EXPIRIENCE OF USING THE SCALE SOFTWARE PACKAGE FOR SUBSTANTIATION OF NUCLEAR SAFETY OF DRY STORAGE OF SPENT NUCLEAR FUEL

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SCALE software package has been used to model a ventilated concrete container for the dry storage of spent nuclear fuel. The methodology of calculations, which isused to substantiate the nuclear safety of dry storage, is described.

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INTRODUCTION

Handling of spent nuclear fuel (SNF) is one of the main tasks to be solved by nuclear industry of Ukraine. Nuclear power plantssafe operation becomes more problematic because of the limited volumeof SNF export in Russia for storage and own reactor storage poolsare overflowed.

Nuclear power plants (NPP) in Ukraine lack capacity pools for spent fuel storage. For this reason in 2001 the State Committee of Nuclear Regulation of Ukraine granted a license to operate the dry storage of spent nuclear fuel (DSSNF) facilityat the Zaporizhska NPP, which is the biggest in Europe with the total electrical capacity of 6000 MW.

In addition, the SNF in the most countries with advanced nuclear industryis considered as a nuclear waste, which is stored without processing. However, the SNF contains a lot of fissile materials such as unburned ^{235}U and accumulated ^{239}Pu, which can be reusedlater. Therefore, building of DSSNF is the proper and considered decision.

1. ADVANTAGES OF DRY SPENT FUEL STORAGE

During SNF storing in a reactor pool their heat power and radioactivity are decreased considerably. The maximum heat power of a singlespent fuel assembly (SFA) is 0.99 kW. The SFA can be stored safely at a NPP site in special containers, which are providing effective heat removal from SFA and sufficient biological shielding of the NPP staff and the environment from radiation.

Based on a comparative analysis of the possible ways of storing SFA at Zaporizhska NPP was chosen a temporary storage system, which allows to store SFA in ventilated storage casks (VSC) during 50 years. The VSC are installed on a concrete deck, which is located on the NPP site. This system is a variant of the temporary storage system, which is used at US NPPs, and is licensed US regulatory authorities – Nuclear Regulatory Commission (NRC). Such systems include container system of SNF dry storage VSC-17 and VSC-24 in which are stored SFA from nuclear reactors of PWR type. Despite the relatively short period of storage of the SFA in VSC-17 and VSC-24, there is already a considerable amount of scientific and technical developments and operating results, which confirm that the long-term storage of SNF in such systems is safe. The first VSC was commissioned in May 1993 at "Palisades" NPP (USA). Canada, Germany, Switzerland, and United Kingdom have experience to store aspent fuel in the dry storage.

The main advantages of the SFA dry storage technology are:

- Safety, which is the main principle for the design and operation of the system. Protection of the population, NPP staff and environment are fundamental requirements.
- All operations with fuel, including loading the storage container are performing inside the reactor building tominimize the potential possibility of the radioactive pollution. In addition, for loading containers with SFA is used equipment, which the NPP already has.
- During storing of SFA in a storage pool for 5…10 years (depending on the type and fuel burnup) decay heat of each SFA is reduced to 1 kW or less, that one allows to remove heat with natural circulation of air around the basket with 24 spent fuel assemblies.
- Additional safety is achieved withstoringonly the hermetic SFA that had no release of radioactivity beyond fuel elements above prescribed limits during the operation in the active zone of the reactor and keeping in storage pool.
- All SNFDS containers are stored at a NPP site that reduces transportation costs and additional guarding.

The main quantitative design criteria of nuclear safety for the system of spent fuel handling is the effective multiplication neutrons factor value $K_{eff} < 0.95$ which is the same for normal operating conditions and design accidents. In order to load each container of spent nuclear fuel the K_{eff} calculations are carried out. Calculations are performed using the SCALE software package.

