COMPARISON OF DOSE RATES CALCULATIONS WITH MEASUREMENTS AT A CASTOR HAW 20/28 CG CASK

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INTRODUCTION

The Federal Office for Radiation Protection (BfS) as the competent authority for package design approval in Germany is using the SCALE code system (SCALE = Standardized Computer Analyses for Licensing Evaluation) developed by Oak Ridge National Laboratory (ORNL) as a calculation tool for the shielding evaluation and in particular to decide whether a new package design meets all applicable dose rate limits of the transport regulations.

At PATRAM 2001 satisfactory results have been presented for model calculations with SCALE 4.4a compared with measurements in the neutron and gamma field caused by a CASTOR IIa cask which is used for spent fuel elements [1].

In continuation of this work the paper presents the comparison of dose rates calculated by SCALE 4.4a with the measured ones at a CASTOR HAW 20/28 CG. This type of CASTOR cask is used for the transport and interim storage of vitrified high active waste (HAW) from reprocessing.

For the calculations the SAS4 sequence of SCALE 4.4a has been used. The 28 HAW canisters as well as the basket and the cask body have been modelled in detail by MARS geometry. The measurements were carried out in January 2003.

The main conclusions of this comparison will be described.

DESCRIPTION OF THE CASK AND THE CONTENTS

Measurements and calculations were made on a dual purpose cask of type CASTOR HAW 20/28 CG for the transport and storage of vitrified high active waste (HAW) from reprocessing. The cask is interim stored in the Interim Storage Facility at Gorleben (Fig. 1).

Fig. 1 View on CASTOR® HAW 20/28 CG casks (first two rows) in the Interim Storage Facility at Gorleben

The CASTOR HAW 20/28 CG cask consists of a thick-walled radial finned cylindrical cask body made of ductile cast iron. The cask is closed during transportation from the reprocessing plant to the interim storage facility with a primary lid and during storage with a primary and a secondary lid, each made of stainless steel, and respective screws and seals. For a more effective neutron shielding polyethylene rods are built in into the cask body concentrically and polyethylene plates are placed at the cask bottom and on the primary lid. The outer dimensions of this cask in the storage configuration (without shock absorbers but with secondary lid and a protection cover) are about 2.3 m in diameter and 6 m in height; the loaded cask weight is about 115 tons (Fig. 2).

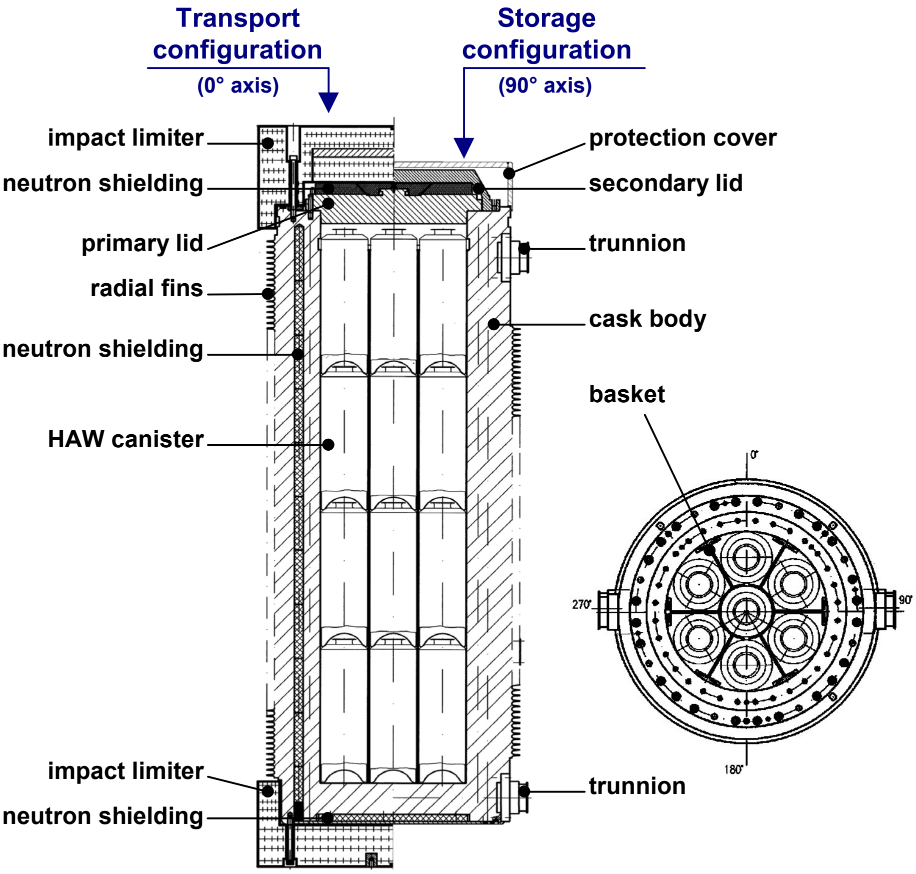


Fig. 2 CASTOR® HAW 20/28 CG cask
Schematic View of Transport and Storage Configuration

