VALIDATION OF THE SCALE 4.4A SOURCE TERM AND SHIELDING SOFTWARE BASED ON MEASUREMENTS AT A CASTOR IIA CASK

Frank-Michael Börst, Ingo Reiche
Federal Office For Radiation Protection, P.O. Box 100149, D-38201 Salzgitter

ABSTRACT

In Germany, the Federal Office for Radiation Protection (BfS) as the competent authority for package design approval has to evaluate whether a new package design meets all applicable dose rate limits of the transport regulations. For demonstration by calculation the SCALE code system (SCALE = Standardized Computer Analyses for Licensing Evaluation) developed by Oak Ridge National Laboratory (ORNL) is used.

This paper presents the validation of the SCALE 4.4a source term and shielding software for use with spent fuel transportation casks of the CASTOR family by comparing model calculations with measurements in the neutron and gamma field caused by a CASTOR IIa cask.

The measurements were carried out in 1995 at various distances above the lid and in the cask midplane and have been presented partly at PATRAM 1998 [1].

For the calculations the SAS4 sequence of SCALE 4.4a has been used. The fuel elements as well as the basket and the cask body have been modelled in detail by MARS geometry. To cover all detector locations, the model is based on the upper half of the cask. The source description has been taken from SAS2 calculations. Results have been obtained for different fluence-to-dose conversion factors.

BRIEF DESCRIPTION OF THE SCALE CODE SYSTEM

The SCALE code system was originally developed by ORNL to have a computational tool in the licensing procedure of package designs for the transport of spent fuel for performing criticality safety assessments. Since that time the code has been standardized to get an easy-to-use modular analysis system of well-established programs and data and extended with several modules to perform source term calculations, shielding and heat transfer analysis too.

For the work presented in this paper it was necessary to use the so called „control modules" SAS2 (SAS=Shielding Analysis Sequence) for source term calculation and SAS4 to perform the dose rate calculations.

SAS2 (also known as SAS2H, an enhanced version of SAS2) controls the calls of the „functional modules" to calculate via ORIGEN-S the isotopic concentrations, radiation source spectra and strengths needed for the dose rate assessment. The capability of SAS2 to obtain dose rates with the one dimensional XSDOSE module was not used.

The SAS4 control module uses amongst other functional modules the three-dimensional Monte Carlo code MORSE-SGC to calculate the dose rates at a cask. [2]

DESCRIPTION OF THE CASK AND THE CONTENTS

The validation was based on a CASTOR IIa cask because of the available accurate measurement data and because the cask design is similar to some CASTOR casks currently being approved. The CASTOR IIa cask, a dual purpose cask for the transport and storage of spent fuel elements, consists of a thick-walled finned square cask body with rounded edges made of ductile cast iron. The cask is closed 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 within the secondary lid. The outer dimensions of this cask in the storage configuration (without shock absorbers) are about 2m in length and width and 6m in height; the loaded cask weight is about 116 tons (Fig. 1).

The cask is loaded with 9 fuel elements of the Philippsburg Nuclear Power Plant (PWR). The fuel elements have a square lattice with 236 fuel rods which are filled with UO_2 pellets and 20 guide tubes (16x16-20). The clad of such fuel elements is made of Zircalloy-4, the end fittings (top fitting and fuel element carrier) are made of stainless steel.