2. DESCRIPTION OF THE SCALE SOFTWARE PACKAGE

SCALE is a modular system of standardized computer analysis for licensing, developed by Oak Ridge National Laboratory (USA) for the US Nuclear Regulatory Commission (US NRC) to perform criticality, radiation, heat transfer and burn up analysis [1, 2]. In this code, neutron transport equation issolved by Monte Carlo method. All calculations in substantiation of nuclear safety of container loadings at Zaporizhska NPP are performed using the calculation sequence CSAS6, which is included in SCALE software package. This calculation sequence is used in the analysis of criticality and consists of sequential call executable modules CSAS26, BONAMI, NITAWL, and KENO-VI. The basic module for the calculation of criticality is code KENO-VI, for which BONAMI and NITAWL modules prepare multigroup cross-sections library for the compositions defined in the calculation model.

The calculation sequence CSAS6 applicability for fuel of WWER reactors was substantiated in article [3], in which was shown that 44-group neutron cross-section library 44 GROUPNDF5 gives some overestimation of the calculated values of the neutron multiplication factorin compare with experimental data and was recommended for use in the criticality calculations.

An analysis of the applicability of the package SCALE versions 5.0 and 5.1 for WWER reactor nuclear fuel was madein [4]. The analysis wasbased on 60 critical experiments. It was shown that the calculation sequence CSAS6 for all type neutron cross-section libraries systematically underestimates the value of the neutron multiplication factor, therefore to get the final K_{eff} value code systematic error and error variance must be considered. Systematic error and variance for version 5.0 and 5.1 package SCALE were defined in [4] for all types of libraries.

Currently, to obtain a conservative estimate of the value of K_{eff} the following formula is used:

$$k_{95} = k_{\mathrm{KENO}} + 2 \cdot \sigma_{\mathrm{KENO}} \,, \qquad (1)$$

where k_{KENO} – K_{eff} value is calculated by code KENO-VI; σ_{KENO} – standard deviation.

Formulas (1) gives the final value of k_{95}, which cannot be exceeded at 95% of cases.

The results of the K_{eff} calculations are processed in according with formula and the k_{95} final value is compared with the limit value 0.95. In articles [3, 4] the applicability of SCALE package for the criticality calculations for the nuclear fuel of WWER-1000 reactors are substantiated.

3. DESCRIPTION OF THE FUEL ASSEMBLY AND CONCRETE CONTAINER

The geometrical dimensions of the fuel assembly TVSA for the WWER-1000 reactors are shown in Fig. 1 and are consistent with the data from documents [5, 6].

For intermediate storage of spent nuclear fuel assemblies ventilated concrete containers are used (Fig. 2).

Twenty-four SFA are stored in the hexagonal pipe covers (guide channels) that are located in the cylindrical multiplace hermetic storage basket (Fig. 3). The basket is made of carbon steel. Multiplace basket is also a radiator for removing SFA residual heat into ventilated concrete container volume. Helium inside the basket forms and maintains dry inert heat transfer medium during the storage period.

