Neutronic Analysis of Ventilated Dry Storage Cask with Monte Carlo Method

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ABSTRACT

Turkey is constructing its first nuclear power plant of VVER-1200 type. The necessity of beginning of planning on nuclear waste management at the early stages of the construction is a known fact. In the meantime, dry storage casks are widely used worldwide for interim storage until final repository is on-line. Therefore, it is very important to build capacity on neutronic, thermal-hydraulic, and mechanical analysis of dry storage casks. This study focuses on neutronic analysis and gamma dose calculations of air ventilated dry storage cask design VSC-24 which is in use for VVER-1000 type reactor spent fuels. The MCNP6 code, which is based on the Monte Carlo method, was selected as simulation tool. Spent fuels with different pool residence times (5 and 10 years) were considered. In addition, different options of concrete thicknesses on dose rates are investigated. Depending on the VVER-1000 core at hand, the spent fuel composition therefore the source term was calculated with MCNP6.2 for assembly which has 16.63 MW-day/kgU burnup. In criticality calculations, k_{eff} value obtained as 0.6746 (below the limit value 0.95) with standard deviation of 0.0001. Dose rates consisting of gamma radiation were calculated at outer surface of the cask were calculated with fuel assemblies for 5 years and 10 years pool cooling times, 26 μSv and 0.68 μSv were obtained, respectively. Therefore, the capacity of modelling and simulating dry storage casks, is enhanced.

Keywords: dry storage, VSC-24, MCNP, dose rate

1 INTRODUCTION

The radioactive spent fuels removed from the reactor at the end of the determined cycles are kept in the storage pools at the reactor site. Due to the limited capacity of the pools, it is often preferred to apply dry storage before the final disposal. The advantages of this method are the large space capacity in dry storage, long-term storage, high economic efficiency and dual-purpose use (storage and transportation).

Although the dry storage method is used primarily in the USA, Canada, Russia and China, there are also studies of this method in countries such as South Korea, France and Germany. Castor 440/84 design dry storage casks are used for VVER-400 fuels in Czech Republic [1].

The fuels can be placed in the steel dry casks after they have been kept in the spent fuel pool for at least 1 year. When the cask is removed from the water, it is closed with bolts or welded and filled with filler gas to improve the heat transfer generated inside. It is then placed inside another cask section, which will provide shielding and is usually made of concrete [2].

Safety assessments should be supported with numerical tools, in order to minimize the harmful effects of the fuel on the environment during the storage period, until it goes to the final disposal area and to create estimates for long-time storage periods. Monte Carlo codes using automatic variant reduction techniques [3] are the most precise methods for these predictions.

Radiation level criteria applicable to storage units may vary depending on regulatory agencies. Considering the criteria in Belgium [4], the criteria applied for storage are similar to the criteria for transportation. The radiation level should not exceed 2 mSv/h at any point on the outer surface of the cask and should not exceed 0.1 mSv/h at the distance of 2 m. To give an example from the Switzerland regulations, the radiation level on the cask surface should not exceed an average of 0.5 mSv/h [4].

In this study, the shielding efficiency of the VSC-24 dry storage cask, which can be used for the transportation and storage of spent nuclear fuels that will be formed as a result of the operation of the VVER type reactor has been investigated. For this purpose, the current MCNP6.2 Monte Carlo transport code [5] and ACE type continuous energy cross-section library based on ENDF/B-VII.1 evaluated data [6] were used in gamma dose calculations. The effects of different pool cooling times on the gamma dose on the radial and axial outer surfaces of the cask were compared. In addition, the effect of changes in the thickness of the concrete section, which is the outermost layer of the cask, on the gamma dose on the surface was also investigated.

2 VSC-24 CASK DESIGN

The VSC-24 is manufactured by the Sierra Nuclear Corporation [7]. The containers are updated to VVER-1000 hexahedral fuel assemblies, each storing 24 fuel assembly. The VSC-24 system is designed for a 50-year lifetime.

