 15. The temperature distribution in the TUK-153 with SNF computational model on the side surface (a) and in the most intense cross-section (b)

Using a conservative approaches described above, there was developed a computational model of TUK-153 with SFA characterized by maximum residual heat during most "strained" conditions of transport (by environment temperature and insolation). The calculated value of the maximum surface temperature of TUK-153 loaded with SNF was 73 ºC (see Figure 15), which is below the required value (85 ° C) according to NP-053-04 and SSR-6.

Conclusions

The independent evaluations of safety parameters is an integral and necessary part of safety assessment of nuclear facilities and activities in the field of nuclear energy. Such evaluations improve the quality of safety assessment and finally leads to increase of safety.

The quality of independent evaluations of safety parameters carried out by the technical support organizations have to be confirmed by computational or/and experimental studies of evaluated parameters. SEC NRS best practices of such confirmations presented in this paper.

The results of describe above computational and experimental studies of safety of VVER-1000 SNF transport in TUK-153 confirm the conservatism of using in SEC NRS approaches during preparation of computational models for independent evaluation of safety parameters during SNF transport.

Acknowledgments

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