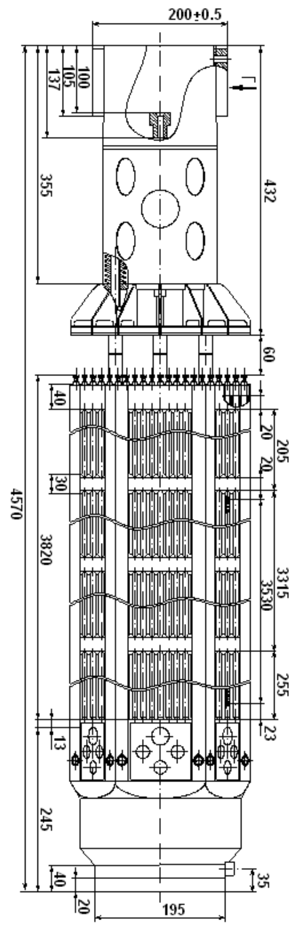

Fig. 1. Design of the fuel assembly type TVSA for the WWER-1000 reactor

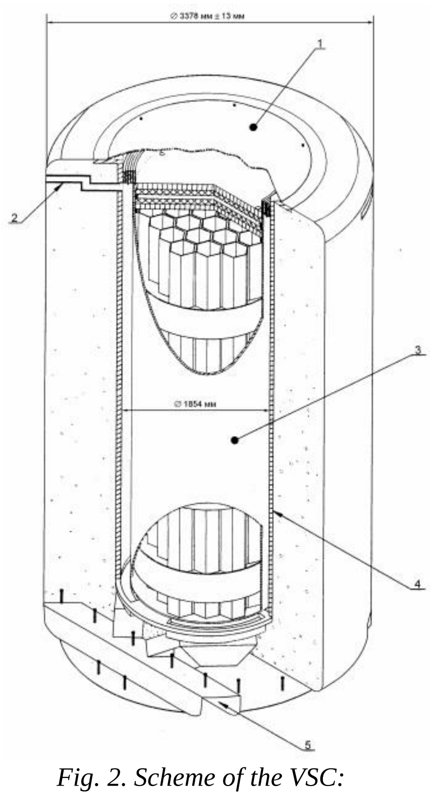

Fig. 2. Scheme of the VSC:
1 – container cover; 2 – air outlet;
3 – hermetic basket storage; 4 – coating;
5 – air input and guide for transportation

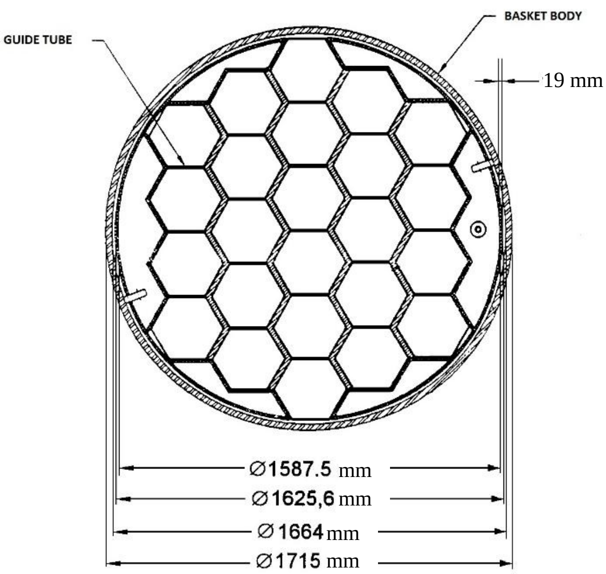

Fig. 3. Section of the hermetic multiplace basket for storage of the spent fuel assemblies

The basket transportation within the reactor unit is carried out in a special shipping container that performs the following functions: biological shielding of staff who are involved in the transportation; basket protection against a mechanical damage during the transportation; protection of the basket outer surface from pollutionsource of which is located in the storage poolwater.

The hermetic multiplace basket is placed in a ventilated concrete container, which performs the following functions: removal of residual heat from the basket; protection the basket from climatic, mechanical, and thermal impacts; biological shielding of thestaff, which is serving DSSNF; providing steady vertical position of the basket with the SFA during transportation and storage.

Downloaded concrete container is placed on the special storage area that is located on the guarded NPP site. This area has own guarded boundary and performs the following functions: prevention of free access of unauthorized persons to the zone of radiative impact of DSSNF; stability of the transport and lifting tools, which are used in the transport and technological operations with containers; drainage of rainwaters from DSSNF. In general, DSSNF system is passive and after the concrete containers installation on the concrete deckfor storing does not require significant maintenance.

4. DESCRIPTION OF THE FUEL ASSEMBLY AND CONTAINER CALCULATION MODEL

The geometrical dimensions and material composition of structural elements of the SFA type TVSA of the WWER-1000 reactor, which are used in the development of the model and the calculations are shown in Tabl. 1.

