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INVESTIGATION OF THE SHIELDING SAFETY PROBLEM OF THE ROMANIAN VVR-S REACTOR SPENT FUEL STORAGE

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ABSTRACT

The paper presents the investigation of the shielding safety problem of the German container CASTOR MTR 2 loaded with the spent fuel of the Romanian VVR-S reactor. The SCALE-4.3 code system, namely the SAS4 module with MORSE, as the shielding Monte Carlo code, based on the (27n-18g)-coupled group energy cross-sections generated from ENDF/B-V file has been used for solving the problem. Gamma and neutron dose rates estimated on the surface of the container, and respectively: surface +1m, +2m, +3m in both radial and axial calculation directions for normal, loading and audit conditions are presented and discussed.

Key Words: spent fuel storage, shielding, Monte Carlo transport

1. INTRODUCTION

Romanian VVR-S reactor was shut down in December 1997 after more then forty years of operation without any major modification from the original Russian design. Lasting next four years, the status of the reactor was “under conservation” and studies and analyses have been performed in order to decide which is the best strategy to be followed: i) up-grading for restarting, or ii) decommissioning. In 2002 the Governmental decision to start the reactor decommissioning was taken and necessary planning activities are now in progress.

The cumulative amount of spent fuel arising during the entire reactor operational period is stored inside away from reactor ponds. Until now no decision has been taken concerning the final disposal of reactor spent fuel elements. There are two possible strategies: i) returning to origin for ever, or ii) emplacement into a geological repository.

For the first variant mentioned above a container licensed for transportation through Europe is necessary. The second one needs a more safety intermediate solution, because spent fuel clad corrosion has already occurred in the ponds. Dual purpose transport/storage CASTOR MTR2 German cask can be a suitable tool for both accounted situations and therefore methods have been developed and calculations have been carried out in order to analyze its compatibility to the specific case.

CASTOR MTR 2 container [1] is a cast iron cylinder with two stainless-steel lids. The interior is completely occupied by a cylindrical aluminum body with seven cylindrical loading channels, one of them is central and the other six are arranged equally spaced around the central one. Loading units carrying the fuel elements will be inserted into each of these loading channels. A loading unit is an aluminum cylinder with one central boron rod and some internal holes tailored in accordance with the geometrical shape of the fuel elements.

Most of the fuel elements of the Romanian VVR-S reactor have a square outer shape, deformed by 1, 2, or 3 beveled-corners and the others have a right square shape. According to these different shapes of fuel elements, there are two types of loading units: Type B carrying four fuel elements with a right square shape, and Type C for six fuel elements with the deformed shape. Independently on the geometrical outer shape, there are two types of fuel elements differing by the rod type: The EK10 fuel element consists of 16 rods with MgO and UO_2 (10% enriched uranium), and the S36 fuel element consists of 15 rods containing a U-Al alloy (36% enriched uranium).

For the evaluation of the possibility of transport/storage of the Romanian VVR-S reactor spent fuel in the CASTOR MTR2 cask, it is necessary to study if the safety criteria are observed. The paper presents the investigation of the shielding safety problem. To be on the safer side, conservative assumptions have been done. Thus the paper presents only the results on S36 spent fuel type inside the C variant of CASTOR MTR2 cask.

2. GAMMA AND NEUTRON SOURCE CALCULATION

The calculation of gamma and neutron sources for entire amount of spent fuel elements of both types i.e. EK10 and S36, has been previously performed [2] by means of ORIGEN-S code [3] applying specific problem dependent libraries defined for the classes of fuel elements. The reliability of the burn-up calculations has been tested by comparison against similar calculations, for the same selected fuel elements, performed by ORIGEN-JR and HELIOS codes in VKTA Rossendorf [4].

Because, up to now the details of the different CASTOR loading are not yet established it has been considered that the entire loading of a cask consists of fuel elements having identical properties with a representative one. Such representative fuel element used in calculations has been defined as follows:

i) for normal situation, when upper limits of the expected dose rates are desired to be calculated, the dummy element is characterized by the maximum values of gamma and neutron sources;

ii) for loading and audit conditions, when the detailed loading scheme must be taken into account the conservative criterion is not valuable and therefore an averaged value of both gamma and neutron sources has been taken into account.

Constant radial and axial distributions for active part of the sources have been assumed.

The contribution of the end parts of the fuel assembly has been considered also for axial dose rate estimates over open cask. With a view to obtaining the gamma source arising from the end of fuel element an ORIGEN-S calculation has been performed separately considering only 5g of light structural elements.