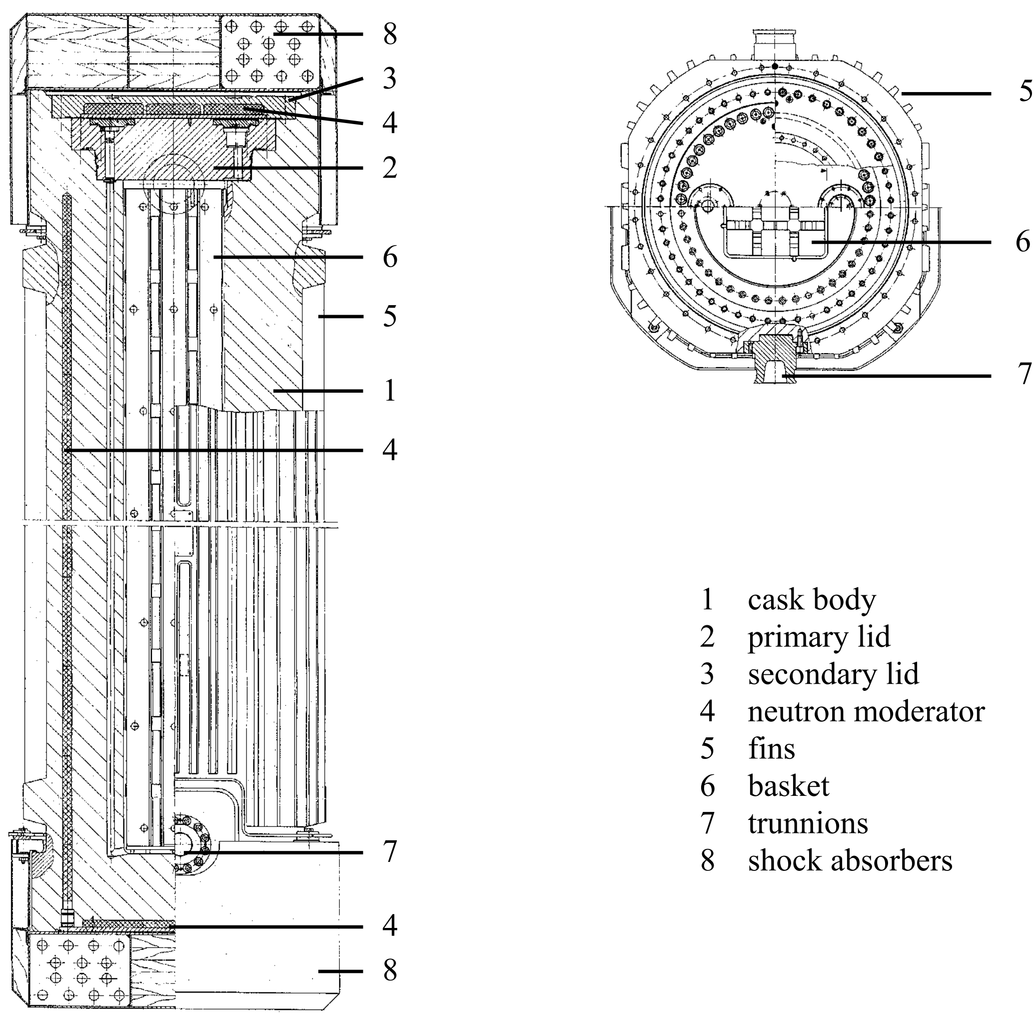

Figure 1: Main parts of the CASTOR IIa cask

The other parameters as well as the irradiation history are summarized in Tables 1 and 2.

Table 1: Main parameters of the fuel assembly

UO_2 density	g/cm³	10.35
diameter of fuel pellet	mm	9.08
inner diameter of clad	mm	9.3
outer diameter of clad	mm	10.75
rod pitch	mm	14.3 x 14.3
active fuel length	mm	3900
max. mass of top fitting	kg	14.54
max. mass of fuel element carrier	kg	13.85
max. cobalt concentration of end fittings	mg/kg	2000

Table 2: Irradiation history for all fuel assemblies

			mean	deviation
U_{tot}		kg	537.3	+1.2
				-0.8
Enrichment		%	2.507	+0
				-0.01
1. cycle	Time	days	397	
	Power	MW/fuel ass.	22.76	+0.16
				-0.22
	cooling time	days	30	
2. cycle	Time	days	344	
	Power	MW/fuel ass.	17.15	+0.21
				-0.20
	cooling time	days	420	
4. cycle	Time	days	299	
	Power	MW/fuel ass.	8.17	+0.29
				-1.93
	cooling time	days	2295	

DESCRIPTION OF THE INPUT FOR THE CALCULATION WITH SCALE

For the source term calculation with SAS2 all necessary data are listed in Tables 1 and 2. In addition some non-negligible parameters during reactor operation are used. These are:

- Temperature of the moderator (subcooled water) 583 K
- Mean boron concentration of the moderator 500 mg/kg
- Mean pressure 160 bar
- Mean density of the moderator 0.706 g/cm³

The recommended 44-Group ENDF/B-V library for cross-section data and default SCALE values for all other parameters were used.

In the SAS4 input file the geometry of the casks upper half and the fuel elements were modelled using the MARS geometry (see Fig. 2) and reflected according to SAS4 requirements. The source strength was derived from SAS2 calculations taking into consideration the axial burn-up profile as it is shown in Fig. 3 and the activation of cobalt of the end fittings.

