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# DEVELOPMENT OF OPTIMAL FUEL LOADING CONFIGURATIONS FOR ARMENIAN NPP SPENT NUCLEAR FUEL TRANSPORT CASK

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Since transition of the fuel used in NPP from initial enrichment of 3.6% to 3.82% a necessity arises of revaluation of the security of transportation containers of spent nuclear fuel and based on it appropriate design modifications. To ensure subcriticality of containers borated steel insets are used. In this paper the results of study of optimal spatial configuration of borated and regular stainless steel guide sleeves configurations that allow meeting regulatory requirements on criticality safety. The model of the SNF transport cask was developed by KENO-VI code of SCALE 6.1 package. Isotopic composition of WWER-440 SNF was calculated by ORIGEN-S code of SCALE 6.1 package.

***Keywords*:** spent nuclear fuel, criticality safety, spent fuel transport and storage, burnup credit.

**Introduction.** Armenian NPP (ANPP) SNF transport cask is used for transport of WWER440 type spent fuel assemblies (SFA) form ANPP spent fuel pool into NUHOMS type dry spent fuel storage [1]. Criticality safety of the transporting SFA is provided by spacing between them as well as using borated stainless steel (BSS) guide sleeves. Initial design of the cask for 3.6% enriched fuel assemblies includes 24 BSS and 32 SS guide sleeves. Increasing of the enrichment of the fuel assemblies used by ANPP requires increase of the number of BSS guide sleeves. This work devoted to the development of optimal configurations of BSS and SS guide sleeves containing minimal number of BSS sleeves while meeting regulatory requirements on criticality safety [2] ($k_{eff} \leq 0.95$).

**Criticality Safety Analysis.** In order to assess the criticality safety of the SNF dry storage transfer cask the following conservative assumptions were applied:

✓ fresh fuel was used in the model instead of burned fuel;

✓ the enrichment of nuclear fuel was chosen 4%, which corresponds to the maximal enriched fuel rod enrichment in 3.82% average enriched fuel assembly [3];

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✓ density of fuel pallet was chosen $10.7 g/cm^3$, which is corresponding to the maximal value of fuel pallet density [3];

✓ diameter of fuel pallet was chosen 7.57mm which is corresponding to the maximal value of fuel pallet diameter [3];

✓ the boron concentration in water was neglected (0 ppm Boron);

✓ for the stainless steel density used in the dry storage transfer cask (DSTC) cells was modeled the minimal value $7.73 g/cm^3$ [3];

✓ moderator temperature in the DSTC was assumed $21^0 C$;

✓ the structural materials such as top and bottom nozzle, grids are neglected;

✓ the Zn–Nb alloy for the cladding material of the fuel rod and assembly skeleton is assumed to be pure Zr, which is less absorbent than the actual alloy;

✓ the assembly wall thickness has been taken with the minimum value which is more conservative.

ANPP DSC and WWER 440 fuel assembly models were developed by KENO-VI code [4] of SCALE 6.0 system, which is verified for WWER-440 fuel [5]. As neutron-nucleolus interactions cross-sections library 238 group ENDF/B-VII.0(V7-238) library was used [6]. For generation of self-shielded multi-group cross sections CENTRM [7], program was used. Spent nuclear fuel Assemblies isotopic composition were calculated by ORIGEN [8] program. Independent uncertainty parameters were evaluated separately and the total $k_{eff}$ uncertainty was calculated as the root sum square of the individual $k_{eff}$ uncertainty values: $\Delta k = \left(\sum_i \Delta k_i^2\right)^{1/2} = 0.037$. Combined $k_{eff}$ uncertainty due to fabrication tolerances, depletion uncertainty and statistical uncertainty was estimated $\Delta k = 0.037$. To ensure proper sampling and source convergence as well as statistical reliability of calculated $k_{eff}$ values following Monte Carlo simulation parameters were used: numbers of neutrons per generation, of modeled neutron generations, of skipped neutron generations are 10000, 500, 200 respectively.