The radioactive waste (fission products and actinides) is incorporated into a stable glass matrix in a hermetically sealed stainless steel canister. The canisters are filled with HAW up to approximately 88% of their internal volume.
The gamma radiation around this cask mainly results from the fission products ^{134}Cs, ^{137}Cs and ^{154}Eu. Neutron radiation is mainly caused by spontaneous fission of ^{244}Cm and (α,n)-reactions of ^{244}Cm and ^{241}Am with oxygen or boron.
The cask is loaded with 28 canisters in four levels with 6 canisters arranged concentrically and one central canister in each level. For the calculations an unequal loading of the canisters with following ranges had to be considered:

^{134}Cs	31.2 ... 109.5 TBq/canister
^{137}Cs	4346 ... 5351 TBq/canister
^{154}Eu	74.6 ... 113.6 TBq/canister
^{241}Am	579.7 ... 956.4 g/canister
^{244}Cm	22.5 ... 39.8 g/canister

These values were derived from official data sheets but the accuracy of these values is not known.

RESULTS OF THE DOSE RATE MEASUREMENTS

The dose rate measurements were carried out at a stand-alone CASTOR HAW 20/28 CG at the Gorleben interim storage facility. For measuring gamma and neutron dose rates the following commercial devices (Fig. 3) were used:

gamma dose rate:

- LB 1236 (EG&G Berthold)
- AD 5 (Automess)

neutron dose rate:

- LB 6411 (EG&G Berthold)

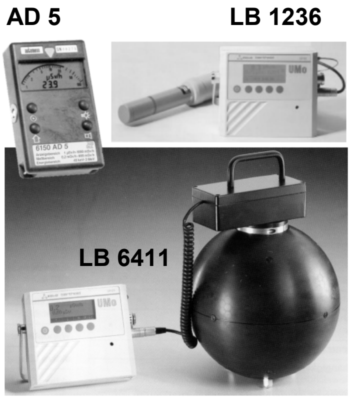
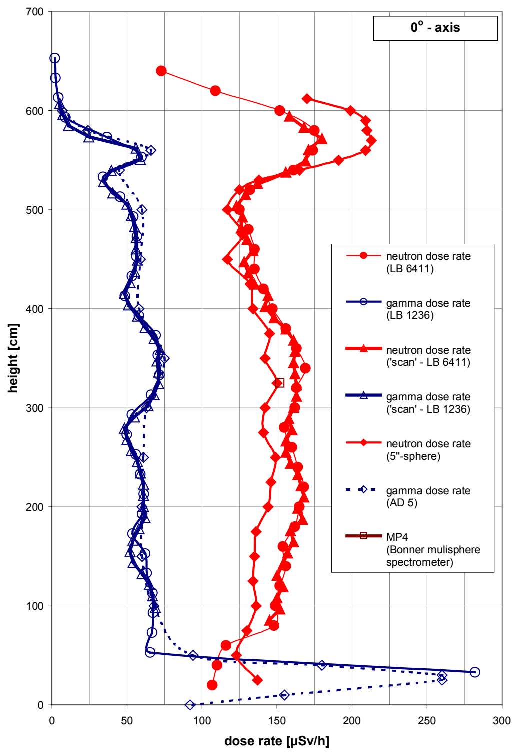
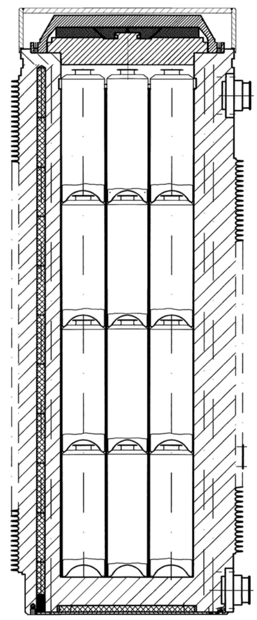