All these values used as input data in Monte Carlo shielding calculations have been calculated for 31st December 1999. Upgrading of those data have been done for five supplementary cooling time of 5, 10, 25, 50 and 100 years after the accounted reference time.

3. MODELING AND PERFORMED CALCULATIONS

The very complex and nonstandard geometry of the cask does not allow the elaboration of the calculation model with the help of standard casks available in the SAS4 [5] sequence of SCALE4.3 system and therefore the detailed geometry has been built by using MARS [6] module.

Using this utility a very complicated geometry model has been created as input for Monte Carlo MORSE code [7]. The calculation model used assumes (see Fig. 1) that the materials of fuel elements and their sources have been smeared over the horizontal cross section of the fuel element.

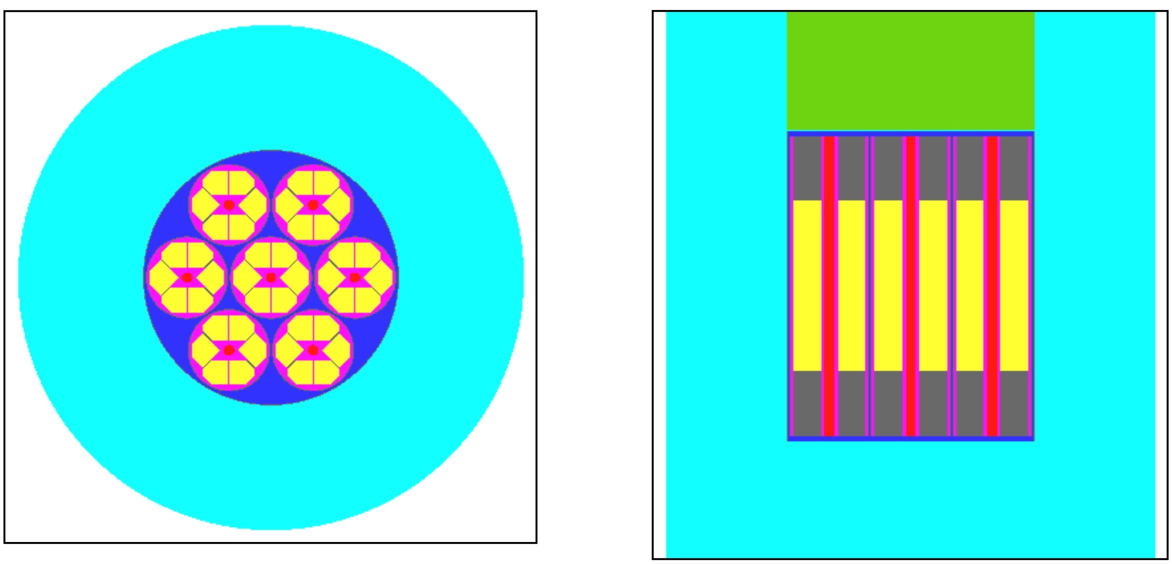

Figure 1 CASTOR MTR2. Geometry model
transversal cross section for z=0 **vertical cross section for x, y =0**

Additionally, other approximations of the calculation model done, causing the overestimating of the dose rates can be characterized shortly:

i) the wall of the fuel assembly has been added to the surrounding loading unit, hence the smearing of the fuel materials has been adapted for the remaining surface;
ii) symmetric homogen cylinders with 5 cm height have been modeled of the ends of the assembly and the corresponding remained space in the end zones has been filled with air;
iii) vertical symmetry of the cask has been assumed, replacing the existing slight asymmetry of the cask.

A full characterization of the spent fuel cask has been performed using this calculation model:

i) neutron and gamma doses have been computed in separated calculations based on the SAS4 automated biasing procedure;
ii) different calculations have been performed for radial and axial detectors.

For both gamma and neutron dose rate estimates over the open cask only axial calculations have been performed. A supplementary calculation, for this last configuration has been needed for the estimation of gamma dose contributed by the end part material.

The surface detectors are located radial or axially on the outermost surface of the cask and 1, 2 and 3 m from this outermost surface.
The axial detectors are circular discs, while the radial detectors are side surfaces of cylinders having appropriate radii and heights given by the height of active part of the fuel element.

The ANSI standard flux-to dose conversion factors have been used. These response factors correspond to the coupled library (27 neutrons-18 gamma) energy groups used for all the calculations performed.

4. RESULTS AND DISCUSSIONS

The obtained results for both closed and open cask configurations are presented in Table I and II.

