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Criticality Safety Analysis of WWER-1000 Spent Nuclear Fuel Storage

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Abstract. Nuclear safety of spent nuclear fuel management is ensured by implementation of the basic safety functions: providing subcriticality, residual heat removal and retention of radioactive products within the physical barriers. To ensure subcriticality during both normal operation and design basis accidents the effective multiplication factor of neutrons K_{eff} must be lower than 0.95. An evaluation of criticality of spent fuel facilities have been made by the modular code system SCALE. The basic calculations are performed with version 6.1 and are validated with version 6.0 of the code system. Spent fuel assemblies type TVSA are modeled as they are representative for WWER-1000 nuclear fuel and cover the characteristics of the earlier modifications of the fuel assemblies. The modeling of the spent fuel containers and equipment is in accordance with actual geometric dimensions and material composition. In all performed calculations, the results demonstrate that the criticality safety criteria are achieved and the effective multiplication factor K_{eff} is lower than the regulatory requirements.

Keywords: sub-criticality, spent fuel facilities, criticality safety analysis

1 Introduction

Nuclear safety of spent nuclear fuel management is ensured by implementation of the basic safety functions: providing subcriticality, residual heat removal and retention of radioactive products within the physical barriers. To ensure subcriticality during both normal operation and design basis accidents the effective multiplication factor of neutrons K_{eff} must be lower than 0.95. An evaluation of criticality of spent fuel facilities have been made by the modular code system SCALE. The basic calculations are performed by version 6.1 and are validated by version 6.0 of the code system. Spent fuel assemblies type TVSA are modeled as they are representative for WWER-1000 nuclear fuel and cover the characteristics of the earlier modifications of the fuel assemblies. The modeling of the spent fuel containers and equipment is in accordance with actual geometric dimensions and material composition. In all performed calculations, the results demonstrate that the criticality safety criteria are achieved and the effective multiplication factor K_{eff} is lower than the regulatory requirements. The calculations are performed with SCALE 6.1 code version by ENPRO Consult [20] and validated by INRNE-BAS with SCALE 6.0 code version [4].

2 Criticallity Safety Calculation

The calculations are performed with the modular code system SCALE – Standardized Computer Analyses for Licensing Evaluation [6-8]. The modular code system SCALE is widely accepted tool for spent nuclear fuel analysis. The new version SCALE6 was applied for criticality safety analysis of WWER-1000 spent nuclear fuel transport and storage facilities. SCALE is fully validated for WWER type reactors with results from experimental data and comparative calculations [9-17].

The standardized automated procedures process SCALE cross sections using the Bondarenko method (via BONAMI) and either the Nordheim Integral Method (via NITAWL) or collapsing of pointwise continuous energy cross sections using a problem dependent pointwise continuous flux (via WORKER, CENTRM, and PMC) to provide a resonance-corrected cross-section library based on the physical characteristics of the problem being analyzed. This cross-section library can also be utilized by KENO-VI, another three-dimensional (3-D) multigroup Monte Carlo criticality program, or XSDRNPM, a one-dimensional (1-D) discrete-ordinates code for transport analysis. TRITON couples NEWT with ORIGEN-S to perform two-dimensional depletion calculations for prediction of isotopic concentrations, source terms, and decay heat as a result of time-varying neutron fluxes calculated in a two-dimensional deterministic approach.

SCALE 6.0/6.1 includes the evaluated nuclear data library ENDF/B-VI.8 and ENDF/B-VII.0 containing pointwise continuous energy data. A 238-group based ENDF/B-VII library is used for criticality analyses (including WWER-1000 spent fuel) in the presented calculations.

Criticality Safety Analysis Sequence with KENO-VI (CSAS6) was developed to provide automated cross-section processing for KENO-VI in the SCALE system. These cross sections are then used in to determine the effective neutron multiplication factor K_{eff}. The codes and their functions are given below.