Fig. 3 Commercial instruments for dose rate measurements

At selected points the neutron dose rates were derived from neutron spectra measured by means of a Bonner multisphere spectrometer as reference value for the evaluation of the measured neutron dose rates by commercial devices.
The results of the axial distribution are diagrammed in Fig. 4 and 5. It should be noted that the distances from the center of the counter to the cask surface varied from 11 cm to 18 cm for gamma dose rate measurements and 16 cm to 23 cm for neutron dose rate measurement depending on the used measuring system. For a better visualisation of the measured data a schematic sectional view of the cask is added. The total uncertainty of the measured dose rates is assessed to be about ± 10 % (in most cases ± 2.5 %).





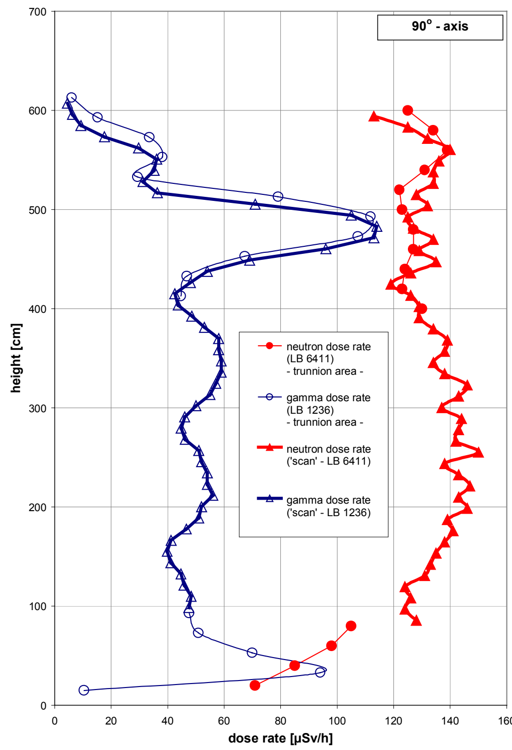


Fig. 4 Dose rate distribution along 0° axis

Fig. 5 Dose rate distribution along 90° axis

In this paper the measured dose rates are only discussed in a degree which is necessary for modelling the cask and the contents. For a more detailed discussion see the publication BfS-SE 01/03 [2].

At the 0° axis trunnions are not located (see Fig. 2). It can be seen in Fig. 4 that in addition to the increase of the gamma dose rates coming from the design and the filling of the canisters themselves two other areas with higher dose rates are measured at the top and bottom area resulting from the decrease of the body wall thickness for fixing the impact limiter. The higher neutron dose rate is a result of the arrangement of the polyethylene rods within the cask wall and the polyethylene plate between the two lids.

In Fig. 5 the dose rate distribution along the 90° axis is displayed where two trunnions are located. The areas with increased gamma dose rates, already found along the 0° axis, are also existent at the 90° axis. In addition milled areas around the trunnions (especially around the upper trunnion) which are necessary for using the lifting device lead to higher gamma dose rates. Due to a lower actinide contents of the corresponding canisters no increase of the neutron dose rate at the top area had been measured.

MODELLING THE CASK AND THE CONTENTS

In the SAS4 input file the geometry of the 28 HAW canisters as well as the basket and the cask body were modelled using the MARS geometry (see Fig. 6). Special attention was payed on modelling the regions where higher dose rates were measured (trunnion area, lid and bottom area - see Fig. 4 and 5).