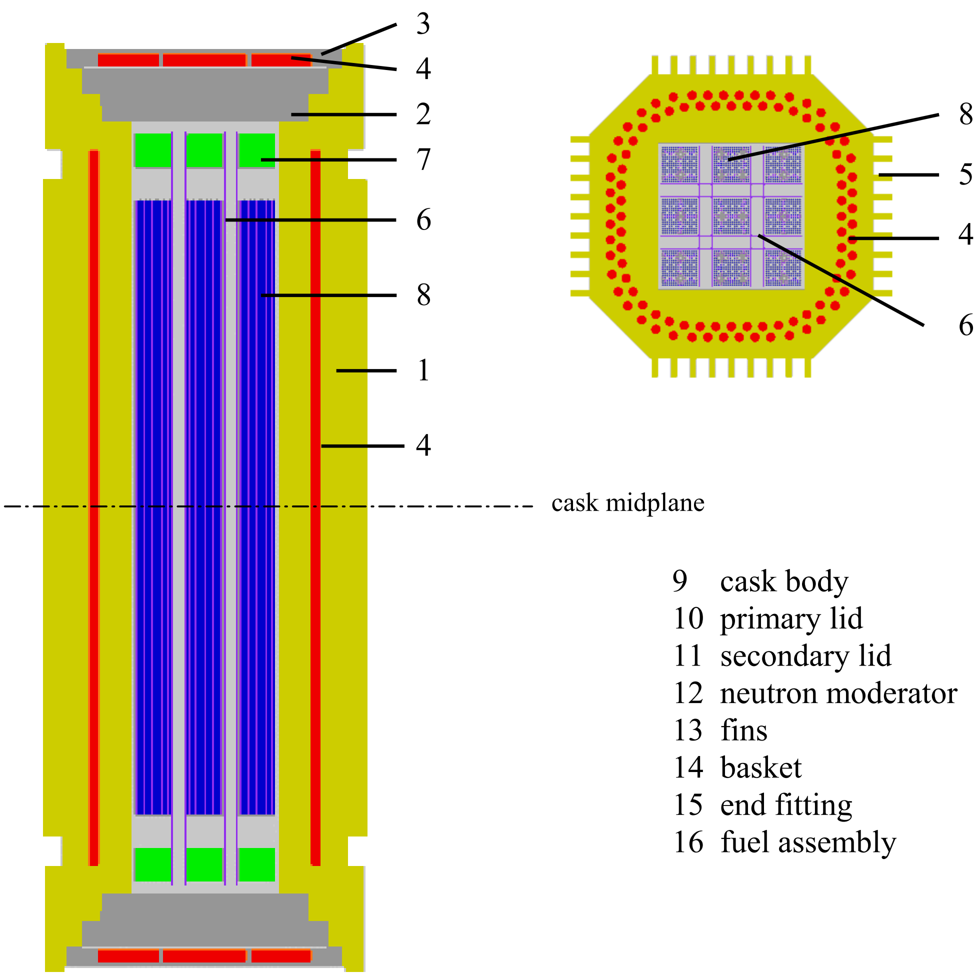

Figure 2: Cask model used for SAS4

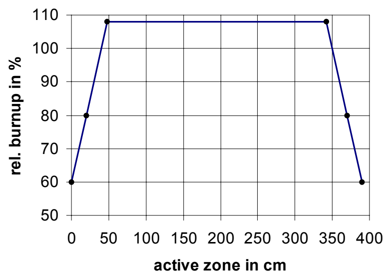

Figure 3: Axial burn-up profile used for SAS4

For the shielding calculation the coupled 27 neutron group, 18 gamma-ray group library which is based on ENDF/B-IV cross section data was chosen.

To obtain the dose rates at the interesting points outside the cask (Fig. 4) both possibilities of SAS4, the point detectors and the surface detectors, were used, but only the surface detector results are listed in Table 3.

In the case of neutron radiation the built-in fluence-to-dose conversion factors for the calculation of the 'maximum dose rate equivalent' $\dot{\hat{H}}$ according to ICRP 21 [3] were used. To obtain the 'ambient dose rate equivalent' $\dot{H}^*(10)$ according to ICRP 60 [4] conversion factors based on ICRP 74 [5] were used which represent the maximum conversion factor within the appropriate SCALE neutron energy group.

