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INVESTIGATING THE SHIELDING EFFECTIVENESS FOR SPENT FUEL DRY STORAGE CASK USING MCNP CODE

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Abstract: Nuclear spent fuel management has become a key activity within national nuclear strategies. In Jordan, the national policy for radioactive waste and spent nuclear fuel management stated establishing storage facilities near the nuclear research reactor for spent fuel storage. In this study, MCNP calculations of the neutron and gamma-ray doses rates arising from a proposed spent-fuel storage cask were described to examine its shielding effectiveness. The analysis includes, calculating the dose rate near and far from the cask, for different spent fuel cooling periods and for various concrete layer thicknesses. While the gamma-ray source strengths were used for spent fuel from the MIT research reactor, the neutron source strengths and energy spectrum were used for spent fuel from the IAEA 10 MW MTR benchmark reactor. Neutron calculated dose rates were found to be low with no potential increase in the total dose coming out of the spent fuel. However, both gamma and neutron dose rates were analysed and cask design was optimized to achieve effective shield with low cost.