NEUTRON SHIELDING FOR A HIGH CAPACITY STORAGE/TRANSPORT CASK

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Abstract

Neutron shielding for a large spent fuel storage cask (NAC S/T) is described. Difficulties in the analysis methods used included unconservative results observed for one-dimensional models, spectral shift with decay time for gammas which affected total dose calculations, and the necessity of neutron transport calculations to supplement point-kernel calculations based on the neutron moments method. Potential unconservatism of Health Physics surveys of such casks due to the fast neutron dose contribution is described.

The Nuclear Assurance Corporation (NAC) Storage/Transport Cask is a newly designed high capacity spent fuel cask that can contain up to 31 intact PWR assemblies or 56 consolidated PWR assemblies. A lead/steel cask body was selected to minimize weight for the high fuel loading and to eliminate concern for brittle fracture under low temperature conditions. A design fuel burnup of 35,000 to 45,000 MWD/MTU causes the neutron source to become the dominant shielding concern, given cool times of 5 to 10 years. A radial 8" thick neutron shield tank containing borated water maintains neutron and secondary gamma dose rates As Low As Reasonably Achievable (ALARA). A solid borated silicone rubber cap is required for the upper and lower cask ends except during fuel loading and handling, as shown in Fig. 1. Emphasis was placed on maintaining the neutron dose rate (ALARA) because the high energy neutron dose rate contribution is a substantial dose component, as shown in Fig. 2, and is difficult to measure in the field.[1,2] The total dose rate on contact at the ends of a cask being loaded with 31 PWR assemblies with a 35,000 MWD/MTU burnup and 10 year cool time is 300 mrem/hr, but a cursory Health Physics survey might not detect this at all as 93 percent of the dose rate is due to fast neutrons.

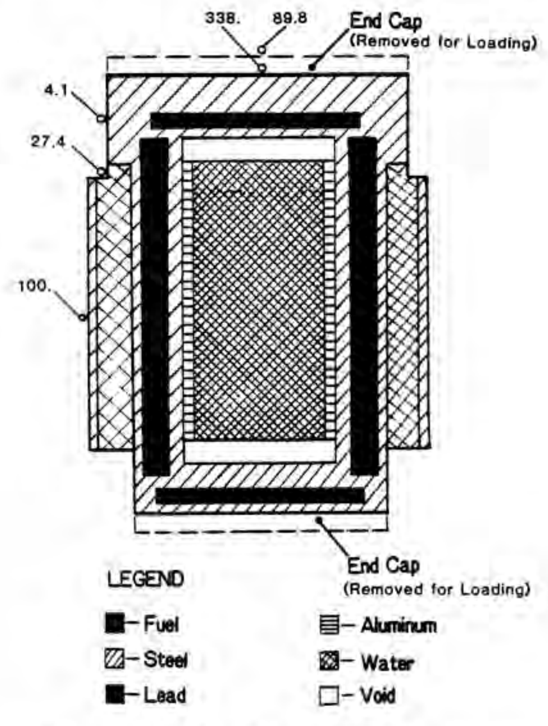

Fig. 1. NAC S/T Cask Dose Rates (mrem/hr).

It was noted in the shielding calculations that a one-dimensional (1D) equal area radial analysis does not yield the conservative results that would normally be expected, making a three-dimensional (3D) analysis mandatory. The effect of the actual fuel assembly or canister geometry is not discernable by a 1D radial analysis, which would normally be performed by circularizing the fuel region to create an equivalent cylinder with the same cross-sectional area as the actual fuel basket, as shown in Fig. 3. Tests with the QAD-CG[3] code showed that cask surface dose rates using an equivalent area cylinder are underpredicted by 46 percent for gammas and 42 percent for neutrons. This is because the actual cask basket positions some fuel assemblies closer to the cask surface than the average, as defined by the equivalent cylinder, which results in lessened attenuation and a dose rate peak at the adjacent surface. This effect diminishes in importance as the radial distance from the cask centerline increases, but never entirely disappears because of differences in source region attenuation and leakage and basket wall attenuation, between the circularized model and the actual geometry. A circumscribed circle model that represents the fuel region as a cylinder that completely encloses the fuel region has been shown to be conservative[4], but could impose an unnecessary penalty in licensed fuel burnup by overpredicting the resulting dose rates.