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BONAMI performs resonance self-shielding calculations for nuclides that have Bondarenko data associated with their cross sections;

NITAWL applies a Nordheim resonance self-shielding correction to nuclides having resonance parameters;

WORKER creates an AMPX working format library from a master format library;

CENTRM using the pointwise continuous cross-section library and a cell description creates a pointwise continuous flux spectrum;

PMC using the pointwise continuous flux spectrum created in CENTRM, collapses pointwise continuous cross sections to a set of multigroup cross sections;

KENO VI criticality program used to calculate the K_{eff} features a complex geometry package;

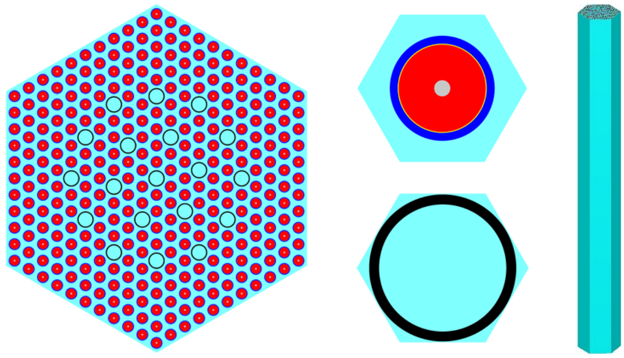
NEWT an arbitrary-geometry, discrete ordinates neutron transport solver that was used for cross-section weighting and generation of few-group constants for lattice physics calculations using a problem-specific multigroup library;

COUPLE automatically couples problem-dependent cross-section constants and flux weighting factors into libraries used by ORIGEN-S for performing calculations of isotopic depletion and generation and their associated radiation sources and decay heat

ORIGEN-S system module to calculate fuel depletion, actinide transmutation, fission product buildup and decay, and associated radiation source terms, the COUPLE/ORIGEN calculations are repeated for each mixture being depleted, as specified in input, using mixture-specific cross-section data and fluxes.

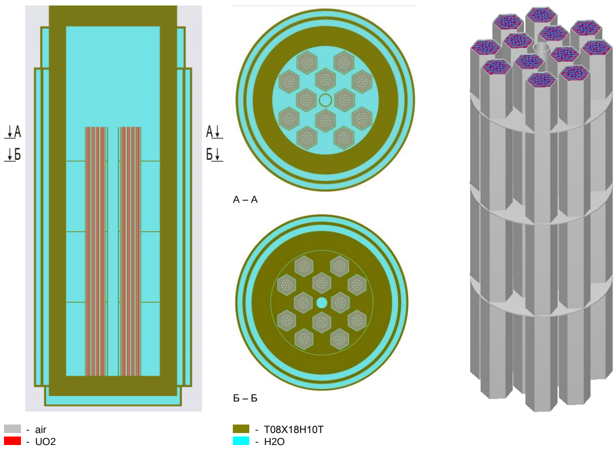
3 Scale Fuel Modeling

In this paper are presented result of calculations performed for: 37/3 basket with 12 spent fuel assemblies WWER-1000 [18,19], TK-13/3 cask [18,19] for transport of 12 spent fuel assemblies from reactor blocks 5 and 6 [18,19], transport site with 4 TK-13/3 casks. In Figures 1–3 below are shown the fuel assembly, basket, transport cask and spent fuel storage, modeling by SCALE 6.1. The criticality safety criterion is based on the effective multiplication factor K_{eff} as the regulatory requirements is to be lower than 0.95.



(a) (b) (c)

Figure 1. (a) – TVSA fuel assembly; (b) – fuel cell and guide tube; (c) – FA in axial.



(a) (b) (c)

Figure 2. (a) – Spent fuel cask in axial; (b) – Cask with water and air; (c) – Basket axial.