The inside of the cask is a cylindrical box containing basket structure used to support fuel assemblies called MSB (Multi-Assembly Sealed Basket), also shown in Figure 1. This basket is placed inside a cylindrical reinforced concrete ring called VCC (Ventilated Concrete Cask) with an outer diameter of 3.35 m and a total height of 5.3 m. One other part of cask is MSB Transfer Cask (MTC) and it’s used to move the concrete cask on and off the heavy haul transfer trailer.

The cask includes two cover sections, shielding lid and structural lid. In addition to the RX-277 neutron shielding material and the protective section made of steel, there is a structural cover made of a 76 mm steel disc. In order to increase the heat transfer inside the cask, before the covers are welded, the cask is filled with helium gas at atmospheric pressure. Also, some geometrical parameters are shown in Table 1.

Figure 1. VSC-24 dry storage container [8]

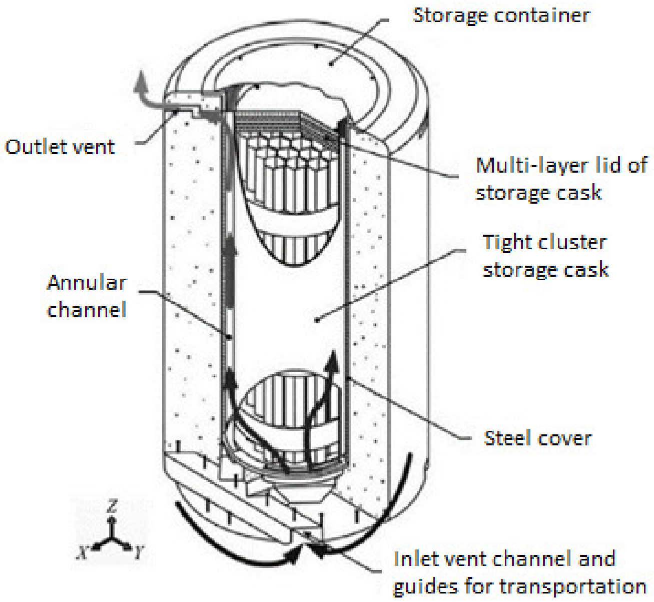

Figure 2. VSC-24 Cask component [9]

Table 1: VSC-24 cask parameters [9]

Parameter	**Value**
Fuel assembly	
Shape	hexahedron
Number of fuel rods	312
Length	3.837 m
Storage cask	
Num. of spent fuel assemblies	24
Height	4.973 m
Diameter	1.715 m
Inner medium	Helium
Pressure inside	1 atm
Container	
Height	5.809 m
Diameter	3.378 m
Width of annular channel	0.070 m

3 METHODOLOGIES

3.1 Gamma Source Distribution of SNF Assembly

In order to obtain the spent fuel compositions of the VVER-100 assemblies that to be loaded into the dry storage cask and to get plot the gamma source spectra associated with it, the VVER-1000 core simulation [10] with full power (3000 MW thermal) was run with MCNP6.2. Cinder90 module of MCNP6.2 was used for depletion analysis.

Cooling periods of 5 and 10 years were applied to the core fuel assemblies burned at full power for 1 year. One assembly selected for cask calculation has reached 16.63 MW-day/kgU burnup. It is assumed that the 24-fuel assembly to be loaded into the VSC-24 cask are identical and these assemblies contain 4.2% and 3.7% enriched UO2 fuels rods before they burn, and accordingly, gamma resource distributions were obtained from the spent fuel composition to make the cask calculations. The MISC, which is the MCNP6 intrinsic source constructor module [11], was used to generate the source energy spectrum. While preparing the input of the MISC, the atomic fractions were specified together with the isotopic material compositions that the spent fuel consists of. The gamma number depending on the energy ranges was obtained in the results, and the averaged gamma particle flux with their energies are drown depending on the different cooling times (5 and 10 years) are shown in results.

