2D/3D Shielding Calculation and Dose Rate Estimation for a Spent Fuel Storage Cask considering an Axial Burn-up Profile

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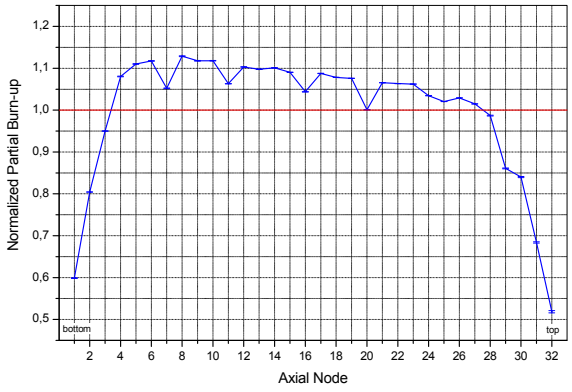
During irradiation in the reactor core, the burn-up distribution of nuclear fuel is not homogeneous but shows an axial dependency amongst others due to variations of in-core coolant temperature. This phenomenon called axial burn-up profile (ABP) is well known in criticality safety analyses but also comes into focus in the frame of shielding and dose rate calculations.
First investigations prove an influence of the ABP on the local dose rate on the surface of casks for spent nuclear fuel, with locations showing an above-average dose rate when considering the ABP in comparison to neglecting it.

> Typical PWR ABP from GKN II in 32 Nodes



Calculation Methods

Simplified (10 instead of 32 nodes), homogenized fuel assemblies with ABP inside a exemplary CASTOR V cask are modelled, featuring UO_2 fuel with 3.6 wt% ^{235}U initial enrichment, 40 GWd/tHM average burn-up and 10 years decay time.
Two different, independent calculation methods, MCNP5 and DORT, are applied. Axial inventories and sources were calculated with OREST and NGSRC and used by both codes.

MCNP5

3D stochastic Monte Carlo N-particle transportation code version 5, using sophisticated variance reduction and convergence granting methods.

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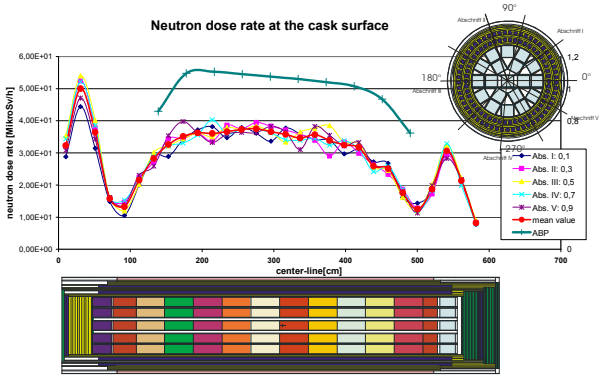
2D deterministic transport code system, based on DORT code from DOORS package, used in R-Z-geometry mode.

Results

MCNP5



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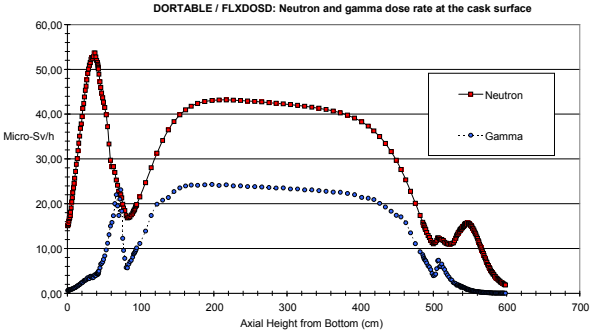

Good agreement found in both models for the dose rates along the cask surface. Peaks at the top and bottom are due to leakage effects at moderator plugs of the cask.

Conclusion

Introduction of ABP in shielding and dose rate calculations yields to an improved, realistic model of casks for transportation and storage of spent nuclear fuel. Local positions with increased dose rates as compared to models neglecting ABP were identified using two independent methods.
Further impacts will be investigated in the future.

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