As an additional absorber the burned two – component control rods (CR) are used in the container loading. The CR dimensions and material composition are given in Tabl. 2 [7]. The CR top part consists of boron carbide (B_4C) and the bottom part with length of 30 cm consists of titanate dysprosium ($Dy_2O_3TiO_2$). The bottom part of the CR for the model simplification is substituted with the tip from boron carbide with the same absorption ability. The absorber ^{10}B during the operation CR in the core is burned and in the model conservatively the burnup is taking account by reducing the B_4C density of 25%. Thus, the B_4C density at the upper part of the CR is equal 1.275 g/cm^3, and at the bottom part is 0.161 g/cm^3.

Table 1

Description of the fuel assembly type TVSA for reactor WWER-1000

Name of the element	Value
Fuel rod	
Height of the fuel rod	3530 mm
Overall length of the fuel element	3837 mm
Fuel pellet	UO_2, UO_2+Gd_2O_3
Outer radius	3.785 mm
Center hole diameter	1.4, 1.5 mm
Fuel cladding	
Inner radius	3.865 mm
Outer radius	4.55 mm
Material	alloy Э110
Instrumental tube	
Inner radius	5.5 mm
Outer radius	6.5 mm
Material	alloy Э635
Guide tube	
Inner radius	5.45 mm
Outer radius	6.3 mm
Material	alloy Э635
Stiffness angle	
Number	6
The heated part length of the bunch	3530 mm
Width	52 mm
Thickness	0.65 mm
Material	alloy Э635
Fuel assembly	
Type	hexagonal lattice
Lattice pitch	12.75 mm
Number of elements in a lattice	331
Number of fuel rods	312
Number of guide tubes	18
Fuel weight (UO_2), kg*	491.4 ± 5 494.5 ± 5
Enrichment, %	For calculation 4.4%
**Value of the top and low fuel hole diameter*	

The TVSA fuel assembly model based on the original data with the burned CR and discrete burnable absorber (DBA) rods for the SCALE software package has been developed. The head and bottom nozzle of the fuel assembly in this model has not been modeled, only part of assembly containing the fuel rods has been modeled. The TVSA calculation model diagram in the vertical cross-section view is shown in Fig. 4.

Table 2

Description of absorbing elements

Absorbing element	
Absorber element	B_4C
Outer radius	3.5 mm
Density	1.275 g/cm^3
Composition, at.%	^{10}B (19.8%), ^{11}B (80.2%)
Height of the absorber	3710 mm
Height of the control rod	4240 mm
Cladding of the control rod	
Inner radius	3.5 mm
Outer radius	4.1 mm
Density	7.8 g/cm^3
Composition, wt.%	Steel (Fe-69.5, Cr-18, Ni-11, Mn-1.5%)