**Isotopic Composition Analysis.** Implementation of the burnup credit approach requires calculation of such an isotopic composition that leads to conservative results in case of criticality safety analysis. The major parameters that significantly influence on the reactivity of spent fuel in depletion analyzes for WWER [9]:

✓ Specific power: in case of Actinides, only approach high specific power produces the most conservative result. Therefore, for getting conservative results three year fuel cycle was assumed with maximal possible specific power to reach desired discharge burnup.

✓ Fuel temperature: to get conservative isotopic composition maximal fuel temperature was assumed (1000 $K$), since at higher temperatures, resonance absorption in U-238 is increased due to Doppler broadening, resulting increased plutonium production.

✓ Moderator temperature/density: to get conservative isotopic composition maximal moderator temperature/minimal density was assumed, since it leads to hardening of neutron spectrum and consequential increased plutonium production.

✓ Soluble boron acid concentration: maximal value of average soluble boron acid concentrations in the past ANPP fuel cycles ($4g\ H_3BO_3/kgH_2O$) was used, since it leads to hardening of neutron spectrum and consequential increased plutonium production.

The depletion analysis of WWER-440 fuel assembly with average 3.82% enrichment was carried out by ORIGEN-S program for different burnup values: 10, 20, 30, 40 GWd/TU. Based on ISG8-R3 recommendations [9] and nuclide importance analysis carried out in [10] following isotopes were used in criticality safety analysis: U-234 ,U-235, U-238, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, Am-241. Since in case of cooling pool, fuel assemblies can be immediately loaded from reactor cores, therefore, shortest value of final cooling time was chosen that brings to the maximal reactivity (100 days).

**Results.** Criticality study analysis showed that the DSC filled with average 3.82% enrichment fuel assemblies meets the regulatory requirements on criticality safety using 30, 32 and 34 BSS with corresponding spatial configurations (see Fig. 1). The results of the Burnup Credit methodology implementation using only Actinides are shown in Fig. 2.

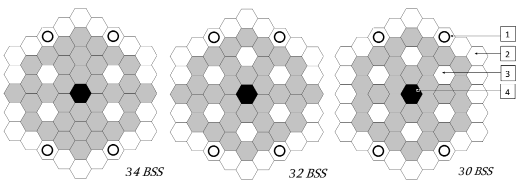


Fig. 1. DSTC optimal spatial configuration with 34, 32 and 30 BSS cells in the case of Fresh Fuel approach: 1 – Guide sleeves; 2 – SS cells; 3 – BSS cells; 4 – Central water filled cell.

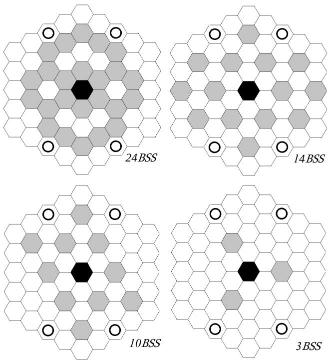


Fig. 2. Spatial configurations with minimal BSS cells in the case of Burnup Credit approach for 10 (24 BSS), 20 (14 BSS), 30 (10 BSS) and 40 (3 BSS) GWd/TU burnup values.

The values of $k_{eff}$ are shown in Table:

*The results of criticality safety study of the DSTC*

| Method | BSS minimum number | $k_{eff}$ | $k_{eff}+3\sigma$ |
|---|---|---|---|
| Fresh Fuel | | | |
| Case 1 | 30 | 0.91391 | 0.94909 |
| Case 2 | 32 | 0.89905 | 0.93423 |
| Case 3 | 34 | 0.89217 | 0.92735 |
| Burnup Credit | | | |
| 10 GWd/TU | 24 | 0.9147 | 0.94990 |
| 20 GWd/TU | 14 | 0.9130 | 0.94818 |
| 30 GWd/TU | 10 | 0.9060 | 0.94118 |
| 40 GWd/TU | 3 | 0.9090 | 0.94418 |

**Conclusion.** The results of the optimal configuration and criticality study of DSC showed that in the case of fresh fuel assumption regulatory requirements on criticality safety of the DSC could be met using 30 BSS cells. In the case of Actinides Only method of the Burnup Credit approach, the regulatory requirements on criticality safety for the 10, 20, 30 and 40 GWd/TU burnup values can be satisfied using 24, 14, 10 and 3 BSS cells correspondingly .

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