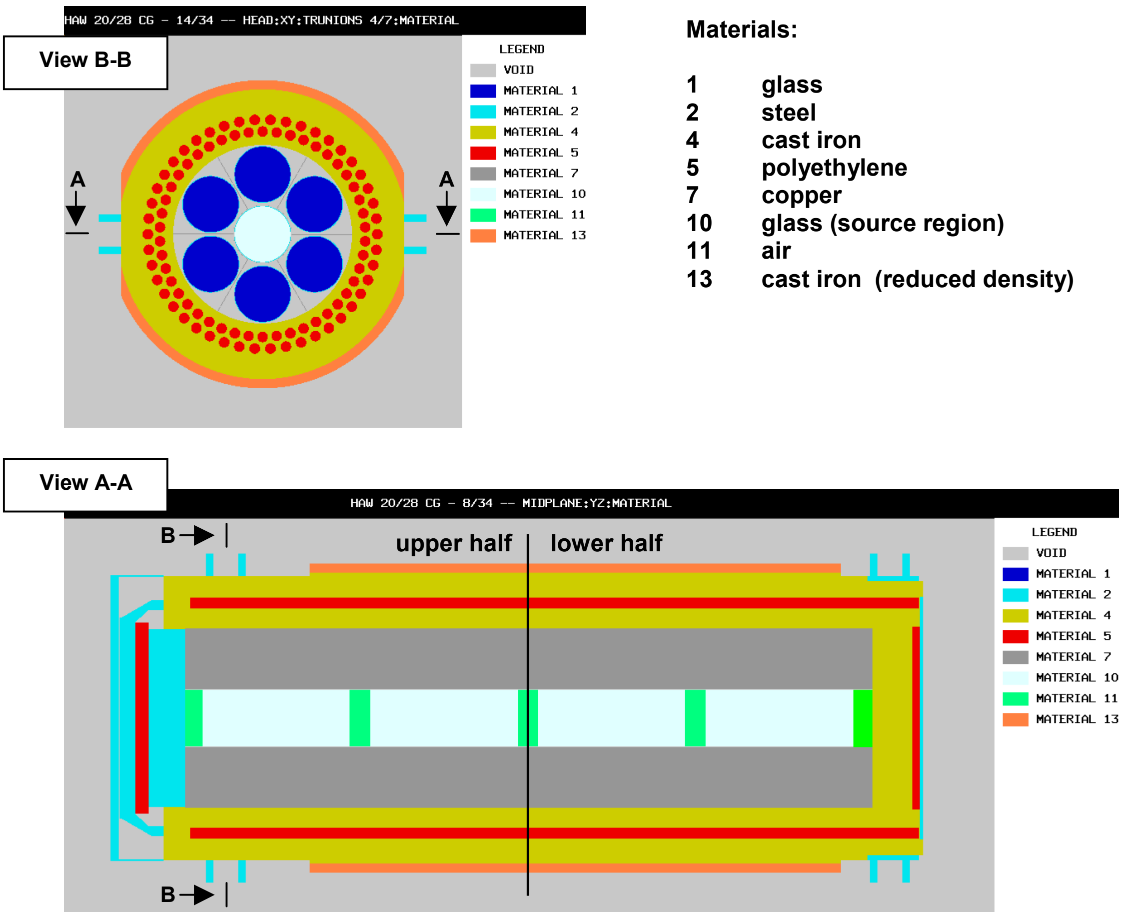


Fig. 6 Cask model

Due to internal parameter limitations of SCALE 4.4a and to reduce the modelling effort the radial fins have been modelled as a homogenised region of cast iron and void with reduced cast iron density. Nominal dimensions and material densities were used, except for the cask body material where a measured density was considered.
For the shielding calculation the usually used coupled 27 neutron group, 18 gamma-ray group library was chosen. Instead of the built-in conversion factors, conversion factors based on ICRP 74 [3] which represent the maximum conversion factor within the appropriate SCALE neutron energy group were used to obtain the 'ambient dose rate equivalent' $\dot{H}^*(10)$ according to ICRP 60 [4].

The dose rates were obtained from the SAS4 surface detectors. To minimise calculation efforts and to receive sufficient accurate results the surface detectors have been segmented every 10° around the cask with a height of 20 cm and located 16 cm from cask surface for both the gamma dose rate calculation as well as for the neutron dose rate calculation.
According to SAS4 requirements the calculations were carried out for head and bottom cask halves separately.

The content is originally described by fission products and actinides with their activities and masses, respectively. For the calculations these data had to be formatted according to SCALE requirements. The energy dependent source strengths were obtained by decay calculations using the ORIGEN-ARP software [5]. The axial deviations due to the different loading of the HAW canister have been considered by using a source strength profile.

RESULTS OF THE CALCULATION

The results of the calculation and the comparison to the measured dose rates are shown graphically in Fig. 7 and 8 for the gamma dose rate and Fig. 9 and 10 for the neutron dose rate.