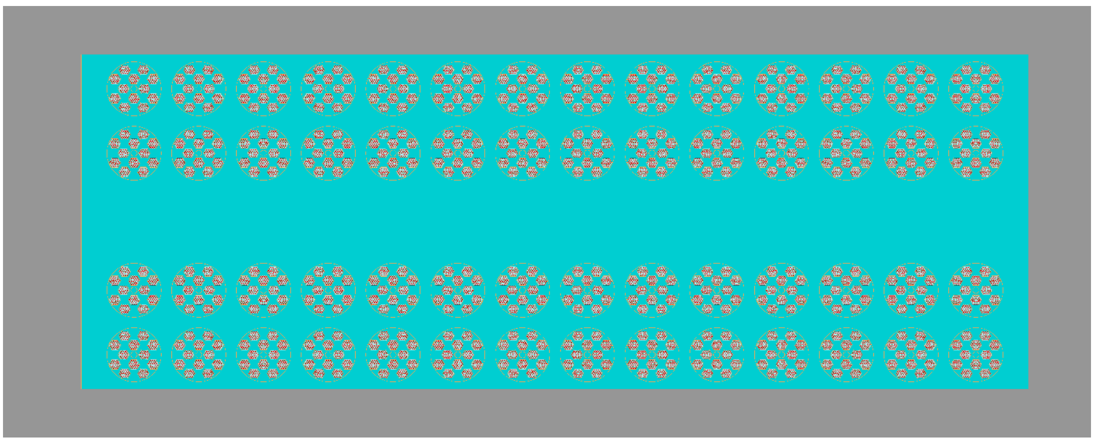


Figure 3. Spent fuel pool model with 56 baskets.

During the analyses following assumptions were made in order to have needed conservatism:

- K_{eff} is calculated for fresh nuclear fuel with highest enrichment of ^{235}U – 4.4% for TVSA fuel assembly;
- the burnable absorber is not considered – only guide tubes with water;
- the UO_2 density is considered with the highest value – 10.7 g/cm^3;
- neutron absorbers are not considered in the spent nuclear storage;
- such a combination of geometrical characteristics of the equipment is considered, that K_{eff} is greatest;
- the maximum amount of baskets consider is the one that can be physically located in the spent nuclear storage;
- only the part of fuel assembly in which is located the UO_2 fuel with height 353 cm is modeled;
- the central tube is modeled like guide tube, spacer grids are not modeled because of their negligible influence on subcriticality.

4 Results Obtained with SCALE 6.1

Results obtained from criticality calculations – K_{eff} values of single TVSA fuel assembly in water and 37/3 basket with 12 spent fuel assemblies TVSA are presented in Table 1.

In Table 2 and Table 3 are presented the criticality calculations of dry spent fuel cask TK-13/3 ready for transportation (Table 2) and filled with water on the site of Spent Fuel Storage Facility (Table 3).

Table 4 presents criticality calculations for spent fuel for two cases: normal operation with 56 baskets and emergency conditions with 70 baskets. The calculations are performed with two sets of water temperature corresponding to operational and safety level ($45°$C and $65°$C).

Table 1. K_{eff} of fuel assembly and basket in water (neutrons per generation: 1 000; number of generations: 500)

Water density, g/cm^3	K_{eff}	
	Fuel assembly TVSA in water	**Basket 37/3 with 12 TVSA FA in water**
0.9997001 ($T = 10°$C)	0.8602 ± 0.0013	0.9061 ± 0.0011
0.9989511 ($T = 15°$C)	0.8596 ± 0.0012	0.9080 ± 0.0011
0.9982032 ($T = 20°$C)	0.8589 ± 0.0012	0.9073 ± 0.0012
0.9956192 ($T = 30°$C)	0.8574 ± 0.0012	0.9073 ± 0.0013
0.9921619 ($T = 40°$C)	0.8589 ± 0.0011	0.9077 ± 0.0010
0.9880440 ($T = 50°$C)	0.8564 ± 0.0011	0.9094 ± 0.0015
0.9831874 ($T = 60°$C)	0.8533 ± 0.0011	0.9082 ± 0.0012
0.9778038 ($T = 70°$C)	0.8510 ± 0.0013	0.9071 ± 0.0011
0.9719110 ($T = 80°$C)	0.8485 ± 0.0011	0.9050 ± 0.0012
0.9653441 ($T = 90°$C)	0.8431 ± 0.0011	0.9065 ± 0.0011

Table 4. K_{eff} of spent fuel storage with 56 and 70 baskets (neutrons per generation: 1 000; number of generations: 500)

Model	K_{eff}	
	56 baskets	**70 baskets**
Water fulfilled, $T_{\text{water}} = 45°$C	0.9064 ± 0.0011	0.9077 ± 0.0012
Water fulfilled, $T_{\text{water}} = 65°$C	0.9055 ± 0.0011	0.9063 ± 0.0011

Table 5 presents calculation results for K_{eff} for 4 spent fuel casks placed in Central Hall of Spent Fuel Storage Facility.