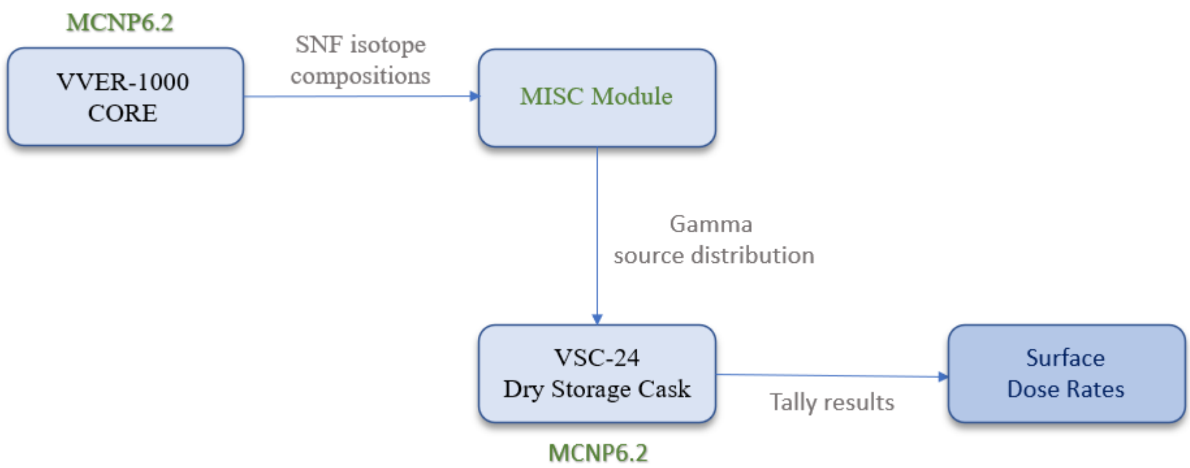

Figure 3. Flow chart of VSC-24 Cask surface dose calculations

3.2 VSC-24 MCNP6 Model

While creating the VSC-24 cask model, the geometric dimensions of the cask were adhered to the final safety analysis report [9]. The radial and axial section views in Figure 4 are coloured according to the materials. In the innermost part of the model, there is a hexahedral basket structure, where the fuel bundles are located. Each fuel bundle is defined to contain homogeneous clad, fuel and helium gas materials. There are MTC cylindrical shells consisting of steel, RX-277 and lead materials, respectively, from the inside to the outside, providing neutron and gamma shielding of the fuel bundles.

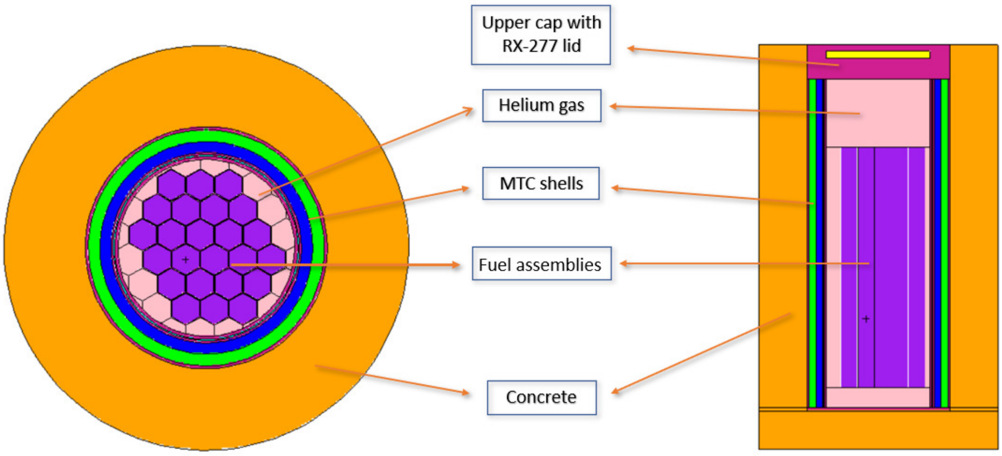

Figure 4. VSC-24 Cask MCNP6.2 model (radial and axial sectional views)