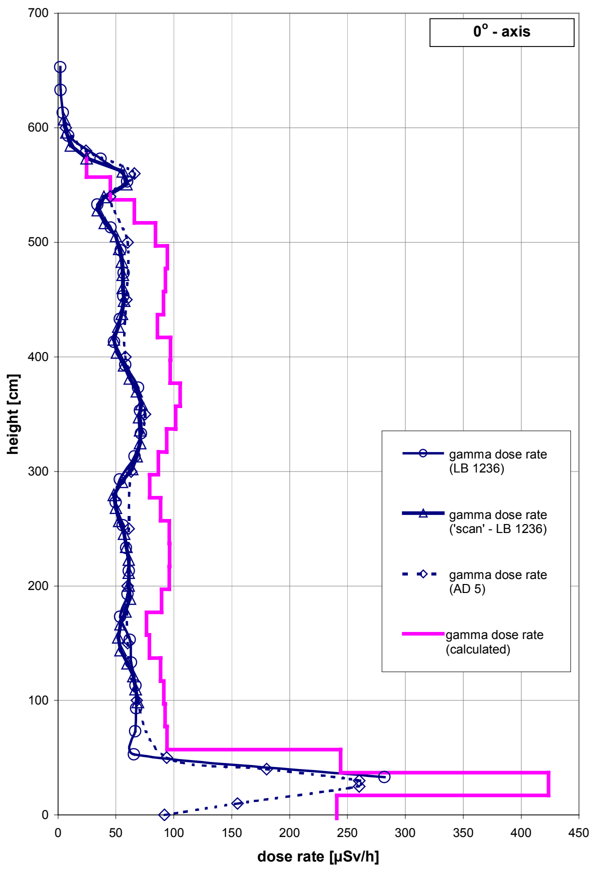


Fig. 7 Calculated and measured gamma dose rate along 0°-axis

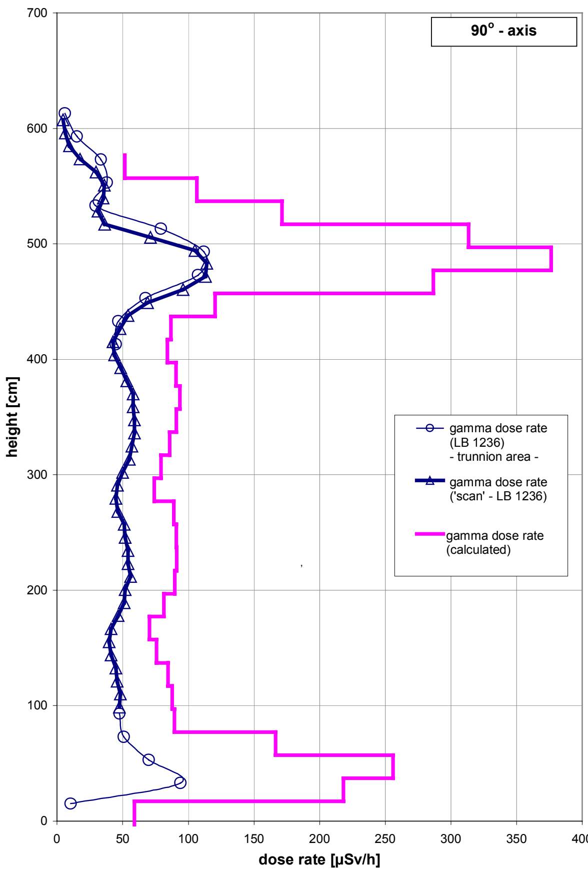


Fig. 8 Calculated and measured gamma dose rate along 90°-axis

The distinct axial distribution of the gamma dose rates as well as the weak points of the shielding are reflected by the calculation very well.
The measured gamma dose rates are overestimated by the calculated dose rates by up to a factor of 1.6 on the 0° axis of the cask. On the 90° axis the overestimation is approximately 1.7 for the middle part of the cask, and for the areas with a decreased wall thickness up to a factor of 2.7 ... 3.3. A circumferential slot at a cask height of approximately 5.5 m was not modelled and in so far the corresponding peak of the gamma dose rate couldn't be reflected by the calculation.