Table I. Maximum dose rate values outside the CASTOR MTR 2 cask

	Total sources		**Dose Rates** **Surface**	**+1m**	**+2m**	**+3m**
			μSv/h FSD*	μSv/h FSD*	μSv/h FSD*	μSv/h SD*
γ	[γ/s] 9.7142+14	**Radial**	68.63 0.02	6.32 0.02	3.22 0.02	1.89 0.02
		Axial	23.32 0.04	4.03 0.04	1.85 0.05	0.97 0.05
n_0[1]	[n/s]	**Radial**	2.45-1 0.01	2.19-2 0.01	9.59-3 0.01	5.21-3 0.01
	7.8246+4	**Axial**	2.73-1 0.01	2.51-2 0.01	8.78-3 0.01	4.33-3 0.01

*FSD fractional standard deviation

Table II. Dose rate values over the open CASTOR MTR 2 cask

	Total Sources	**Dose rates** **Surface**		**+1m**		**+2m**		**+3m**	
	[γ/s]	Sv/h	FSD*	Sv/h	FSD	Sv/h	FSD	Sv/h	FSD
γ	**Active part** 5.8997+14	1.88	0.05	1.22	0.02	0.64	0.04	0.35	0.05
	End parts 1.5107+9	2.29-5	0.02	1.28-5	0.01	6.37-6	0.02	3.73-7	0.02
n_0^1	[n/s] 3.1391+4	3.66-7	0.01	9.74-8	0.04	4.08-8	0.01	2.20-8	0.02

* FSD fractional standard deviation

The statistical errors (under 5%) are comparatively insignificant to the more important errors that come from modeling limitations. The error due to the smearing of the fuel materials is about 30% [8]. The done approximations for radioactivity and further gamma and neutron sources ORIGEN-S calculation induce an error of about +/-15% per fuel element. But, the large number of fuel elements of the cask loading (42) allows the compensation of these errors arising from uncertainties given by the various positions of the fuel elements during irradiation.
The values of dose rates exterior to the cask charged with VVR-S spent fuel (see Table I) are below the limit i.e. 2mSv h^{-1}, accepted by the National Regulations for Nuclear Safety.

In comparison with the gamma dose rates arising from active part of the fuel elements, those that came from the end part materials are negligible (see Table II).

5. CONCLUSIONS

It is concluded from this computational study that from the point of view of the shielding safety, the CASTOR MTR 2 cask offers a safe containment of the Romanian VVR-S reactor spent fuel.

The obtained results assist to the achievement of the working procedures that must be observed during the loading of cask as well as for audit activities.

The conservative approximations made must be taken into account in order to have more precise basic input needed to establish the conditions of personnel operation in the working area and to assure the safety of staff upon the ALARA principles.

REFERENCES

1. B. Schneider, G. Hofmann, “Konditionierung bestrahlter Brennenelemente aus dem Rossendorfer Forschungsreaktor RFR in Transport–und Lagerbehalter CASTOR MTR-2,” *Rep VKTA-17*, September (1994

2. D. Ene, F. Tigau, G. Cambi, D. G. Cepraga, "Production and application of activation data libraries for VVR-S Reactor Spent Fuel Characterization," *Proceeding of the Sixth International Conference on Nuclear Criticality Safety*, Versailles, France, 20-24 September, pp 1735-1745 (1999)
3. O. W. Hermann, R. M. Westfall, "ORIGEN-S: SCALE System Module to Calculate Fuel Depletion, Actinide, Transmutation, Fission Product, Decay and Radiation Source Term," *NUREG/CR-0200, Rev. 5, Vol.2, Section F7, ORNL* (1997).
4. D.Ene, E. Franke, "Comparison of different computational systems for calculation of the VVR-S IFIN-HH reactor spent fuel characteristics," *Proceeding of the AER Working Group Meeting*, Vuje Trnava, Slovakia, 23-25 April, Printing in progress (2001)
5. J. S. Tang, "SAS4- A Monte Carlo Cask Shielding Analysis Module Using Automated Biasing Procedure", *NUREG/CR-0200, Vol.2, Section S4, ORNL* (1997)
6. J. T. West, M. B. Emmet, "MARS: A Multiple-Array System using Combinatorial Geometry," *NUREG/CR-0200, Vol. 2, Section M9, ORNL* (1997)
7. J. T. West, T. J. Hoffman, Emmet, M.B., "MORSE-SGS for the SCALE System," *NUREG/CR-0200, Revision 5, Vol. 2, Section F9, ORNL* (1997)
8. E. Franke, E. Seifert, "Berechnung der Dosisleistung uber den nach oben offenen CASTOR MTR-2 Behaltern mit Rossendorfer Brennenelementen," *Arbeitsbericht KWK-1/97* (1997)