5 Validation

In order to validate the results of calculations from SCALE6.1 in INRNE-BAS were carried out independent criticality analyses of spent fuel storage by SCALE 6.0 [4]. The input data of the spent nuclear fuel and storage facility correspond to the calculations with SCALE 6.1, with the following differences:

- in one of the cases of SCALE 6.0 calculations, the height of neutron shielding is equal to the height of the fuel cask height – 603.5 cm;

Table 2. K_{eff} of dry cask TK-13/3 prepared for acceptance in spent fuel storage facility (neutrons per generation: 1 000; number of generations: 500)

Model	K_{eff}	
	Neutron shielding antifreeze	**Neutron shielding water**
TK 13/3 cask, without water, $\rho_{\text{air}} = 0.001089$ g/cm^3 ($T = 50°$C)	0.32443 ± 0.00040	0.32380 ± 0.00036
TK 13/3 cask, without water, $\rho_{\text{air}} = 0.001056$ g/cm^3 ($T = 60°$C)	0.32448 ± 0.00039	0.32314 ± 0.00039
TK 13/3 cask, without water, $\rho_{\text{air}} = 0.001025$ g/cm^3 ($T = 70°$C)	0.32325 ± 0.00038	0.32371 ± 0.00043
TK 13/3 cask, without water, $\rho_{\text{air}} = 0.000996$ g/cm^3 ($T = 80°$C)	0.32363 ± 0.00035	0.32300 ± 0.00037

Table 3. K_{eff} of cask TK-13/3 filled with water on site of spent fuel storage facility (neutrons per generation: 1 000; number of generations: 500)

Model	K_{eff}	
	Neutron shielding antifreeze	**Neutron shielding water**
TK 13/3 cask, in with water, $\rho_{\text{water}} = 0.9881423$ g/cm^3 ($T = 50°$C)	0.9089 ± 0.0011	0.9128 ± 0.0013
TK 13/3 cask, in with water, $\rho_{\text{water}} = 0.98328417$ g/cm^3 ($T = 60°$C)	0.9078 ± 0.0011	0.9098 ± 0.0013
TK 13/3 cask, in with water, $\rho_{\text{water}} = 0.9775171$ g/cm^3 ($T = 70°$C)	0.9096 ± 0.0012	0.9094 ± 0.0012
TK 13/3 cask, in with water, $\rho_{\text{water}} = 0.9718173$ g/cm^3 ($T = 80°$C)	0.9073 ± 0.0011	0.9036 ± 0.0012

Table 5. K_{eff} for 4 transport casks (neutrons per generation: 1 000; number of generations: 500)

Model	K_{eff}		
	Neutron shielding antifreeze	**Neutron shielding polythene**	**Neutron shielding water**
4 TK 13/3 casks, water, $\rho_{\text{water}} = 0.9881423$ g/cm^3 ($T = 50°$C)	0.9101 ± 0.0011	0.9103 ± 0.0012	0.9097 ± 0.0011
4 TK 13/3 casks, water, $\rho_{\text{water}} = 0.98328417$ g/cm^3 ($T = 60°$C)	0.9111 ± 0.0011	0.9118 ± 0.0012	0.9090 ± 0.0011
4 TK 13/3 casks, water, $\rho_{\text{water}} = 0.9775171$ g/cm^3 ($T = 70°$C)	0.9104 ± 0.0011	0.9090 ± 0.0011	0.9072 ± 0.0012
4 TK 13/3 casks, water, $\rho_{\text{water}} = 0.9718173$ g/cm^3 ($T = 80°$C)	0.9101 ± 0.0010	0.9106 ± 0.0011	0.9059 ± 0.0013

- in calculations with SCALE 6.0 the water level in the pool corresponds to operational and safety level (620 cm and 580 cm) as the 0.8 m below the bottom elevation is not modeled;
- all SCALE 6.0 calculations are performed for water density 0.9689 g/cm^3, corresponding to 85°C;

Sensitivity analyses of K_{eff} are performed:

- fuel assembly modeling with and without spacer grids;
- neutrons per generation (1000 and 500) and number of generations (15000 and 1000).