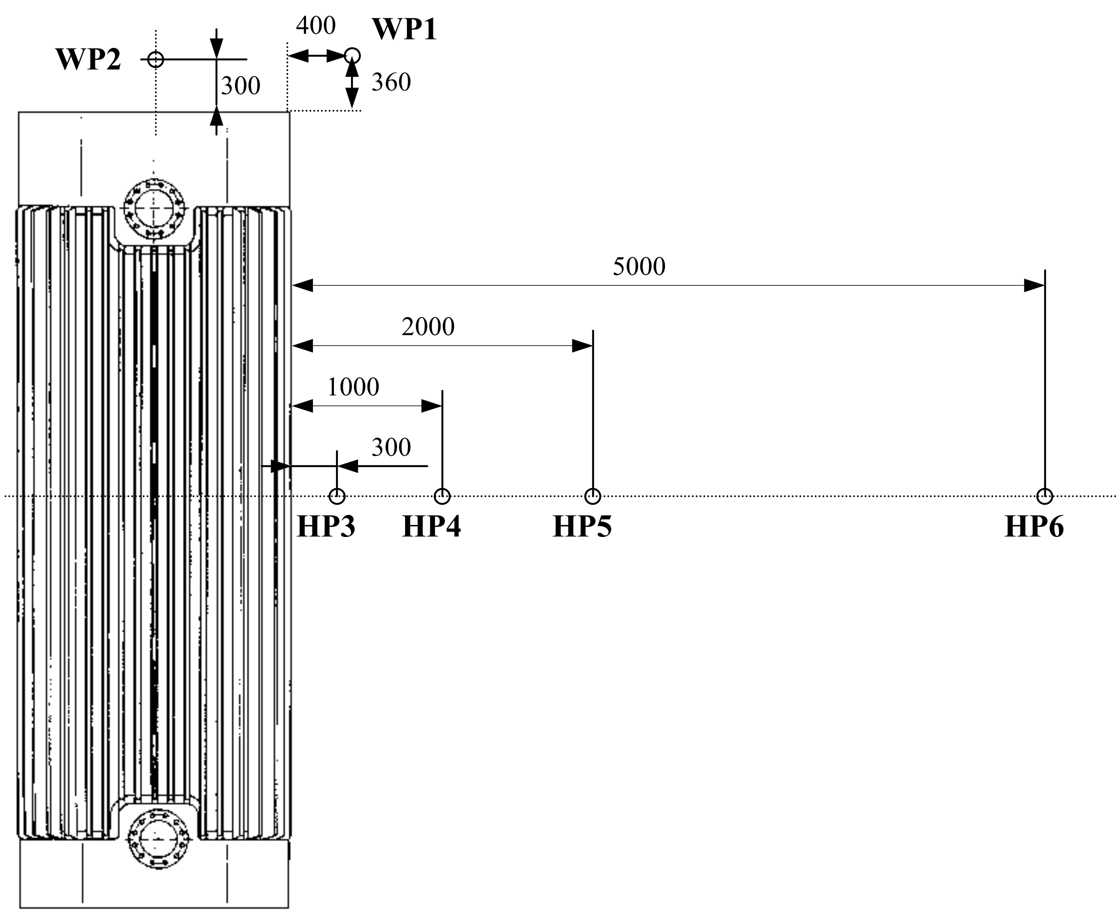

Figure 4: Measuring points

Table 3: Comparison of calculated and measured dose rates

MP	Neutron Dose Rate [1)]					
	ICRP 21			ICRP 60		
	$\dot{\hat{H}}_{calc}$ [µSv/h]	$\dot{\hat{H}}_{meas}$ [3)] [µSv/h]	$\frac{\dot{\hat{H}}_{calc}}{\dot{\hat{H}}_{meas}}$	$\dot{H}^*(10)_{calc}$ [µSv/h]	$\dot{H}^*(10)_{meas}$ [3)] [µSv/h]	$\frac{\dot{H}^*(10)_{calc}}{\dot{H}^*(10)_{meas}}$
WP1	1.6	4.2	0.38	3.5	5.9	0.59
WP2	3.8	4.6	0.83	6.2	6.9	0.90
HP3	47.6	51.0	0.93	74.0	77.1	0.96
HP4	25.1	26.4	0.95	39.2	39.8	0.98
HP5	13.0	13.7	0.95	21.0	20.6	1.02
HP6	3.6	3.8	0.94	5.8	5.6	1.03