All 24 assembly locations of cask stored with same burned assembly which has 239 fuel rods with 4.2% and 66 fuel rods with 3.7% enriched UO_2 fuels and 6 burnable gadolinium rods at the begin of VVER-1000 core [10]. End of 1 year burn calculation pin-averaged burnup resulted with 16.63 MWd/kgU for this assembly. Outermost ring, there is a concrete cask. In order to accurately read the tally results in gamma dose calculations and to reduce the relative error, the outermost concrete section was divided into evenly spaced annular cells, and the importance of each cell was increased (with multiplying 2) from the inside to the outside. This geometry splitting and importance enhancement method allowed to reduce the relative errors in the tallies for particle counts taken at the outermost surface. Otherwise, since the number of particles reaching the outer surface in the simulation is low (especially for this low burnup-loaded cask scenario), it is difficult to get a low relative error rate for each energy group determined on the outermost surface. In addition, ENDF/B-VII.1 neutron and photon libraries were used in MCNP6 calculations.

4 RESULTS

4.1 Gamma Dose Rate Calculation

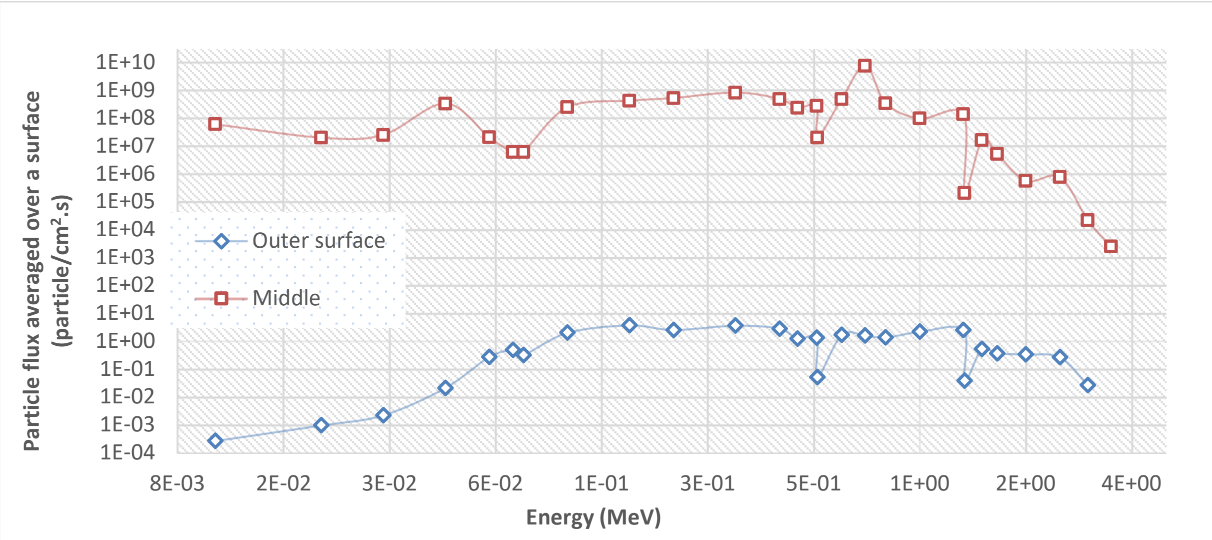

Figure 5. Cask gamma particle energy distribution with 10 years cooled spent fuel

Energy spectrum graphs of particle numbers on the middle and outermost surface of the cask, where cooled fuels are placed for 10 years, are shown in Figure 5.

4.2 Criticality Calculation

In the criticality calculation using the MCNP6.2 code and Continuous energy cross section libraries based on ENDF/B-VII.1 of the VSC-24 cask, which consists of 24 fuel bundles filled with helium gas and containing UO_2 fuels with an average of 4% enrichment, the final result of k_{eff} value was 0.67464 ± 0.00010. And also confidence intervals of 68% (0.67454 to 0.67474), 95% (0.67445 to 0.67483) and 99% (0.67438 to 0.67490) are below the criticality limit. MCNP calculations were carried out with 2E+5 particles and 30 inactive and 130 active cycles. This result is acceptable in terms of safety criteria (<0.95).