The analysis of sensitivity for these parameters shows an insignificant influence – within the statistical uncertainty. Therefore, comparative analyzes are performed without modeling of the spacer grids with following conditions: number of generations 500, neutrons per generation 1000. The results of the comparative analysis are summarized and presented in Table 6.

The negligible differences in K_{eff} for the various models shows that the spent fuel (TVSA fuel assemblies) and the spent fuel equipment for transport and storage are correctly modeled.

6 Conclusions

The results in all performed calculations demonstrate that the criticality safety criteria are met and the effective multiplication factor K_{eff} is lower than the regulatory requirements for subcriticality 0.95, taking into account the uncertainties of the models and input data.

Independent criticality analyses of spent fuel storage were carried out in order to validate the results from calculations The intercomparison for K_{eff} of both code versions demonstrates good agreement as shown in Table 6 and are bellow regulatory requirements for subcriticality 0.95.

The transport and technological equipment and conditions at Kozloduy NPP spent fuel storage facility provide safety transport and storage of spent nuclear fuel.

Table 6. Intercomparison between SCALE code versions 6.0/6.1

Model	$K_{\text{eff}} \pm \sigma$		$\Delta K = K_{\text{eff}}$
	INRNE	**ENPRO**	**(INRNE)** $-K_{\text{eff}}$ **(ENPRO)**
1. Fuel assembly TVSA, 4.4%U-235, water $T = 20°$C, fuel cell pitch 32.5 cm	0.8593 ± 0.0011	0.8589 ± 0.0013	0.0004
2. Basket 37/3 with 12 FA, $T_{H_2O} = 20°$C, fuel pitch in pool – 160 cm	0.9060 ± 0.0012	0.9061 ± 0.0012	-0.0001
3. Pool with 56 baskets, $T_{H_2O} = 45°$C, $H_{H_2O} = 620$ cm	0.9042 ± 0.0013	0.9069 ± 0.0012	-0.0027
4. Pool with 56 baskets, $T_{H_2O} = 65°$C, $H_{H_2O} = 580$ cm	0.9035 ± 0.0012	0.9065 ± 0.0013	-0.0030
5. Pool with 70 baskets, $T_{H_2O} = 45°$C, $H_{H_2O} = 620$ cm	0.9051 ± 0.0011	0.9027 ± 0.0011	.0024
6. Pool with 70 tightly placed baskets, $T_{H_2O} = 45°$C, $H_{H_2O} = 620$ cm	0.9072 ± 0.0011	0.9098 ± 0.0012	-0.0026
7. Pool with 70 baskets, $T_{H_2O} = 65°$C, $H_{H_2O} = 580$ cm	0.9036 ± 0.0012	0.9058 ± 0.0010	-0.0022
8. HOG with 70 tightly placed baskets, $T_{H_2O} = 65°$C, $H_{H_2O} = 580$ cm	0.9047 ± 0.0012	0.9062 ± 0.0011	-0.0015
9. TK-13/3 cask with 12 FA in water, with neutron shielding*, $T_{H_2O} = 85°$C	0.9082 ± 0.0011	0.9081 ± 0.0010	-0,0001
10. TK-13/3 cask with 12 FA in air, with neutron shielding*, $T_{\text{air}} = 20°$C	0.37335 ± 0.00051	0.32590 ± 0.00041	0.04551
11. TK-13/3 cask with 12 FA in air, with neutron shielding**, $T_{\text{air}} = 45°$C	0.32423 ± 0.00043	0.32590 ± 0.00041	-0.00167
12. 4 fuel casks with water, $T_{H_2O} = 85°$C, pitch 400 cm	0.9067 ± 0.0012	0.9104 ± 0.0012	-0.0037

*neutron shielding modeling with height equal to the height of the fuel cask;

**with real geometry modeling of the fuel cask neutron shielding

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