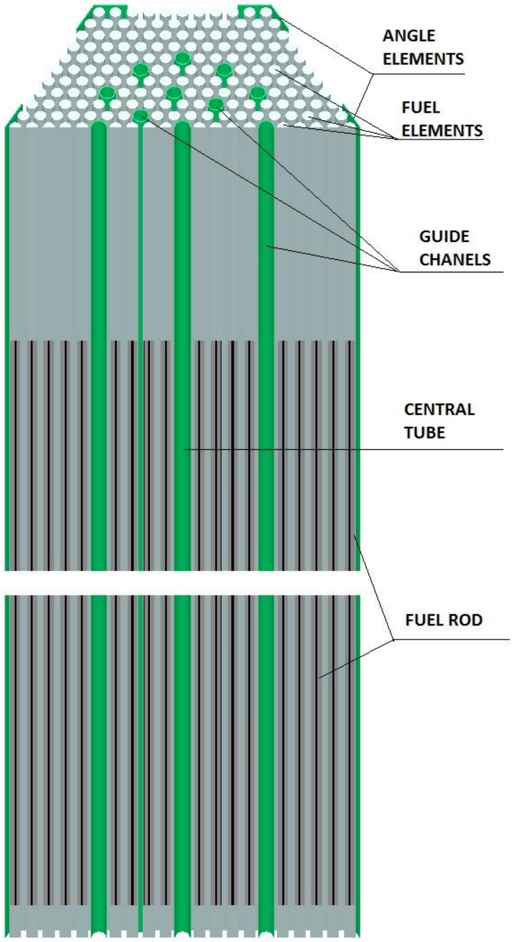

Fig. 4. Scheme of calculation model of the fuel assembly type TVSA

As one can see on the Fig. 4 the assembly calculation model is constructed from a set of fuel rods. The fuel rods length is equal 3837 mm; the hexagonal lattice pitch is 12.75 mm and the assembly full size is equal to 234.8 mm.

The fuel rod upper part, spring and the bottom part are modeled in the calculation model as solid zirconium cylinders. The fuel rod and its calculation model are shown in Fig. 5.

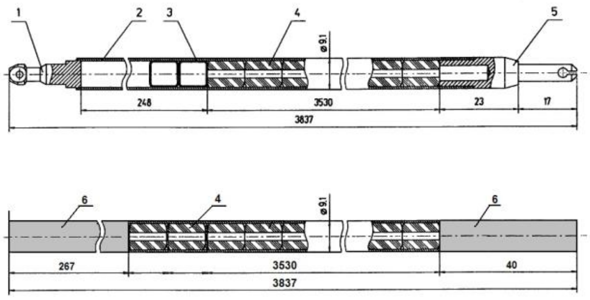

Fig. 5. WWER-1000 reactor fuel rod:
1 – top nozzle; 2 – cladding; 3 – spring; 4 – fuel pellet; 5 – bottom nozzle; 6 – zirconium solid cylinder

Geometrical and material parameters of the multiplace hermetic basket and concrete container, which are used in the model, are presented in Tabl. 3 [7].

Table 3

Parameters of ventilated concrete container

Element	Parameter	Value
Concrete cladding	Outer diameter Inner diameter Thickness Height Material Density	3378 mm 2007 mm 686 mm 5809 mm Concrete 2.34 g/cm^3
Coating	Outer diameter Inner diameter Thickness Height Material Density	2007 mm 1854 mm 73 mm 5200 mm 10ХНСД 7.9 g/cm^3
Container lid	Diameter Thickness Material	2184,4 mm 57 mm 10ХНСД
Hex block tubes	Outer diameter of the power ring Inner diameter of the power ring Block height Thickness of the power belt Material	1625.6 mm 1587.5 mm 4320 mm 19 mm 10ХНСД
Hexagon guide tubes	Distance between the outer sides Distance between the inner sides Full length Fuel zone length Backend zone length of the fuel assemblies Heads zone length of the fuel assemblies Number of tubes Material	259 mm 249.6 mm 4320 mm 3530 mm 425 mm 270 mm 24 10ХНСД
Shell ring of the basket storage	Outer diameter Inner diameter Thickness Length Material	1715 mm 1664 mm 25.4 mm 4973 mm 10ХНСД
Power lid	Diameter Thickness Material	1651 mm 76.2 mm 10ХНСД
Protective lid	Diameter Thickness Material	1657.4 mm 241.3 mm 10ХНСД

To ensure the in dependence of the K_{eff} values from the boundary conditionschoice, the hermetic basket is surrounded with additional 10 cm water layer on the sides and 22.7 cm at the top. This additional layer of water gives in the test calculations the same K_{eff} value, both boundary conditions vacuum and mirror reflection.

The full-scale model of multiplace basket and concrete container was developed for the normal operation condition calculations (Fig. 6). The concrete container and hermetic basket filled with helium which density is $1.78 \cdot 10^{-4}$ g/cm^3. The mirror boundary conditions on the concrete container outside are set up.