4.3 Concrete Thickness Alteration

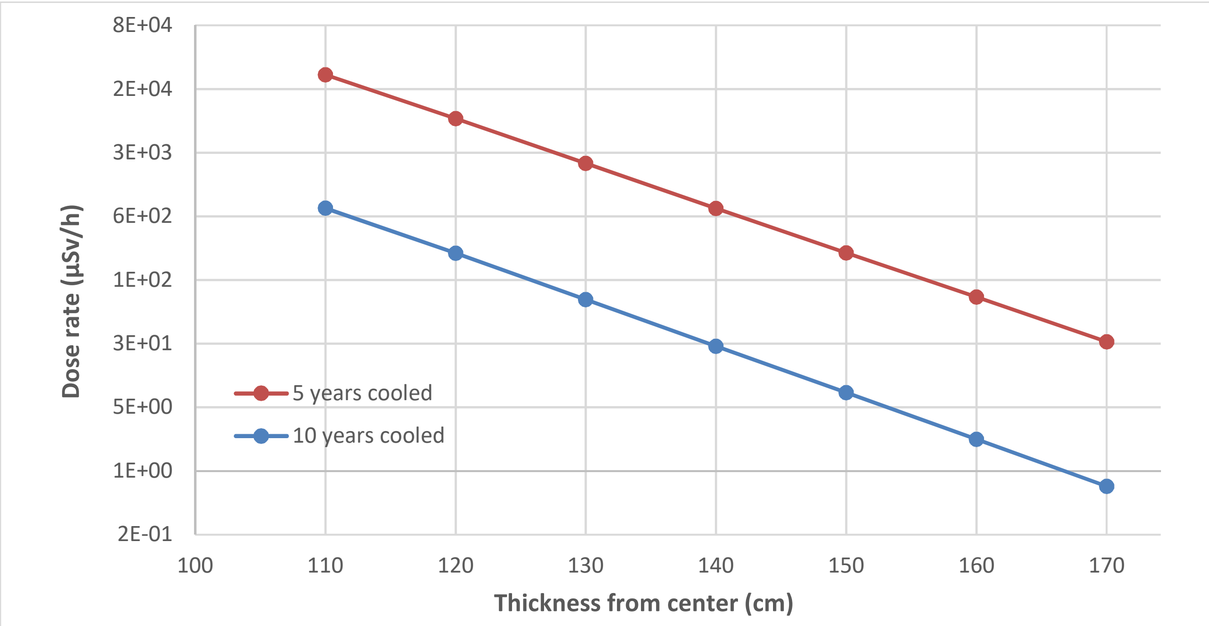

Figure 7. Gamma dose rates at cask surface after 5- and 10-years cooling period for reduced concrete thicknesses

By reducing the concrete thickness in 10 cm intervals, the dose rates on the surface area of the cask were calculated with the flux to dose rate conversion factors specified in the 1977 ANSI/ANS document [12] and shown in Figure 7. The reduction in concrete thickness resulted in a significant increase in dose rate.

5 CONCLUSION

The criticality calculation and shielding capacity of the VSC-24 type storage cask were examined. The dose rates to be formed by placing the used VVER-1000 fuels, which were kept in the pool for 5 and 10 years after being burned in the reactor for 1 year to reached average burnup of 16.63 MW-day/kgU, were taken into consideration. At the end of 10 years, a similar decrease in dose rate was observed as a result of significant degradation of short-lived radionuclides.

The criticality level of the cask filled with fresh fuel (average 4% enriched) and helium criticality results below the values determined as safety criteria. In addition, it can be concluded that the reduction of the thickness of the section consisting of concrete will cause significant dose changes for longer-term burnt fuels, although the surface dose does not exceed critical levels. When the dose rates on the surface of the cask were calculated with fuel assemblies for 5 years and 10 years pool cooling times, 26 μSv and 0.68 μSv were obtained, respectively. This study has improved the capacity by using current methods in the analysis of dry casks that can be used in the transport and storage of VVER type reactor fuels, the creation of the source term, and calculation of criticality and dose rates.

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