The gamma source spectrum becomes "softer" (less energetic) as the cool time (decay time) increases, and the total source strength decreases. The spectral shift prevents a simple scaling of the dose rate calculated for one cool time via the ratio of source strengths to obtain the dose rate at a different cool

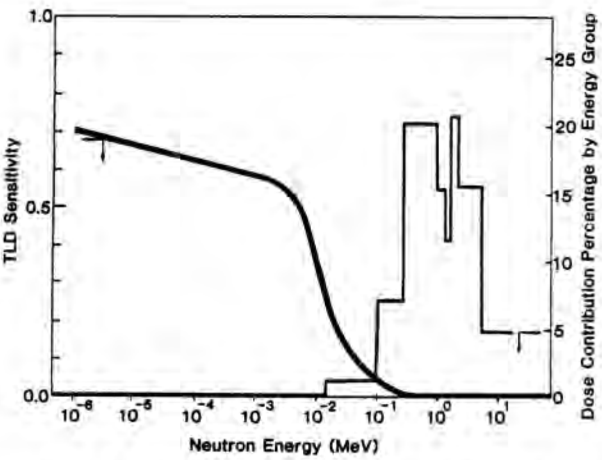

Fig. 2. Comparison of TLD Neutron Sensitivity to Dose Contribution for Neutrons of Energies from 1 eV to 10 MeV.

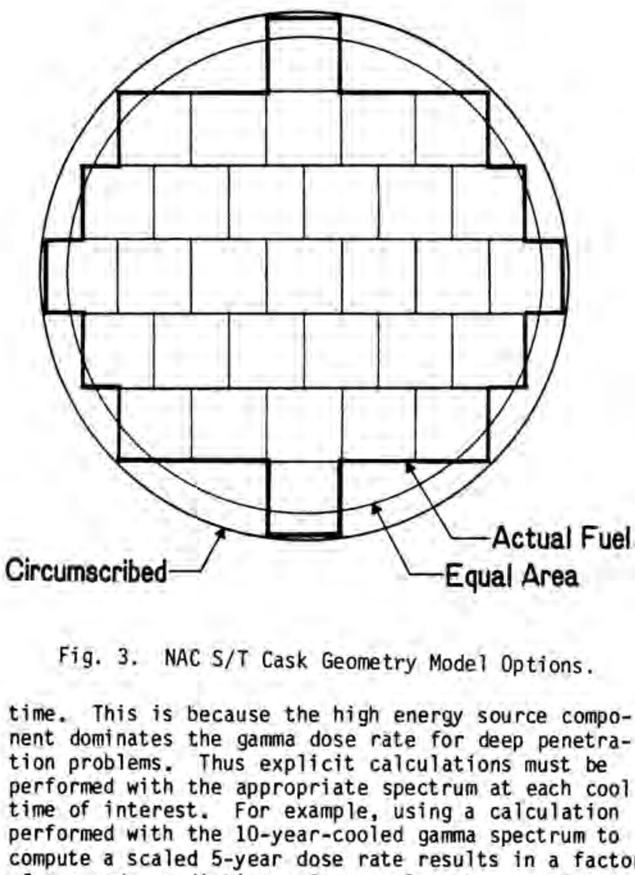

Fig. 3. NAC S/T Cask Geometry Model Options.

time. This is because the high energy source component dominates the gamma dose rate for deep penetration problems. Thus explicit calculations must be performed with the appropriate spectrum at each cool time of interest. For example, using a calculation performed with the 10-year-cooled gamma spectrum to compute a scaled 5-year dose rate results in a factor of two underprediction. Conversely, the use of a spectrum for a short cool time gives conservative, but unnecessarily high, dose rates for all later times.

The QAD computer code is often used for computation of 3D cask dose rates because of its ease of use and it is generally considered to be adequate for the analysis of conservative cask designs. The neutron moments method is used to approximate neutron transport, and this method normally produces reasonable or conservative results. Modern spent fuel casks usually employ an annular neutron shield that contains hydrogenous material as the outer layer of the cask body. At the cask ends, however, fabrication considerations discourage the use of neutron shield material and 3D QAD calculations would indicate that neutron shield material is not required at the cask ends. Such a conclusion would be erroneous, however, because the QAD neutron treatment does not apply in the absence of light nuclei (this is pointed out in the QAD manuals). QAD neutron calculations for the ends of the NAC cask underpredicted the dose rate by a factor of 20. This necessitated additional calculations with more accurate neutron transport methods to supplement the 3D QAD analysis.

The shielding analyses for the NAC Storage/Transport cask for high burnup fuels indicate that the neutron dose rate from fast neutrons is significant and can be seriously underpredicted through improper use of computer codes. Perhaps the most important aspect of these results is that Health Physics surveys must include sensitive, correctly calibrated neutron measurements as well as gamma surveys.

References

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