MP	Gamma Dose Rate [2)]		
	$\dot{D}_{calc}$ [µSv/h]	$\dot{D}_{meas}$ [3)] [µSv/h]	$\frac{\dot{D}_{calc}}{\dot{D}_{meas}}$
WP1	0.11	0.4	0.28
WP2	3.3	0.5	6.6
HP3	9.3	12.6	0.74
HP4	7.4	8.4	0.89
HP5	5.2	4.8	1.09
HP6	2.2	1.4	1.6

maximum uncertainties:

1) calculation: ± 10%
   measurement: ± 15%

2) calculation: ± 20%
   measurement: ± 20%

3) measured values taken from part A of [6]

RESULTS

With the above mentioned data and techniques the dose rates for gamma and neutron radiation at the various measuring points according to Fig. 4 were calculated.

While the calculated neutron dose rates agree very well with the measured ones at all distances at the midplane (ICRP 21: within 7% below the measured values; ICRP 60: ± 4%) the calculated gamma dose rates show greater variations for the different points. Near the cask (0.3m and 1m) the measured dose rates are underestimated by the calculated dose rates by up to 26%. It may be one reason for this deviation that structural material in the active zone of the fuel element was ignored in the model for the fuel elements. For the larger distances from the cask the calculated gamma dose rates are higher than the measured dose rates by up to 60%. The main reason should be the excessive dose rate portion due to end fittings by using maximum cobalt concentration in the steel.

The end fittings get a greater influence on the total gamma dose rate with growing distances with 9% at 2m and 24% at 5m, respectively.

Since for the calculations the upper limit of the cobalt concentration in the fuel hardware was taken, too high gamma dose rates at the lid center (WP2) were calculated. Further, the cask model does not include a so called protection plate made of steel with a thickness of approximately 80mm that was present during the measurement. This would reduce the dose rate at this point by a factor of approximately 5 to 6. Except that the greater distance between the secondary lid to the measuring point was ignored, these effects do not have a great influence on the neutron dose rate so that sufficient good results were calculated with 83% and 90% of the measured values with conversion factors according to ICRP 21 and ICRP 60, respectively.

The only point where the calculated dose rates differ significantly from the measured ones is WP1. The main reason for these deviations should be that the cask and its content were primarily modelled to calculate the dose rates along the cask midplane. Therefore, parameters having small impact on midplane dose rates as the axial position of the active zone and the burn-up profile at the ends of the active zone have not been investigated exactly. These parameters could, however, influence the dose rate at WP1.

CONCLUSIONS

The SCALE 4.4a source term and shielding software has been successfully validated for use with CASTOR spent fuel transportation casks specially in the cask midplane. The conservatively chosen ICRP 60 flux-to-dose rate conversion factors for the 44 neutron groups of the SCALE library led to a good agreement with the dose rates measured according to ICRP 60. The calculation of dose rates around the lid requires an enhanced modelling effort in the region of the end of the active zone and the fuel hardware.

REFERENCES

[1] *Börst, F.-M.; Rimpler, A.; Scheib, H.*
Dose Rate Measurements at Several Casks for Transport and Interim Storage.
Proceedings of the 12th Internatinal Conference on the Packaging and Transportation of Radioactive Materia (PATRAM '98), Paris 10-15 May 1998, S. 1064-1073.

[2] SCALE - Standardized Computer Analyses for Licensing Evaluation
SCALE homepage, last modified version from 19.01.2001

[3] *International Commission on Radiological Protection (ICRP)*
ICRP Publication 21
Data for Protection against Ionizing Radiation from External Sources
Pergamon Press, Oxford, 1971

[4] *International Commission on Radiological Protection (ICRP)*
ICRP Publication 60
Recommendations of the International Commission on Radiological Protection
Annals of the ICRP Vol. 21, No. 1-3
Pergamon Press, Oxford, 1991

[5] *International Commission on Radiological Protection (ICRP)*
ICRP Publication 74
Conversion Coefficients for use in Radiological Protection against External Radiation
Annals of the ICRP 26, No. 3/4
Pergamon Press, Oxford, 1999

[6] BfS-ET-24/97
Heimlich, F.H. (Hrsg.)
Messungen im Neutronen- und Gamma-Strahlungsfeld eines beladenen Castor-IIa-Behälters im Transportbehälterlager Gorleben und Vergleich der Meßergebnisse für Neutronen mit Monte-Carlo-Rechnungen
Salzgitter, Januar 1997