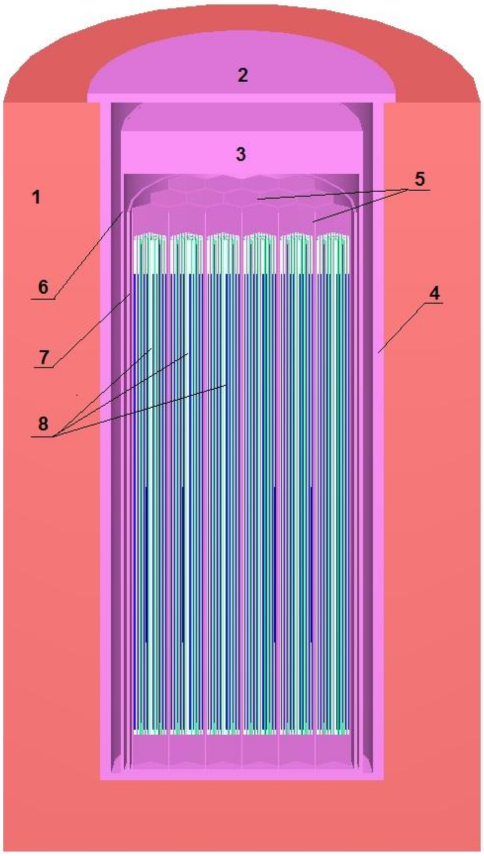

Fig. 6. The concrete container model with the hermetic basket: 1 – concrete container; 2 – lid; 3 – power and protectivelids; 4 – coating; 5 – block of the hexagon guide tubes; 6 – the hermetic basket shell; 7 – power ring; 8 – SFA

5. CALCULATION RESULTS

The main requirement for nuclear safety is not exceeding K_{eff} of value 0.95 for the normal operation conditions and design-basis accidents.

All calculations for the nuclear safety substantiation of the DSSNF container loadings are performed with taking account of the fuel burn up. The isotopes, amount of which can be defined with sufficient accuracy and lifetime of which exceeds time period that is covered in nuclear safety validation are taken into account. Such fissile isotopes as ^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu, and ^{241}Pu [8] are considered. The minor actinides and other neutron absorbers are not considered in the burned fuel isotope content, which leads to additional conservatism.

According to the height of fuel assembly the burn up distribution has significant non-uniformity. The upper and lower ends of the fuel assembly deplete less than the central part because of axial neutron leakage. Therefore, following conservative approachin the nuclear safety analysis, the burn up of fuel assembly is equal to the average burn up of assembly end parts.

In the calculations is assumed that the container is filled by water with density of 1 g/cm^3.

A container loading pattern is presented in Fig. 7. For developing of loading patterns the following approach is used – the least burned fuel assemblies are located mainly in the center of container, and the most burned fuel assemblies are located in the peripheral rows. CRs are installed in the fuel assemblies located in the center and DBAs are installed in the peripheral rows. Such approach allows to form loading with a minimal number of CR and with aminimal K_{eff} for normal operation conditions and to satisfy the $K_{eff} < 0.95$ requirement for all accidents.

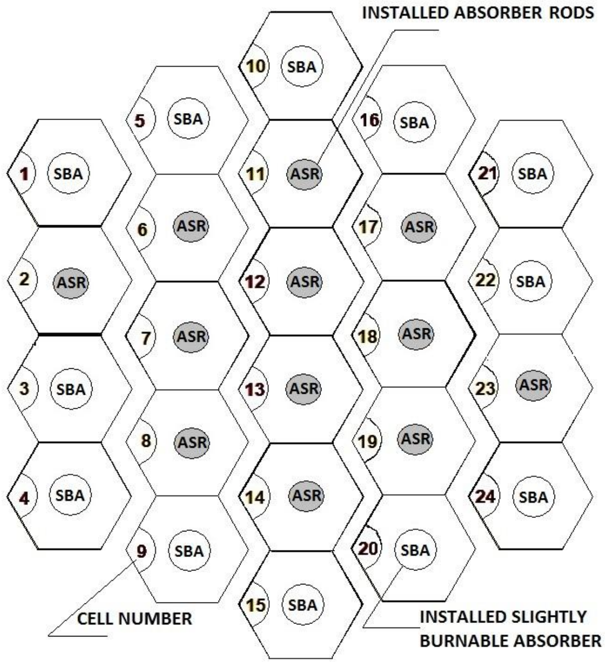

Fig. 7. A container loading pattern

The nuclear safety substantiation of the SNFDS container loadings consists of the next steps:

– loading pattern determination with a minimal K_{eff} value.