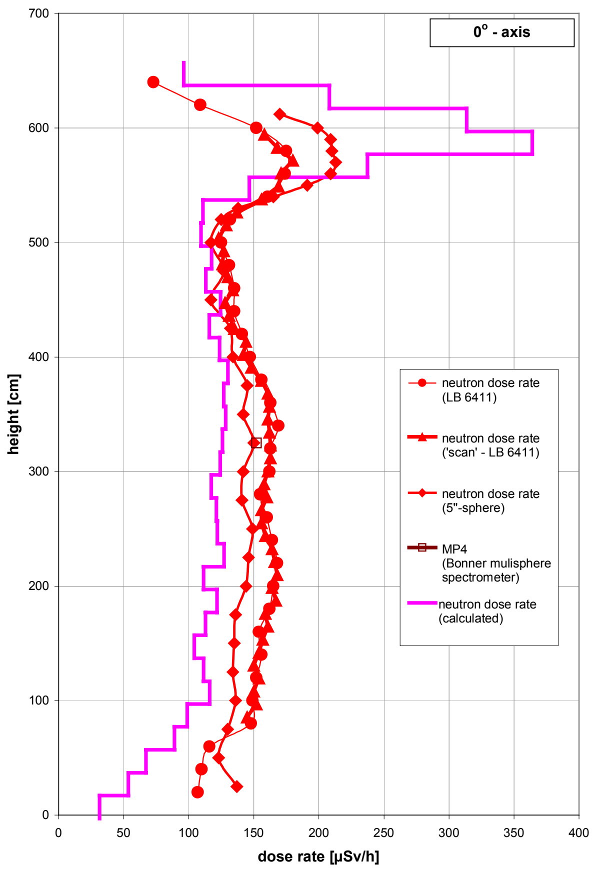


Fig. 9 Calculated and measured neutron dose rate along 0°-axis

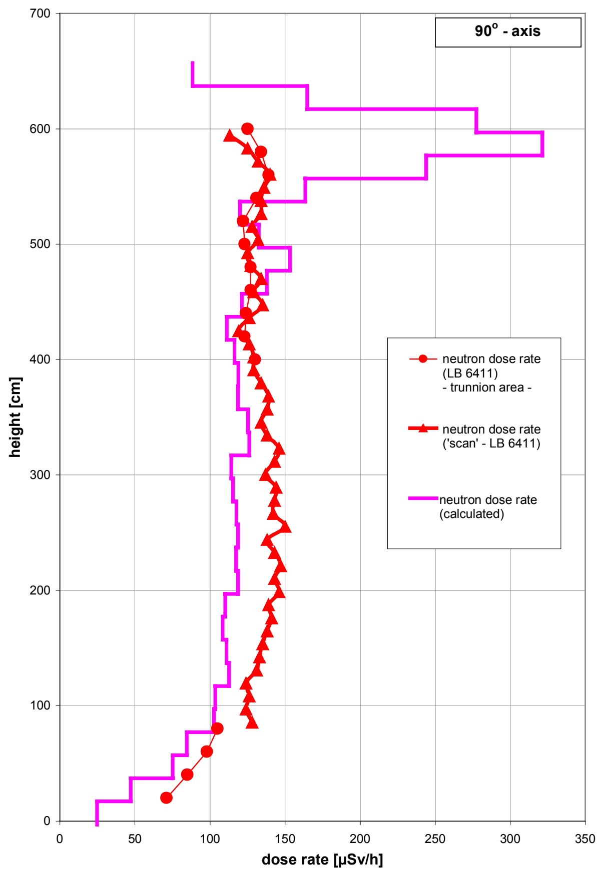


Fig. 10 Calculated and measured neutron dose rate along 90°-axis

Regarding the neutron dose rate distribution it can be noted that the calculated dose rates could reflect the distribution very well but the absolute dose rates are underestimated by the calculation by up to a factor of 1.4 for the middle cask part. This underestimation grows up to a factor of about 2 for the bottom area. At the top area of the cask where the neutron rods end the calculated neutron dose rates overestimate the measured ones. This can be attributed to the following main reasons:

- nominal filling heights of the HAW canisters have been considered (it was estimated that the real filling heights for the appropriate canisters can be up to 4 cm less than the considered nominal heights)
- simplifications due to modelling the canisters and to actinide inventory (e.g., for the 90° axis it was estimated that the real contents of the appropriate canisters is about 12% less than the considered one)

The calculated gamma and neutron dose rates have a standard deviation of ± 10 % at maximum.

CONCLUSIONS

The SCALE 4.4a shielding software can be used for assessment tasks (compliance with dose rate limits) and benchmark calculations of casks loaded with HAW successfully. Moreover, dose rate distributions due to an unequal loading of the HAW canisters can be calculated with sufficient results. This applies also to weak points of the shielding which are reflected by the calculation.
Further investigations will be made to clarify the remaining deviations between calculated and measured dose rates.

But nevertheless, great importance should be attached to:

- a detailled modelling of the cask according to the requirements of SAS4
- the accuracy of modelling the contents (specially the contents distribution)
- accuracy of the given source strengths (uncertainties)

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