Accidents analysis:

– erroneous loading of a DBA rod instead of a CR;

– erroneous loading of a fresh fuel assembly with maximum enrichment instead of spent fuel assembly;

– axial shift of all absorbing rods at 10 cm that can be considered as modeling of container turnover;

– displacement all SFA to the center of multiplace basket within the hexagonal tubes;

– water levellowering inside the multiplace basket is modeling of water drainfrom a multiplace basket for subsequent drying and filling with helium;

– calculation of the final K_{eff} values isperformed in according with the formula.

At the moment safety critical calculations for the 80 containers loadings are performed in the Kharkov Institute of Physics and Technology with using the SCALE software package.

CONCLUSIONS

The nuclear safety calculations of DSSNF container loadings are performed in accordance with a technical specification and based on SNF data, which are provided by NPP. The container loading pattern is chosenso that $K_{eff} < 0.95$ requirement for normal operating conditions and accidents is meet. Technological tolerances, systematic and calculation statistical errors are taking into account too. SFA are loaded into containers with the necessary number of CRs and DBAs.

Experience of performing the nuclear safety calculations shows that the calculation model is excessively conservative. First, it concerns the SFA isotopic composition, which includes only the main fissile isotopes [9], and the SFA depletion is taken as the average value between the upper and lower parts of the fuel assembly, i. e. a burnup profile is not taken into account in the model. Secondly, in the world at the legislative level by more flexible approaches to the substantiation of nuclear safety are approved. For example, if limit value of K_{eff} for normal operating conditions is equal 0.95, then the K_{eff} limit value is 0.98 for accidents and for some type of accidents K_{eff} must not exceed 1.0 [10]. This decrease in margin for the value of the estimated value of K_{eff} to the limiting value 0.98 (or 1.0) is associated with an increase in the accuracy of codes used for the substantiation of nuclear safety, and the use of more accurate neutron cross-section libraries. Ukrainian regulations (for example [11]) need to be revised and updated to international level requirements in the field of nuclear safety. This will reduce the excessive conservatism and reduce the cost of SNF storage.

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ОПЫТ ИСПОЛЬЗОВАНИЯ ПАКЕТА ПРОГРАММ SCALE ДЛЯ ОБОСНОВАНИЯ ЯДЕРНОЙ БЕЗОПАСНОСТИ СУХИХ ХРАНИЛИЩ ОТРАБОТАВШЕГО ЯДЕРНОГО ТОПЛИВА

С.В. Черницкий, С.А. Солдатов

Представлена модель вентилируемого бетонного контейнера для сухого хранения отработавшего ядерного топлива, разработанная для пакета программ SCALE. Описана методология расчетов, которая применяется для обоснования ядерной безопасности сухого хранения.

ДОСВІД ВИКОРИСТАННЯ ПАКЕТА ПРОГРАМ SCALE ДЛЯ ОБҐРУНТУВАННЯ ЯДЕРНОЇ БЕЗПЕКИ СУХИХ СХОВИЩ ВІДПРАЦЬОВАНОГО ЯДЕРНОГО ПАЛИВА

С.В. Черніцький, С.А. Солдатов

Представлена модель вентильованого бетонного контейнера для сухого зберігання відпрацьованого ядерного палива, що розроблена для пакета програм SCALE. Описано методологію розрахунків, яка застосовується для обґрунтування ядерної безпеки сухого зберігання.